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ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60439

PRELIMINARY SAFETY ANALYSIS REPORT
ON THE
ARGONNE ADVANCED RESEARCH REACTOR

by

L. W. Fromm
J. F. Matousek
R. R. Rohde
D. H. Shaftman*
J. H. Tessier

M. Levenson, Project Manager

Compiled and Edited

by

I. Charak, Safety Coordinator

Reactor Engineering Division

June 1968

*Reactor Physics Division

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Preface

On April 6, 1966, the first draft of the Preliminary Safety Analysis Report (PSAR) on the Argonne Advanced Research Reactor (AARR) was submitted to the AEC for review. As a result of meetings with and reviews by the Chicago Operations Office (CH), the Division of Reactor Development and Technology (DRDT), and the Division of Reactor Licensing (DRL), numerous questions, which required answers and/or amplification by ANL, were officially transmitted to ANL on September 12, 1966. The reactor under review at that time was of the flux-trap type with a UO₂-stainless steel cermet core.

On October 1, 1966, the AARR Project effort was redirected to employ the Oak Ridge National Laboratory (ORNL) High Flux Isotope Reactor (HFIR) core and associated internals (to the maximum extent possible), and termination of all efforts on the core package described in the PSAR was effected. Due to the fact that the HFIR Accident Analysis and its review had already been well-documented by ORNL, and due also to the ever-present problems of schedule and cost, it was decided to utilize to the fullest extent possible all existing HFIR documentation pertaining both to design description and accident analyses where the HFIR design was pertinent to the discussion. It was recognized, however, that ANL would continue to review the ORNL analyses throughout the evolution of the final design, and that any additional analyses required would be reported in the AARR Final Safety Analysis Report.

Subsequent to the HFIR core adoption for the AARR, review of the PSAR by the AEC continued, resulting in extensive additional documentation by the Project Staff. During the PSAR review, it became apparent that many facility design details pertinent to safety could not and would not be available until after the start of construction. Hence, the Commission agreed to review the PSAR in two stages. The first stage consisted of a review of all the material in this report. As a result of this documentation, the AEC's Advisory Committee on Reactor Safeguards (ACRS) concluded that, "... it is the opinion of the ACRS that there is reasonable assurance that a reactor facility of the type proposed (AARR) can be constructed and operated at the Argonne National Laboratory site without undue hazard to the health and safety of the public." (See Part 2, Appendix A.)

The second stage of review was to consist of a review of the design details of the containment structure, reactor vessel and safety-related instrumentation when they became available. However, the Project was cancelled on April 2, 1968, thereby terminating all future work on AARR and associated documentation. It is, nevertheless, considered prudent to publish the present PSAR so that a single unified report will exist for the record, and so that the efforts expended in the writing of this document may be of possible use to others.

The design of the facility described in Section IV, Part 1 of this report is that which existed on September 1, 1967, at which time the detailed facility design was about 50% complete. The bulk of the report, however,

was not revised to be consistent with Section IV, and hence there may be some cases where disagreement exists between Section IV and other parts of the text. In all cases, however, Section IV is considered to accurately portray the facility design at the time of the ACRS approval.

Part 1 of this report consists of all the writing which was originally transmitted to the AEC for review (and periodically revised for currency). Part 2 is a compilation of all formally-transmitted AEC questions and the corresponding ANL answers. Editorial license has been exercised only to the extent of adding question numbers for easy reference and making the questions and answers chronologically self-consistent. Many of the answers refer simply to the text of Part 1 which had been extensively revised at the time of the core change to include answers to specific questions where it was deemed practicable.

Due to the wide-ranging scope of the questions, it is not considered feasible to categorize them. However, a keyword-type index to the questions is included in Part 2 for easy reference.

The Editor

Major contributions to this report were made by *
the following members of the AARR Project Staff:

R. P. Anderson (RE)	J. J. Dickson (RE) [†]	E. L. Martinec (RE)
J. F. Bartusek (PE)	V. J. Elsbergas (EL)	M. Musko (PE)
C. F. Bremigan (PE)	A. P. Gavin (RE)	C. V. Pearson (RE)
R. C. Brubaker (RE)	H. H. Hooker (RE)	H. B. Perkins (PE)
N. Bulut (PE)	A. R. Jamrog (RE)	A. W. Pierce (RO)
T. C. Cameron (RE)	W. J. Kann (RE)	D. F. Pozar (RE)
I. Charak (RE)	V. M. Kolba (RE)	R. P. Savio (RO)
E. J. Ciesla (RO)	B. N. Kristianson (RE)	R. W. Seidensticker (RE)
R. G. Cohen (PE)	M. Levenson (RE)	R. C. Skaardal (RE)

The very valuable advice and assistance rendered
by the following Laboratory personnel other than
members of the AARR Project Staff are gratefully
acknowledged:

T. C. Acierno (PE)	D. A. Kucera (RP)	H. F. Reed (RP)
M. A. Bognar (RPY)	L. Leibowitz (CEN)	G. S. Rosenberg (RE)
T. R. Bump (RE)	D. H. Lennox (RE)	A. D. Rossin (MET)
J. E. Carson (RPY)	P. A. Lottes (RE)	D. J. Roy (CS)
Y. W. Chang (RE)	D. L. Matson (COM)	U. K. Sick (PE)
G. Cinelli (RE)	A. E. McCarthy (RP)	H. C. Stevens (PRO)
W. D. Dillow (IHS)	J. F. Mech (PRO)	G. C. Vellender (RO)
H. K. Fauske (RE)	D. Mohr (RE)	R. J. Weatherhead (RE)
B. M. Hoglund (RE)	H. O. Monson (LDO)	R. E. Wilson (CEN)
C. N. Kelber (RP)	H. Moses (RPY)	

* Organization Abbreviations:

CEN - Chemical Engineering Division
CS - Central Shops
COM - Communications and Special Services Department
EL - Electronics Division
IHS - Industrial Hygiene and Safety Division
LDO - Laboratory Director's Office
MET - Metallurgy Division
PE - Plant Engineering Division
PRO - LMFBR Program Office
RE - Reactor Engineering Division
RO - Reactor Operations Division
RP - Reactor Physics Division
RPY - Radiological Physics Division

[†] Deceased

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PART 1
PRELIMINARY SAFETY ANALYSIS REPORT
ON THE
ARGONNE ADVANCED RESEARCH REACTOR

I. INTRODUCTION

A. Purpose of the Facility

The Argonne Advanced Research Reactor (AARR) is a multi-purpose research tool which will provide scientists of Argonne National Laboratory and midwestern universities with facilities necessary to conduct basic nuclear physics, nuclear chemistry, and solid state science research programs to levels of advancement considerably beyond those possible in the CP-5 Research Reactor and other lower power research reactors currently available at the Argonne site.

The principal experiment facilities provided are the beam tubes having thermal neutron fluxes approaching 1×10^{15} neutrons/cm²-sec at their innermost tips, and an internal thermal column wherein an unperturbed flux approaching 5×10^{15} neutrons/cm²-sec is available, at 100 MW power level. The internal thermal column will have suitable sample transfer facilities for the preparation and rapid examination of transplutonium and transcurium elements as well as for long-term high flux irradiation. The beam tubes will be used to conduct neutron physics and solid state experiments. Vertical irradiation facilities, laboratories, and a rabbit-handling pool are also provided to afford experimenters a wide spectrum of research potentialities and to assure maximum utility of the reactor.

B. History and Status of the Project

For a number of years Argonne National Laboratory has recognized a steadily increasing need for a source of higher neutron fluxes to permit the logical extension of its basic research programs; accordingly, it has carried on a program to develop design concepts for general purpose high flux research reactors to fill this need.¹⁻⁴ The current design concept for the Argonne Advanced Research Reactor, as described in this report, is the result of that development program.

In 1962 and 1963, in conjunction with the Laboratory's studies and based on the AARR design concept at the time, United Nuclear Corporation performed two supporting studies, a preliminary feasibility and cost study, and later a revised preliminary cost study to bring the earlier cost estimate into line with later concept modifications. Technical results of these studies have been factored into the design as presented herein.

An earlier concept⁴ for AARR called for use of an aluminum base core having a life of from 10 to 14 days at 100 MW. Subsequent work, however, led to the adoption of a UO₂-stainless steel cermet core in order to gain such advantages as significantly longer core life (around 90 days at 100 MW), virtual elimination of the shutdown xenon problem during most of

the core cycle, reduction in the required canal storage capacity for spent cores, and a greater potential for future reactor operation at power levels significantly higher than 100 MW. However, in September 1966, in order to effect a considerable saving in research and development cost, the program of developing the UO_2 -stainless steel core was dropped and a decision was made to utilize in AARR the U_3O_8 -aluminum involute plate core design already developed and in use in the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory, as well as HFIR designs for closely associated reactor components such as the control plates, control plate drive mechanisms, inner removable and semi-permanent beryllium reflector, and various related supporting structures. Since that date the design effort and supporting development on AARR have proceeded on this basis. This Preliminary Safety Analysis Report, therefore, is based on a 100 MW AARR plant utilizing the HFIR core and associated reactor components.

Concurrent with the development of the AARR design criteria, preliminary design concept and Title-I design, supporting research and development programs have been in progress. Prototypes of critical plant components are being designed and fabricated for test in order to verify design and reliability. These components include the beam and through tube internal assemblies, and irradiation facility mechanisms. Theoretical physics studies, heat transfer investigations involving some experimentation, and reactor dynamics and stress studies are being carried out, with particular emphasis on use of the HFIR core to fulfill AARR functional goals.

Since the final design is contingent both upon the results of the research and development program (of which those portions important to safety are outlined in Appendix D), and on further design development by the Laboratory and the architect-engineer, modifications to the design as described in this report may become necessary. However, it is not expected that such changes will have a significant effect upon the tentative parameters and general design approaches used in the current design; it may therefore be assumed that an assessment of the feasibility of construction of the plant, based upon the current design presented herein, will remain valid as the detailed engineering design is developed. The principal safety and design criteria delineated in this report will serve as the basis for design modifications from the current concept, and will not be compromised in any case.

In summary, it may be stated that the current design concept, taken as the basic design of the plant for assessment of construction feasibility, together with the criteria which provide the rules under which modifications may be made to arrive at the final detailed design, constitute an adequate basis for evaluating the safety analyses presented in this Preliminary Safety Analysis Report.*

*Ed. note: The Title-I design for the plant had been completed by the architect-engineer, and the Title-II design was nearing completion when the project was terminated on April 2, 1968. At that time, excavation and rough grading had been completed and construction was expected to begin in FY 1969. The reactor vessel, primary coolant pumps and certain vessel internal components were under design and/or construction.

C. Report Objective

This report has the objective of demonstrating that the Argonne Advanced Research Reactor may be constructed at the DuPage County, Illinois site of Argonne National Laboratory and operated at power levels up to 100 MW thermal without undue hazard either to the general public or to employees of the Laboratory, provided the facility fulfills the safety criteria presented, and is designed in general accordance with the current design concept.

The report fulfills this objective in the following manner:

1. The safety and design criteria for the facility are delineated. Rather than specifying particular numerical design values for the various plant parameters, these criteria represent the rules according to which the numerical parameters and design features of the current design concept will be adjusted in order to achieve a safe and practical final design.
2. The site upon which construction of the facility is planned is described in detail.
3. The current design for the facility is described, and tentative numerical parameters important to safety are presented. Although the parameters and design features delineated cannot yet be considered final, the design development has proceeded sufficiently that no major modifications in design principles or in general parameter levels are anticipated.
4. The general administrative philosophies now planned to govern operation of the facility are outlined.
5. An analysis of credible accidents is presented, assuming pertinent features of the current design as the basis for analysis, and the means are indicated by which such accidents can be effectively prevented or the consequences mitigated if the stated safety criteria are followed in design.
6. The worst accident deemed credible in the facility (the "Maximum Credible Accident," or MCA) is defined and described, and it is demonstrated that the normal population of the Laboratory site and the general public will be adequately protected in such an eventuality.

The HFIR core and related components which are to be duplicated in AARR have now operated successfully in the HFIR at full design power (100 MWt) through several core cycles, and most of the safety features of the plant have been tested under actual operation. For example, the facility has been subjected to at least one total failure of utility electric power without incident. Although many of the system and plant design features of AARR differ from those in HFIR because of the differences in experimental function of the two facilities, the principles upon which the design of AARR is being based are either the same as those employed for HFIR, or are more

conservative. Operating conditions such as core inlet coolant temperature, pressure and flow, and power level will be identical in the two facilities. Based on the HFIR operating experience, therefore, it is certainly a reasonable expectation that AARR can indeed be designed and constructed to operate under these conditions without undue hazard either to on-site or off-site personnel.

Because the decision to utilize the HFIR core and related components in AARR was made late in the Project history, the Laboratory has not had the opportunity to conduct entirely independent safety analyses on all aspects of the revised facility; indeed, because of the successful operation of HFIR, the Laboratory does not believe this to be necessary at the preliminary safety analysis stage. Consequently, this report depends heavily on much of the safety analysis work performed by Oak Ridge National Laboratory on the HFIR and reported in the final safety analysis report (including supplements and addenda) on that facility.⁵⁻⁸ In many areas, descriptive and analytical material have been extracted verbatim (except for table, figure and reference numbers) and reproduced herein for the convenience of the reviewer. Such direct extracts are identified throughout the report by indentation from the main body of the text, and by references where appropriate.

References for Section I

1. L. E. Link, et al., "The Mighty Mouse Research Reactor Preliminary Design Study," ANL-5688 (March 1957).
2. L. E. Link, et al., "Terminal Report on the Mighty Mouse High-Flux Research Reactor Project," ANL-5928 (September 1959).
3. L. E. Link, et al., "Argonne High-Flux Research Reactor-AHFR Conceptual Design Study," ANL-5983 (June 1959).
4. D. H. Lennox, et al., "Status Report on the Argonne Advanced Research Reactor," ANL-6451 (November 1961).
5. F. T. Binford, "The High Flux Isotope Reactor-Accident Analysis," ORNL-3573 Draft (February 1965).*
6. T. E. Cole to A. M. Weinberg, "High Flux Isotope Reactor-Safety Review Questions and Answers," Memorandum ORNL-CF-65-11-29 (November 12, 1965).*
7. T. E. Cole to A. M. Weinberg, "High Flux Isotope Reactor-Safety Review Questions and Answers," Memorandum ORNL-CF-65-11-29 Supplement No. 1 (February 1, 1966).*
8. T. E. Cole to A. M. Weinberg, "High Flux Isotope Reactor-Safety Review Questions and Answers," Memorandum ORNL-CF-65-11-29 Supplement No. 2 (August 15, 1966).*

*These documents have been combined into a single report, ORNL-3573, April 1967.

II. PRINCIPAL SAFETY AND DESIGN CRITERIA

In establishing the safety and design criteria for the AARR facility, the 27 general design criteria used by AEC in judging the safety of proposed nuclear power plants (issued as an attachment to AEC Press Release No. H-252, dated November 22, 1965) have been used as a guide. Under each of the categories of criteria presented below, the appropriate AEC criterion has been reproduced, for convenience, and related AARR criteria follow. These latter criteria, together with functional requirements, for the most part constitute the basis for design of the AARR facility.

A. Facility

1. General Quality and Performance Standards

AEC Criterion 1

Those features of reactor facilities which are essential to the prevention of accidents or to the mitigation of their consequences must be designed, fabricated, and erected to:

- (a) Quality standards that reflect the importance of the safety function to be performed. It should be recognized, in this respect, that design codes commonly used for nonnuclear applications may not be adequate.
- (b) Performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces imposed by the most severe earthquakes, flooding conditions, winds, ice, and other natural phenomena anticipated at the proposed site.

AARR Criteria

- 1.1 All features of the AARR facility which are required for the prevention of accidents or for protection of the general public or Laboratory employees after an accident will be designed, fabricated and erected in accord with quality standards which reflect the importance of the safety function to be performed.
- 1.2 The reactor primary system components and all engineered safeguards will be designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III -- Nuclear Vessels, or the ASA Code for Pressure Piping, whichever is applicable.
- 1.3 The instrumentation, control systems and safety interlock components will be fabricated using the highest quality materials and workmanship commensurate with the application.

- 1.4 The AARR facility is to be located in a region of seismic probability designated as Zone I in the Uniform Building Code. All structures, systems and components important to safety (Class I)* will be designed such that the plant will maintain its capability of shutting down, removing decay heat and being safely contained, and that malfunctions will not tend to increase the core reactivity. The necessity of performing a dynamic analysis of each Class I component will be determined, and the work performed, by a recognized seismic expert. The earthquake forces will be considered to act simultaneously with dead load, live load and pressure and temperature load resulting from the MCA (see Criteria 17.2 and 17.5). The maximum stresses produced by the above loadings will be combined and appropriately related to the yield stresses of the materials.
- 1.5 The Reactor Containment Building structure, excluding insulation and roof covering, will be designed to withstand the total force that can be developed by a wind with a basic velocity of 300 mph at 30 feet above grade and applied to the projected elevation of from 0 to 50 feet above grade and for the total force developed by a wind with a basic velocity of 340 mph applied to the projected elevation of from 50 to 75 feet above grade.
- 1.6 The facility will be designed so that all components required for safe, orderly plant shutdown, including the reactor control room and emergency power supplies, will be located either within reinforced concrete structures or below ground, so that their function will be unaffected by wind velocities up to 300 mph. In addition, the facility will be designed so that all components required for the removal of decay heat from the reactor after such a shutdown will be similarly protected.
- 1.7 The Reactor Containment Building will be designed, fabricated and initially tested consistent with the intent or spirit of the ASME Boiler and Pressure Vessel Code, Section III - Nuclear Vessels, and in accordance with applicable sections of the ACI (318-63) and AISC codes.

*Ed note: This report makes mention of "Class I" in two different ways. In this case, Class I refers to components which must survive an earthquake. In later sections of this report, the words "Class I will be applied to the uninterrupted d.c. power supply to vital motors and instrumentation.

2. Chemical Reactions

AEC Criterion 2

Provisions must be included to limit the extent and the consequences of credible chemical reactions that could cause or materially augment the release of significant amounts of fission products from the facility.

AARR Criteria

- 2.1 The Maximum Credible Accident upon which the design of the Reactor Containment Building is based will include the release to the containment of the energy produced by the maximum metal-water reaction considered credible in the reactor core, as well as the energy produced by combustion of the hydrogen evolved in the reaction. (See Criterion 17.5).
- 2.2 Special design precautions will be taken to prevent any significant metal-water reaction in the reflector as a result of any credible accident.
- 2.3 Care will be taken in designing partitioning, shielding enclosures, etc., in the Reactor Containment Building to avoid pockets in which hydrogen evolved in a metal-water reaction and released from a primary system leak or rupture or from the primary off-gas system, may collect to form explosive mixtures with air. Special precautionary measures (such as inert gas atmospheres) will be considered in sealed areas enclosing portions of the primary coolant system, if any such areas exist.

3. Missile Protection

AEC Criterion 3

Protection must be provided against possibilities for damage of the safeguarding features of the facility by missiles generated through equipment failures inside the containment.

AARR Criteria

- 3.1 Filler plugs to be inserted in unused beam port facilities will be designed such that no missiles which could reach and damage the Reactor Containment Building walls will result from failure of the beam tube internals.
- 3.2 Beam tube experimental equipment and other equipment in pressurized systems will be examined for missile generation potential. If a possibility of missile generation exists, local missile protection in the form of concrete walls, blast shields, etc., will be provided around the equipment.
- 3.3 After the configuration of the top of the reactor vessel, the reactor shield pool, and the containment proper have been finally determined, a thorough analysis of the potential for missile generation and for containment damage will be made. Missile protection and/or hold-down of potential missiles will be provided if needed.
- 3.4 Primary system pipe supports will be designed to withstand reaction forces associated with pipe ruptures ranging in severity up to the complete severance of a main primary system circulation pipe to the extent necessary to prevent motion which could jeopardize the integrity of the containment building.

B. Reactor

4. Anticipated Deviations from Normal Steady State Conditions

AEC Criterion 4

The reactor must be designed to accommodate, without fuel failure or primary system damage, deviations from steady state norm that might be occasioned by abnormal yet anticipated transient events such as tripping of the turbine-generator and loss of power to the reactor recirculation system pumps.

AARR Criteria

- 4.1 The design pressure for the reactor primary system components will be no less than 110 percent of the normal operating pressure for the system, plus 50 psig.*
- 4.2 The design temperature for the reactor primary system components will be at least 50°F higher than the normal operating temperature of the system. In components wherein irradiation heating of structural members is anticipated, the normal operating temperature, for this criterion, will include any temperature increase caused by such heating.*
- 4.3 The design overpower for the reactor will be no less than 115% of nominal full power.
- 4.4 The core design and normal operating parameters will be such that under normal operation and under abnormal but anticipated conditions, e.g., utility power failure, the hot-spot heat flux will not exceed that value at which the onset of nucleate boiling occurs.
- 4.5 All operating parameters important to safety will be monitored. Reactor scram will be initiated automatically by the safety system when any appropriate parameter exceeds a preset safe limit. (Criterion 15.1).
- 4.6 Normal primary coolant flow will be maintained by at least two operating primary coolant pumps, to eliminate the possibility of a single pump failure resulting in complete loss of flow. A similar provision will be made in the secondary coolant system.
- 4.7 Sufficient redundancy will be provided in the primary pressurization system to assure that a single pump failure cannot result in loss of the system function.

* Current design parameters for primary system components are listed in Table IV-A-1.

- 4.8 Suitable provision will be made for pressure maintenance so that in the event of a malfunction other than a pipe rupture, primary system pressure will be maintained for a sufficient period of time to permit the reactor to be shut down prior to the occurrence of nucleate boiling in the reactor core.
- 4.9 A barrier diaphragm will be provided in each horizontal beam and through tube assembly, to minimize the possibility of loss of coolant or reactor pressure in the event of rupture of the main tube thimble within the reactor vessel.
- 4.10 All primary and secondary system valves, and, to the extent feasible, their associated instrumentation and control circuits, will be designed to fail preferentially in a position assuring continued reactor cooling and pressurization.
- 4.11 Emergency cooling, capable of removing all decay heat from the reactor after shutdown, will be provided. (Criterion 10.1 et seq.)

5. Power and Process Variable Transients

AEC Criterion 5

The reactor must be designed so that power or process variable oscillations or transients that could cause fuel failure or primary system damage are not possible or can be readily suppressed.

AARR Criteria

- 5.1 The reactor and plant will be designed so as to minimize the possibility of changes from normal operating characteristics that could lead to an inadvertent increase in reactivity sufficiently large and rapid to cause fuel melting.
- 5.2 The sizes, locations and operating characteristics of experimental facilities, and the range of irradiation sample compositions and sizes, will be limited such that (a) no single failure involving voiding, flooding, and/or breaching of an experimental facility, and/or breaching or motion of the sample, could cause a nuclear excursion of sufficient magnitude to melt fuel or breach building containment; and (b) no accident associated with experimental facilities could result in an accident significantly worse than the Maximum Credible Accident (see Criterion 17.5), or significantly worsen the toxic effects of a MCA on receptors outside the building. All proposed irradiation samples will be rigorously inspected, tested as necessary to ensure that the characteristics are known and that the proposed experiment is safe, evaluated by an appropriately constituted independent review committee, and approved by that committee and by the Facility Manager prior to insertion in the reactor.
- 5.3 One or more safety systems of reactivity control will be provided for rapid injection of negative reactivity in the event that trip limits are exceeded. Initiation of such protective action will be automatic and governed by redundant channels of instrumentation, or manual. (Criterion 8.1; see also Criteria 12.1 et seq. and 15.1 et seq.)

Note: Criteria 4.1 et seq. are applicable here also.

6. Fuel Stamina

AEC Criterion 6

Clad fuel must be designed to accommodate throughout its design lifetime all normal and abnormal modes of anticipated reactor operation, including the design overpower condition, without experiencing significant cladding failures. Unclad or vented fuels must be designed with the similar objective of providing control over fission products. For unclad and vented solid fuels, normal and abnormal modes of anticipated reactor operation must be achieved without exceeding design release rates of fission products from the fuel over core lifetime.

AARR Criteria

- 6.1 The AARR core, which is identical to the core utilized in the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory, will not be operated to burnups in excess of those previously demonstrated in the HFIR facility.

7. Control Rod Worth and Withdrawal Rate

AEC Criterion 7

The maximum reactivity worth of control rods or elements and the rates with which reactivity can be inserted must be held to values such that no single credible mechanical or electrical control system malfunction could cause a reactivity transient capable of damaging the primary system or causing significant fuel failure.

AARR Criteria

- 7.1 The normal rates of reactivity addition by the reactivity control systems will be limited so that even a malfunction leading to continued control plate withdrawal at the maximum rate, up to the point at which the high flux trips actuate the automatic reactivity shutdown system, would not lead to fuel melting. Maximum control plate worths and withdrawal speeds will be specified to satisfy this criterion.
- 7.2 The design of the core shroud and support structure, and the mounting of the core within the reactor vessel and the control plate drive mechanisms exterior to the vessel, will be such that a single failure cannot result in the rapid removal of a shim-safety plate or plates from the core, assuming that the lowermost position of the shim-safety plate is the position of maximum negative reactivity insertion.
- 7.3 The design of the reactivity control system, including the control plate drive mechanisms, will be such that a single failure in the shim-regulating cylinder or its drive mechanism can be adequately compensated by action of the shim-safety plates, in time to prevent a major reactivity excursion.

8. Reactivity Shutdown Capability - Stuck Control Plate

AEC Criterion 8

Reactivity shutdown capability must be provided to make and hold the core subcritical from any credible operating condition with any one control element at its position of highest reactivity.

AARR Criteria

- 8.1 One or more safety systems of reactivity control will be provided for rapid injection of negative reactivity in the event that trip limits are exceeded. Initiation of such protective action will be automatic and governed by redundant channels of instrumentation, or manual. (Criterion 5.3; see also Criteria 12.1 et seq., and 15.1 et seq.).
- 8.2 The reactivity control systems will have the capacity to shut down the reactor at all times during the core life, even if any single component of these systems (e.g., one control plate) is inoperable and in a state of minimum reactivity control effectiveness. The shutdown margin of reactivity will be at least 1% under these circumstances.

9. Backup Reactivity Shutdown Capability

AEC Criterion 9

Backup reactivity shutdown capability must be provided that is independent of normal reactivity control provisions. This system must have the capability to shut down the reactor from any operating condition.

AARR Criteria

- 9.1 A backup reactor shutdown system will be provided. This system will be independent of the system of control plates which normally control reactivity, and of different principle. The rate of introduction of negative reactivity by the backup system will be more than adequate to control the rate of gain of reactivity during cool-down of the reactor coolant after shutdown from equilibrium operation.
- 9.2 The total reactivity-control worth of the backup reactor shutdown system will be adequate to shut down the reactor from any normal operating condition, and to maintain shutdown even with all control plates at their positions of minimum effectiveness.
- 9.3 The backup reactor shutdown system will be designed for periodic testing or monitoring, to assure operability at all times when the reactor is operating.

10. Emergency Heat Removal

AEC Criterion 10

Heat removal systems must be provided which are capable of accommodating core decay heat under all anticipated abnormal and credible accident conditions, such as isolation from the main condenser and complete or partial loss of primary coolant from the reactor.

AARR Criteria

- 10.1 Emergency cooling, capable of removing all decay heat from the reactor after shutdown, will be provided. (Criterion 4.11).
- 10.2 Redundant d-c driven pony motors supplied with Class I uninterrupted power (see Criterion 21.2) will be provided on the primary coolant circulation pumps to assure removal of decay heat from the reactor following shutdown.
- 10.3 The reactor vessel internals will be so designed as to minimize the possibility of uncovering the core in the event of a major rupture anywhere in the primary coolant system (including beam tubes).
- 10.4 The instrumentation in operation during emergency cooling will consist of the necessary nuclear and process channels required to monitor the shutdown reactor and the emergency cooling provisions and will be energized from a Class I uninterrupted power source (see Criterion 21.2).

11. Nil-Ductility-Transition Temperature

AEC Criterion 11

Components of the primary coolant and containment systems must be designed and operated so that no substantial pressure or thermal stress will be imposed on the structural materials unless the temperatures are well above the nil-ductility temperatures. For ferritic materials of the coolant envelope and the containment, minimum temperatures are $NDT + 60^{\circ}F$ and $NDT + 30^{\circ}F$, respectively.

AARR Criteria

- 11.1 The fast neutron damage in the reactor vessel will be limited such that the quantity (nil-ductility-transition temperature plus $60^{\circ}F$) will not exceed the vessel temperature under normal operating conditions.
- 11.2 The primary system will not be pressurized when the materials in the walls are at a temperature less than the nil-ductility-transition temperature plus $60^{\circ}F$.
- 11.3 Steel plate membranes rendering the containment building gastight under the overpressure condition, whether used also as the structural member bearing the pressure load or not, will be suitably insulated and protected from outside ambient conditions so that the temperature of these members is maintained at a value higher than the nil-ductility-transition temperature plus $30^{\circ}F$ at all times when the reactor is operating.
- 11.4 Impact properties of materials subject to propagation failure used to construct components of the primary system and the containment will be determined by test.

12. Control Plate Insertion Under Abnormal Conditions

AEC Criterion 12

Capability for control rod insertion under abnormal conditions must be provided.

AARR Criteria

- 12.1 The design of the shim-safety control plate drive mechanisms will incorporate a means for supplying additional energy over and above gravity for rapid initial insertion of the rods under scram conditions.
- 12.2 Provision will be made for automatic power drive of all control plates into the reactor after a scram signal.
- 12.3 The reactor vessel and all of its internals, as well as the control plates and drive system components, will be designed as Class I for seismic loadings. (See Criterion 1.4).
- 12.4 The reactor vessel and all of its internals, as well as the control plates and drive system mechanical and electrical components, will be protected against tornado damage. (See Criterion 1.6).

13. Control Room

AEC Criterion 13

The reactor facility must be provided with a control room from which all actions can be controlled or monitored as necessary to maintain safe operational status of the plant at all times. The control room must be provided with adequate protection to permit occupancy under the conditions described in Criterion 17 below, and with the means to shut down the plant and maintain it in a safe condition if such accident were to be experienced.

AARR Criteria

- 13.1 The reactor control room will house all controls and instrumentation necessary for safe operation of the reactor and reactor associated systems.
- 13.2 Audible and visual alarms will be actuated upon the occurrence of any potentially unsafe abnormality in plant conditions. Means will be provided to identify the abnormality causing the alarm, and to identify at least the first trip signal involved in a given scram action.
- 13.3 The reactor control room will be provided with any necessary shielding to permit occupancy of the room under emergency conditions for as much time as is necessary to take such corrective actions as may be carried out in the control room for initial mitigation of consequences following a Maximum Credible Accident (see Criteria 17.5 and 19.4).
- 13.4 The reactor control room will be designed such that its function will not be impaired following a direct hit by a tornado (see Criterion 1.6).
- 13.5 The reactor control room will be provided with a separate ventilation system which will maintain the room at a positive pressure with respect to adjacent portions of the Laboratory and Office Building. Fans or blowers supplying this overpressure will be connected to the emergency power system. Access doors to the room will be provided with seals to minimize leakage.

14. Indication of Reactor Reactivity Status in Control Room

AEC Criterion 14

Means must be included in the control room to show the relative reactivity status of the reactor such as position indication of mechanical rods or concentrations of chemical poisons.

AARR Criteria

- 14.1 Individual instruments will be provided in the reactor control room for displaying the position of each control rod during normal operation.
- 14.2 The range of the neutron flux instrumentation provided will extend continuously from source level up to at least 300% of maximum flux at operating power. The neutron flux level will be displayed in the control room over the full range.

Note: Chemical poisons are not used in the AARR during normal reactor operation.

15. Reactor Protection System - Instrumentation and Control Circuitry

AEC Criterion 15

A reliable reactor protection system must be provided to automatically initiate appropriate action to prevent safety limits from being exceeded. Capability must be provided for testing functional operability of the system and for determining that no component or circuit failure has occurred. For instruments and control systems in vital areas where the potential consequences of failure require redundancy, the redundant channels must be independent and must be capable of being tested to determine that they remain independent. Sufficient redundancy must be provided that failure or removal from service of a single component or channel will not inhibit necessary safety action when required. These criteria should, where applicable, be satisfied by the instrumentation associated with containment closure and isolation systems, afterheat removal and core cooling systems, systems to prevent cold-slug accidents, and other vital systems, as well as the reactor nuclear and process safety system.

AARR Criteria

- 15.1 All operating parameters important to safety will be monitored. Reactor scram will be initiated automatically by the safety system when any appropriate parameter exceeds a preset safe limit. (Criterion 4.5).
- 15.2 The reactor safety system will utilize signals derived from certain abnormal conditions in the nuclear and process systems, reactor experimental facilities, and plant environment. Provision for manual actuation of the safety system will also be included.
- 15.3 Redundancy will be employed, where appropriate, to prevent the nullification of the safety system by the failure of any one component or instrument channel. The safety system here includes all instrument channels and scram logic equipment required for a reactor scram.
- 15.4 Each instrument channel of redundant instrumentation monitoring a scram variable will be designed, to the maximum extent feasible, to be physically and electrically independent of the other channels measuring the same variable. An instrument channel is here defined to include the primary element and transmitter, intermediate amplifiers or signal converters, readout equipment and the final control device.

- 15.5 An on-line testing system will be provided for testing each redundant instrument channel monitoring a scram variable. Whenever a channel is made inoperative during a test, it will automatically generate a trip output signal. Thus, where redundant channels are used in a two-out-of-three configuration, tripping of either of the remaining two channels will initiate reactor scram. Testing will be accomplished on a regularly scheduled basis and the results will be recorded. Where technically and economically practicable, the test will involve the complete channel from parameter change to final scram contact.
- 15.6 In groups of redundant instrumentation channels, both of two separate continuous power sources will be utilized. Not all channels will be powered by the same source (see Criterion 21.2).
- 15.7 Criteria 15.1 through 15.6 will apply to essential instrumentation and circuitry connected with containment closure, emergency cooling and other systems vital to safety, as well as to the reactor nuclear and process safety system.
- 15.8 Bypassing of the scram circuits associated with the startup nuclear instruments will be accomplished automatically by contacts associated with the intermediate range instruments. Upon subsequent reduction of reactor power to the startup range, the bypass will be automatically removed. Failure of the operator to re-energize the startup channels prior to this time will result in a reactor scram.
- 15.9 Where deliberate bypassing of interlocks is permitted under certain operating conditions, it will be accomplished by means of key switches. Keys for such switches will be under strict supervisory control.
- 15.10 As a minimum, the nuclear instrumentation will include:
- Two startup range channels
 - Three intermediate/power range logarithmic channels
 - Three power range linear channels
- 15.11 A minimum of three counts per second on the startup channels will be required before reactor startup.
- 15.12 A minimum of one decade overlap will be provided between the effective ranges of the startup and intermediate/power range channels.

16. Reliability of Safety Instrumentation Systems

AEC Criterion 16

The vital instrumentation systems of Criterion 15 must be designed so that no credible combination of circumstances can interfere with the performance of a safety function when it is needed. In particular, the effect of influences common to redundant channels which are intended to be independent must not negate the operability of a safety system. The effects of gross disconnection of the system, loss of energy (electric power, instrument air), and adverse environment (heat from loss of instrument cooling, extreme cold, fire, steam, water, etc.) must cause the system to go into its safest state (fail-safe) or be demonstrably tolerable on some other basis.

AARR Criteria

- 16.1 The safety system will be designed to fail safe wherever possible. With regard to instrumentation and control, the term "fail safe" is here defined as failing to a condition which will neither cause the insertion of positive reactivity nor nullify the safety system capability to scram the reactor or to take other intended action when required. For example, nuclear instrumentation channel trips will result from (1) off normal variable conditions, (2) loss of neutron detector high voltage, (3) disconnection of interconnecting cables, and (4) where feasible, open circuits, short circuits, and grounds in the electronic equipment.
- 16.2 Wherever possible, all nuclear and process channels that have reactor shutdown functions for an excessively high or low value of the measured variable will be arranged to actuate a channel failure signal for apparent extreme values at the opposite end of the range. This is based on the premise that most instrument failures result in either maximum output or zero output.
- 16.3 Component chassis that are associated with alarm or scram functions will be provided with interlocks so that removal of one of these chassis will automatically generate the alarm or scram associated with that channel.
- 16.4 All safety system instrumentation will be specified for operation in ambient temperatures between 5°C and 50°C and a maximum relative humidity of 90%. The supplier will be required to furnish certified results of tests that verify the proper operation of representative samples of all units at these extremes of environmental conditions.
- 16.5 The safety system instrumentation located in the control room will be further tested in situ to demonstrate proper operation at extreme operating temperatures. The control room temperature for this test will be less than 122°F (50°C) specified for the instruments, and is expected to be about 110°F.

- 16.6 Instrument transmitters located elsewhere in the plant will similarly be specified. Although they will, in general, not be located in air-conditioned spaces, representative temperatures will be observed closely during initial operation of the plant. At this time corrective action will be taken as needed to adjust environmental temperatures to acceptable values for all phases of normal plant operation. Failure of any significant portion of the plant cooling systems such that environmental temperatures can be expected to rise beyond acceptable values will be considered cause for appropriate adjustment of operating conditions of the reactor plant.

C. Engineered Safeguards

17. Containment Structure Pressure and Temperature

AEC Criterion 17

The containment structure, including access openings and penetrations, must be designed and fabricated to accommodate or dissipate without failure the pressures and temperatures associated with the largest credible energy release including the effects of credible metal-water or other chemical reactions uninhibited by active quenching systems. If part of the primary coolant system is outside the primary reactor containment, appropriate safeguards must be provided for that part if necessary, to protect the health and safety of the public, in case of an accidental rupture in that part of the system. The appropriateness of safeguards such as isolation valves, additional containments, etc., will depend on environmental and population conditions surrounding the site.

AARR Criteria

- 17.1 The reactor and primary coolant system will be entirely enclosed within the Reactor Containment Building.
- 17.2 The Reactor Containment Building will be designed to withstand the peak overpressure which could occur in the building as a result of the Maximum Credible Accident, as defined in Criterion 17.5, with a factor of safety of not less than 1.1. The peak ambient temperature and temperature history which could occur in the building during the MCA will be taken into account in the design to assure that buckling failure of the liner will not occur. The design will be such as to accommodate earthquake loadings (see Criterion 1.4), tornado loadings (see Criterion 1.5) and missile loadings, in addition to the normal live and dead loads and the MCA loading, in accord with the following table:

<u>Case</u>	<u>Live Load</u>	<u>Dead Load</u>	<u>MCA Load</u>	<u>Earthquake Load</u>	<u>Tornado Load</u>	<u>Missile Load</u>
1	X	X	X	X		
2	X	X	X		X	
3	X	X	X			X

(The containment design pressure and peak ambient temperature are 6.0 psig and 235°F, respectively, based upon the MCA calculations, Section VII-B-2.)

- 17.3 The design pressure for the air locks and other penetrations of the Reactor Containment Building will be the same as that for the building proper. The peak ambient temperature, and temperature history under MCA conditions, will be taken into account in the design of these devices in the same manner as in the design of the building proper, except that the peak ambient temperature will be used as the design temperature for gaskets and other temperature-sensitive materials. The air locks will be designed to hold the design pressure with either the inner or the outer door open.
- 17.4 Any underwater containment penetrations will be designed to withstand the design pressure with or without the water on either side of the penetration.
- 17.5 In these Criteria, the Maximum Credible Accident is defined as the largest credible energy and/or fission product release from the reactor, as described in Section VII of this report.

18. Containment Heat Removal Provisions

AEC Criterion 18

Provisions must be made for the removal of heat from within the containment structure as necessary to maintain the integrity of the structure under the conditions described in Criterion 17 above. If engineered safeguards are needed to prevent containment vessel failure due to heat released under such conditions, at least two independent systems must be provided, preferably of different principles. Backup equipment (e.g., water and power systems) to such engineered safeguards must also be redundant.

AARR Criteria

None.

(No engineered safeguards are necessary to remove heat from within the containment structure to maintain integrity of the structure under the conditions described in Criterion 17.2 et seq.)

19. Containment Leakage RateAEC Criterion 19

The maximum integrated leakage from the containment structure under the conditions described in Criterion 17 above must meet the site exposure criteria set forth in 10 CFR 100. The containment structure must be designed so that the containment can be leak tested at least to design pressure conditions after completion and installation of all penetrations, and the leakage rate measured over a suitable period to verify its conformance with required performance. The plant must be designed for later tests at suitable pressures.

AARR Criteria

19.1 The following definitions describe the terms used in Criteria 19.2 et seq.:

- a. Limiting Radiation Dose - A total radiation dose of 25 rem to the whole body or 300 rem to the thyroid from iodine exposure.
- b. Exclusion Area - The area surrounding the reactor in which the Laboratory has the authority to determine all activities, including the exclusion or removal of personnel and property. (This area coincides with the Argonne National Laboratory site, all of which is U. S. Government property and over which the Laboratory has administrative control. The minimum radius of the area, with AARR at its center, is 1500 meters.)
- c. Low Population Zone - The area immediately surrounding the Exclusion Area, which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident. (For Argonne National Laboratory reactor sites, this is an annular area having an inner radius of 1500 meters and an outer radius of 5000 meters, with AARR at its center, containing about 12,500 residents - see Section III-B. The area, except for one town of about 3400 residents - Lemont, Illinois - is rural in nature, but possesses a system of roads and highways which would be more than adequate for communication, warning, and evacuation if necessary.)
- d. Population Center Distance - The distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents. (The nearest densely-populated center to Argonne National Laboratory is Downers Grove, Illinois, with a 1960 Census population just over 21,000. The nearest boundary of Downers Grove is about 7000 meters from the AARR site, or about 1.4 times the outer radius of the Low Population Zone as defined in paragraph c. above.)

- 19.2 The guaranteed leakage rate^{*} for the Reactor Containment Building, including its penetrations, the means for isolating the containment building upon detection of high radiation levels in building ventilation or other exhaust effluents, and the design of shielding protecting personnel outside the building from radioactive materials released inside the building during an accident, will be specified such that the following conditions would be satisfied following the occurrence of the Maximum Credible Accident (see Criterion 17.5):
- a. At the boundary of the Exclusion Area and beyond, an individual would receive a total radiation dose much less than the Limiting Radiation Dose during any two-hour period following the occurrence of the accident. Persons within the Exclusion Area (except possibly those within the Reactor Containment Building) would be afforded a reasonable probability that they might secure and evacuate to a safe distance, in an orderly manner according to procedures to be established, in such period of time that the total radiation dose received would not exceed the Limiting Radiation Dose.
 - b. At the outer boundary of the Low Population Zone and beyond, an individual exposed to the radioactive plume which might result from the occurrence of the accident would receive a total radiation dose much less than the Limiting Radiation Dose during the entire passage of the plume.
- 19.3 Environmental and distance characteristics of the site will be evaluated on the basis of the procedures described in TID-14844^{**} except where meteorological peculiarities of the area can be shown to justify special evaluation.
- 19.4 Special provisions will be made to permit limited further occupancy of the reactor control room after an accident, as indicated in Criteria 13.3 through 13.5.
- 19.5 Access air locks to the Reactor Containment Building will be designed so that in depressurizing the space between the air lock doors preparatory to opening the outer door for egress of personnel, fission products or other radioactivity contained in the pressurized air will not be released to the area immediately surrounding the air lock exit.

^{*} The containment design specifies a design maximum leakage rate of 0.094%, and a guaranteed maximum leakage rate of 0.94%, of the contained volume per day at the design pressure of 6.0 psig, to be corrected for building temperature at time of testing.

^{**} J. J. DiNunno, et al, "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, 1962.

- 19.6 Provisions will be made in the design for testing the Reactor Containment Building after completion and installation of all penetrations, and at suitable intervals thereafter, to verify the conformance of the leakage rate with the required value. Leakage rate will be determined in a manner which meets the intent or spirit of Proposed ANS Standard 7.60 (December 1964). Initial acceptance will be based on meeting the guaranteed leakage rate (see Criterion 19.2) at the design pressure. Later confirmation of continued conformance will be based on tests at reduced pressure, with results meeting a reduced leakage rate requirement commensurate with the test pressure as determined from the relationship of leakage rate to test pressure established during the initial test.
- 19.7 Special leak testing requirements will be applied during fabrication to those components and systems which will form a part of the Reactor Containment Building boundaries, so that these components do not contribute materially to the leakage rate for the building.

20. Containment Penetrations

AEC Criterion 20

All containment structure penetrations subject to failure such as resilient seals and expansion bellows must be designed and constructed so that leak-tightness can be demonstrated at design pressure at any time throughout operating life of the reactor.

AARR Criterion

- 20.1 Provision will be made in the design for leak detection tests to be performed at maximum accident pressure (see Criterion 17.5) on penetrations through the Reactor Containment Building, and on isolation valves in pipe lines penetrating the containment, both during construction and at any later time during a scheduled reactor shutdown.

21. Electrical Power Sources

AEC Criterion 21

Sufficient normal and emergency sources of electrical power must be provided to assure a capability for prompt shutdown and continued maintenance of the reactor facility in a safe condition under all credible circumstances.

AARR Criteria

- 21.1 The commercial electrical power (Class III) distribution will be arranged to provide power through two separate feeder systems from the primary Laboratory substations to the AARR facility. The two systems will be isolated from each other to prevent common modes of failure. Each feeder system will be sized for capacity to carry the entire AARR load.
- 21.2 Two independent battery sources will provide uninterrupted power (Class I) to vital components, including vital instrumentation and the emergency cooling equipment, upon failure of Class III power. (See Criteria 10.2, 10.4, 15.6 and 22.8.) Capacity will be such as to provide full protection against core damage after reactor shutdown until such time as the core can self-cool. The arrangement of Class I power supply to the emergency cooling equipment will be such that failure of one component, energy source or bus, after the loss of commercial power, will not nullify the capability of the emergency electrical system to supply the power necessary for this system.
- 21.3 Two separate emergency generators (steam, gas, or diesel driven) will be provided to supply power (Class II) to emergency loads which can accept a brief interruption upon failure of Class III power, until the emergency generators can be started up. Each such generator will be sized to carry the entire emergency load (including the Class I load), and will have an independent fuel supply sufficient in capacity to carry this load for a minimum of 24 hours.

22. Containment Isolation

AEC Criterion 22

Valves and their associated apparatus that are essential to the containment function must be redundant and so arranged that no credible combination of circumstances can interfere with their necessary functioning. Such redundant valves and associated apparatus must be independent of each other. Capability must be provided for testing functional operability of these valves and associated equipment to determine that no failure has occurred and that leakage is within acceptable limits. Redundant valves and auxiliaries must be independent. Containment closure valves must be actuated by instrumentation, control circuits and energy sources which satisfy Criterion 15 and 16 above.

AARR Criteria

- 22.1 Pipes and ventilation ducts which penetrate the Reactor Containment Building will be classified into five categories:
- a. Lines which open into either the primary coolant system or the containment volume, and which open or can be opened to the atmosphere outside the Reactor Containment Building.
 - b. Lines which open into either the primary coolant system or the containment volume, and which connect to systems completely closed to the atmosphere outside the containment building.
 - c. Lines which connect to systems completely closed to the primary coolant system and the containment volume, and which open or can be opened to the atmosphere outside the containment building.
 - d. Lines which connect to systems completely closed to the primary coolant system and the containment volume, and which connect to systems completely closed to the atmosphere outside the containment building.
 - e. Fluid inlet lines which are part of systems whose operation is needed after an accident releasing energy and/or radioactive products to the containment building.

- 22.2 Each line in Category 22.1a above will be provided with two isolation valves in series, located as close as practicable to the penetrated containment wall. The valves on each line will be physically separated, and wherever practicable will be located such that one valve is inside the containment building and the other is outside. Operation of the valves will be completely independent. These valves, together with associated controls and valve operators, will constitute the isolation system which must operate to effect containment.
- 22.3 The isolation system will be capable of manual trip from the reactor control room, and will also trip automatically upon receipt of an appropriate isolation signal (such as high activity in the air exhausted to the stack). Criteria regarding redundancy of instrument channels, testing and fail-safe design will be applied to isolation system trip instrumentation in the same manner as for reactor shutdown instrumentation (see Criteria 15.7 and 16.1).
- 22.4 Except in cases where provision of such valves would constitute a safety hazard, each line in Category 22.1b or 22.1c above will be provided with at least one isolation valve, which may be located outside the containment building, but which should be located as close as practicable to the penetrated wall. Valves in this category will be capable of manual trip from the control room. (All lines in these categories will be specifically evaluated during Title II design to determine whether or not such isolation valves should be provided.)
- 22.5 Lines in Category 22.1d above will not require isolation valves, but in cases where no valves are provided, the entire system must meet the requirements of Criterion 22.7 below.
- 22.6 For lines in Category 22.1e above, automatic isolation valves as described in Criterion 22.2 will not be used. Since the normal flow of water in these systems will be inward to the containment building, check valves located in these lines will provide automatic isolation when necessary. The check valves will be closed by reverse (outward) fluid flow.
- 22.7 Isolation valves described in Criteria 22.2, 22.4 and 22.6, and portions of systems which could constitute a part of the containment system because of location of the isolation valves, will be designed to meet strength and leakage requirements appropriate to their containment function.
- 22.8 Electric power connected with operation of containment isolation valves will be supplied from the constant (Class I) power supply. On lines requiring two isolation valves (Category 22.2a), power for each valve will be supplied from a different constant power source. (See Criterion 21.2.) Failure of isolation valve power supply will be detected and annunciated.
- 22.9 Isolation valves will be designed to fail preferentially either closed or in position, whichever provides maximum safety, in the event of motive power supply failure.

23. Testing of Engineered Safeguards

AEC Criterion 23

In determining the suitability of a facility for a proposed site, the acceptance of the inherent and engineered safety afforded by the systems, materials and components, and the associated engineered safeguards built into the facility, will depend on their demonstrated performance capability and reliability and the extent to which the operability of such systems, materials, components, and engineered safeguards can be tested and inspected during the life of the plant.

AARR Criteria

- 23.1 The emergency cooling pony motors will be used to cool the reactor core after shutdown and after the primary coolant pump motors have been shut off, prior to removal of the core at end of cycle. These motors will be energized, and means will be included to assure that they are operating properly, during normal reactor operation.
- 23.2 The emergency generators will be designed to permit testing at any time during reactor operation.

Note: The testing of the Reactor Containment Building and penetrations thereof are discussed in Criteria 1.7, 19.6, 19.7 and 20.1.

D. Radioactivity Control

24. Containment or Confinement of Fuel and Waste Handling Systems

AEC Criterion 24

All fuel storage and waste handling systems must be contained if necessary to prevent the accidental release of radioactivity in amounts which could affect the health and safety of the public.

AARR Criteria

- 24.1 All handling of spent cores prior to storage for after-heat decay, and all such storage of spent cores will be carried out within the Reactor Containment Building. Fuel elements will be removed from this building by underwater transfer, one element at a time, only after a sufficient decay period that the element, if exposed to air, could self-cool by natural air convection without melting.
- 24.2 The reactor off-gas system, and liquid and solid waste handling systems accepting effluent from the reactor primary coolant system, will be located either within the Reactor Containment Building or in confined spaces vented to the facility exhaust stack.

25. Subcriticality, Shielding and Cooling in Fuel Handling and Storage Facilities

AEC Criterion 25

The fuel handling and storage facilities must be designed to prevent criticality and to maintain adequate shielding and cooling for spent fuel under all anticipated normal and abnormal conditions, and credible accident conditions. Variables upon which health and safety of the public depend must be monitored.

AARR Criteria

- 25.1 The reactor vessel will be submerged in a pool of water which will serve as: (1) biological shielding during operation and fuel handling, (2) coolant for spent cores during fuel handling, (3) a pressure suppression device in the event of a primary system rupture within the pool, and (4) an absorber for volatile fission products in the event of a primary system rupture within the pool. Special consideration will be given to the design of reactor vessel, piping and beam tube penetrations through the pool walls and bottom, to minimize the possibility of seal failure and to minimize the rate of water loss should such failure occur.
- 25.2 Facilities for storage of new fuel elements will be located in an area incapable of being flooded, will be adequately drained, and will be provided with suitable devices employing either spatial separation, fixed poisons or both, to assure negligible reactivity interaction.
- 25.3 Fuel element handling and transfer devices will be designed in such a way that an element in transit cannot be moved into close enough proximity with another element to produce criticality.
- 25.4 Suitable structures, employing spatial separation, fixed poisons or both, will be provided in the spent fuel storage pool and defective core storage area to assure subcriticality of stored components and negligible reactivity interaction between storage cells, even in the event of an accidental error in movement of a newly unloaded fuel element into the region. These features will be designed such that the multiplication factor for fuel in any storage area will be kept at 0.9 or less.
- 25.5 All devices involved in the fuel handling process, such as grappling fixtures, transfer mechanisms, carriers, storage fixtures, etc., will be designed to prevent, in a positive manner, (a) any interference with the natural circulation cooling of the fuel element, and (b) the upsetting of an element from its normally vertical position to a position in which the natural circulation cooling of the core would be materially retarded. (AARR fuel handling tools will be identical in design to those in use in the High Flux Isotope Reactor.)

- 25.6 The fuel transfer and storage canal complex and the fuel handling system will be designed such that all movements of highly radioactive fuel elements are carried out under water, with adequate water shielding above the elements to protect personnel above, and in full view of operators at all times.
- 25.7 The reactor pool and the fuel handling canal complex will be provided with a corrosion resistant liner to minimize leakage.
- 25.8 The canal complex will be provided with suitable water-tight separating gates or doors such that the various sections of the complex may be cleared of active materials, isolated, and emptied for cleaning or repairing of leaks.
- 25.9 The cooling system for the reactor pool and fuel handling canal complex will be designed so that decay heat from a fuel element in transit can be removed adequately and for an indefinite time from any isolated region of the complex in which the core could be rendered temporarily immovable by malfunction of the transfer mechanism.
- 25.10 Redundancy and flexibility of valving arrangement will be provided in the pool and canal cooling system to assure adequate cooling of fuel elements stored or in transit while malfunctioning components are under repair or replacement.
- 25.11 The design of the pool and canal cooling system and penetrations into the canals and pools will be such as to minimize the possibility of leaks developing which could drain the canals at significant rates or interfere with the natural circulation cooling of the cores. Adequate means will be provided to detect leaks, isolate leaking components and effect repairs without significantly affecting the safety of the system.

26. Hold-up of Radioactive Effluents

AEC Criterion 26

Where unfavorable environmental conditions can be expected to require limitations upon the release of operational radioactive effluents to the environment, appropriate hold-up capacity must be provided for retention of gaseous, liquid, or solid effluents.

AARR Criteria

- 26.1 Provision will be included in the reactor primary system off-gas system for the automatic retention of high activity effluent gases if activity levels are too high for safe discharge to the exhaust stack.
- 26.2 Suitable hold-up facilities, such as retention ponds and tanks, will be provided to collect suspect operational liquid wastes from the facility for radioactivity monitoring prior to discharge to the normal Laboratory Waste Treatment Plant (non-active) or to natural drainage. Material found to be too radioactive for such discharge will be retained and processed in the Laboratory Active Waste Treatment facility.
- 26.3 Tanks will be provided for the collection of suspect solid wastes (spent demineralizer resins). Such wastes will be removed and disposed of by administrative order.

27. Radioactivity Release Monitoring

AEC Criterion 27

The plant must be provided with systems capable of monitoring the release of radioactivity under accident conditions.

AARR Criteria

- 27.1 All accidental releases of gaseous radioactive materials from the reactor will occur either (1) within the primary system, or (2) within the Reactor Building. A release within the primary system will be discharged via the degasifier to the off-gas system, where it will be monitored and may be retained depending on its activity level. From this system, it is discharged into the exhaust plenum. Releases within the Reactor Containment Building will also be discharged to the exhaust plenum. Stack gas activity will be continuously monitored, and an excessive level will result in automatic action.
- 27.2 Breaching of the main heat exchangers could permit release of radioactive gas via the cooling tower. This event will be detected by monitoring the secondary lines leaving the heat exchangers.
- 27.3 Accidental releases of liquid materials, if they occur, will be in the Reactor Containment Building or the Active Materials Handling Building. All drains in these buildings are routed to the radioactive waste disposal system where the activity is monitored and discharge is controlled. (See Criterion 26.2). Suspect wastes generated in the Laboratory and Office Building are similarly handled, but in a separate system.

III. SITE CHARACTERISTICS

A. Description of Site and Adjacent Areas

The Argonne National Laboratory is located near Lemont, in DuPage County, Illinois, about 40 km southwest of downtown Chicago (see Fig. III-A-1). With the exception of Joliet, 22 km to the southwest, the areas south and west are primarily rural. The more heavily populated Chicago suburbs border closely on the north and east. The Des Plaines River and the Chicago Sanitary and Ship Canal are located beyond the southern boundary of the Laboratory site, thus extending the uninhabited area about 1 km beyond this border. The North, West and East boundaries are approximately 2 km from AARR.

The surrounding land is varied in both ownership and use. Common boundaries with Argonne are held by the Atchison, Topeka and Santa Fe Railroad, the Chicago Sanitary District, the Federal and County Highway Commissions, the County Forest Preserve, and several private residence- and land-owners.

The AARR site, which is northwest of the existing Building 330 which houses the CP-5 reactor, is located in the central portion of the 3700-acre Argonne Site.

The other major reactors at Argonne, namely CP-5, JUGGERNAUT and the ZPR's are located within a 600-m radius of the AARR complex. Access to the AARR site will be provided by a new asphaltic concrete road connected to an existing road along the eastern boundary of the AARR site (see Fig. III-A-2 and IV-44).

B. Population Distribution

1. ANL Site

It is estimated that by 1980, at any given time there will be a maximum of approximately 6400 persons on the Argonne site. Of these, approximately 300 are AEC employees localized in the east area, and 100 are guests that are assumed to be on the site during normal working hours. An estimate of the maximum present and estimated future on-site population within given distances of the AARR reactor is as follows:

<u>Radius (meters)</u>	<u>Total Population</u> <u>Within Indicated Radius</u>	
	<u>June 30, 1965</u>	<u>1980</u>
At AARR	0	146
200	80	226
300	99	246
400	325	475
500	377	530

Fig. III A-1. Regional Map

FIG. III A-2
ARGONNE NATIONAL LABORATORY SITE PLAN

<u>Radius (meters)</u>	<u>June 30, 1965</u>	<u>1980</u>
600	579	732
700	726	880
800	1043	1200
900	1560	1720
1000	2600	2770
1100	3775	4150
1200	3886	4270
1300	4183	4570
1400	4183	4570
1500	4183	4570
2000	5085	5462
Over 2000	5923	6400

Table III-B-1 is a tabulation of the population on a building-by-building basis as of June 30, 1965.

2. Off-Site

a. General

Within a 10-km radius of the reactor site are the communities of Lemont (pop. 3,397), 4 km southwest; Downers Grove (pop. 21,154), Westmont (pop. 5,997), and Clarendon Hills (pop. 5,885), 10 km to the north; Romeoville (pop. 3,574), 9 km to the southwest; and Willow Springs (pop. 2,348), 8 km to the east. The 1960 populations of communities near Argonne are shown in Fig. III-B-1. Of the 874,366 people shown as living within 24 km of Argonne, 11,912 were within 4.8 km, 27,381 were within 8 km, and 213,919 were within 16 km.

Although the nearest ANL site boundary is about 1.5 km from the reactor, the area between this boundary and the 2-km-radius circle is relatively uninhabited. To the south the Des Plaines River, the Chicago Sanitary and Ship Canal, and the Atchison, Topeka and Santa Fe railroad pass through this area. The only inhabitants, who reside in the southwest and northwest quadrants, total 635 in number.

b. Present Population

A map of the metropolitan area west of Chicago showing the population of various villages and cities within a 24-km radius is shown in Fig. III-B-1. A "Township, Area and Population Breakdown," Table III-B-2, is also included, for more comprehensive information. In the table, townships which fall entirely or partly within the 24-km radius are listed alphabetically. For each township the following figures are shown: total, city, and rural population; total, city, and rural area; and persons per sq km in the rural area. The total population was taken from the 1960 census report. The city population is a total of all city populations, whole or part, listed for that township in the census report. The rural population is the difference between city population and total population. The area figures were obtained by measuring with the use of the map scale and a planimeter. The number of persons per sq km was computed by dividing rural population by rural area.

TABLE III-B-1

ARGONNE NATIONAL LABORATORY ON-SITE POPULATION

(ARGONNE, ILLINOIS SITE)

JUNE 30, 1966

BLDG. OR AREA NO.	NAME OF FACILITY	POPULATION*
2	Administration Building	242
3	Cafeteria - East Area	12
4-5-6	Warehouse, Garage	176
8	Firehouse	50
10	Executive Offices	93
11-12-14-19-21	Reactor Engineering, Remote Control Engineering, Industrial Hyg. & Safety, Electronics, Security	278
15-22-24-25	Office of College and University Cooperation	129
16-17-20-40-54	Central Shops	282
37	Plant Services, Industrial Hygiene & Safety	42
60	Radiobiology Experimental Station	9
90	Communications & Special Services, Branch Post Office	28
100 Area	Utilities	46
181	Meteorology	8
200-211	Chemistry, Solid State Science	443
202	Biological & Medical Research	286
203	Physics, Radiological Physics, Library	351
205	Chemical Engineering	278
206-208	Reactor Engineering and Reactor Physics	271
212	Fuels Technology Center, Solid State Science	367
213	Cafeteria - West Area	31
221	Mathematics & Computer Facility	161
301	Hot Laboratory	19
306	Reclamation	42
308	Reactor Engineering	58
310	Chemical Engineering	51
315-316-330-331-335	Research Reactors	186
340	Experimental Animal Quarters	9
350	Plutonium Fuel Fabrication	32
360-380	ZGS Facility	445
362	High Energy Physics	551
600 Area	Guest Facility	7
800 Area	Offices, Laboratories & Shops	458
	ANL Population	5,441
2	Atomic Energy Commission	265
15	Associated Midwest Universities	4
90	Argonne Credit Union	5
801	The University of Chicago Resident Auditor	2
600 Area	Guests at the Lodging Facility	122
	U.S. Army Missile Launching Site	84
	Non-ANL Population	482
	Total Population	5,923

*Maximum at any one time

<u>COMMUNITY</u>	<u>POP.</u>	<u>COMMUNITY</u>	<u>POP.</u>
ADDISON	6741	MARKHAM	11704
ALSIP	3770	MATTESON	3225
BEDFORD PARK	737	MAYWOOD	27330
BELLWOOD	20729	McCOOK	441
BERKELEY	5792	MELROSE PARK	22291
BERWYN	54224	MIDLOTHIAN	6605
BLUE ISLAND	19618	MOKENA	1332
BRIDGEVIEW	7334	NAPERVILLE	12933
BROOKFIELD	20429	NEW LENOX	1750
CAROL STREAM	836	NORTHLAKE	12318
CHICAGO	3550404	NORTH RIVERSIDE	7989
CHICAGO RIDGE	5748	OAK BROOK	324
CICERO	69130	OAK BROOK TERRACE	1121
CLARENDON HILLS	5885	OAK FOREST	3724
COUNTRY CLUB HILLS	3421	OAKLAWN	27471
COUNTRYSIDE	2393	OAK PARK	61093
CREST HILL	5887	OLYMPIA FIELDS	1503
CRESTWOOD	1213	ORLAND PARK	2592
DOWNERS GROVE	21154	PALOS HEIGHTS	3775
ELMHURST	36991	PALOS HILLS	3766
ELMWOOD PARK	23866	PALOS PARK	2169
EVERGREEN PARK	24178	PARK FOREST	29993
FLOSSMOOR	4624	PLAINFIELD	2183
FOREST PARK	14452	POSEN	4517
FOREST VIEW	1042	RICHTON PARK	933
FRANKFORT	1135	RIVER GROVE	8464
FRANKLIN PARK	18322	RIVERSIDE	9750
GLENDALE	173	ROBBINS	7511
GLEN ELLYN	15972	ROCKDALE	1272
HARVESTER	299	ROMEORVILLE	3574
HAZELCREST	6205	SHOREWOOD	358
HICKORY HILLS	2707	STICKNEY	6239
HILLSIDE	7794	STONE PARK	3038
HINSDALE	12859	SUMMIT	10374
HODGKINS	1126	TINLEY PARK	6392
HOMETOWN	7479	VILLA PARK	20391
INDIAN HEAD PARK	385	WESTCHESTER	18092
JOLIET	66780	WEST CHICAGO	6854
JUSTICE	2803	WESTERN SPRINGS	10838
LA GRANGE	15285	WESTMONT	5997
LA GRANGE PARK	13793	WHEATON	24312
LEMONT	3397	WILLOW BROOK	157
LISLE	4219	WILLOW SPRINGS	2348
LOCKPORT	7560	WINFIELD	1575
LOMBARD	22561	WOODRIDGE	542
LYONS	9936	WORTH	8196

Fig. III B-1

Population of Communities Near
Argonne National Laboratory
(Based on 1960 Census)

TABLE III-B-2

TOWNSHIP, AREA, AND POPULATION BREAKDOWN

Township	Population			Area (sq km)			Persons per sq km (Rural)
	Total	City	Rural	Total	City	Rural	
Addison	41,808	27,514	14,294	93	26	67	212
Berwyn	54,224	54,224	-	9.6	9.6	-	-
Bloom	70,530	64,007	6,523	93	31	62	105
Bloomington	14,924	5,852	9,072	93	11.4	81.6	111
Bremen	55,392	44,626	10,766	93	45.8	47.2	227
Calumet	19,299	19,299	-	10	10	-	-
Cicero	69,130	69,130	-	14.8	14.8	-	-
Downers Grove	66,664	44,298	22,366	124	32	92	244
Du Page	4,725	3,470	1,255	93	5	88	14
Frankfort	5,784	2,467	3,317	93	4	89	37
Homer	4,078	-	4,078	93	-	93	44
Joliet	94,116	67,755	26,361	93	31	62	424
Lemont	6,732	3,397	3,335	54.4	3.1	51.3	65
Lisle	20,982	12,847	8,135	93	17	76	106
Lockport	26,882	13,551	13,331	93	14	79	170
Lyons	82,214	65,862	16,352	95	69	26	619
Milton	51,361	40,483	10,878	93	28	65	168
Naperville	8,218	5,668	2,550	93	5	89	29
New Lenox	6,232	1,750	4,482	93	7	86	52
Orland	7,444	3,227	4,217	93	9	84	50
Palos	17,728	13,886	3,842	91	29	62	62
Plainfield	6,655	2,183	4,472	93	3	90	49
Proviso	160,275	157,891	2,384	77	68	9	256
Rich	35,258	32,389	2,869	93	31	62	46
Riverside	17,875	17,732	143	10	9	1	27
Stickney	31,404	7,327	24,077	47	40	7	3321
Thornton	138,444	135,613	2,831	93	72	21	137
Troy	2,679	655	2,024	93	3	90	22
Wheatland	1,023	-	1,023	93	-	93	11
Winfield	16,437	11,531	4,906	93	10	83	59
Worth	107,761	91,988	15,773	96	83.4	12.4	1269
York	89,988	76,218	13,770	93	54	39	354

The area within the 24-km radius was divided into segments by drawing in circles and radii, with Building 331 as a center point. The radii of the circles indicate the distance in miles from the center point. These distance zones are identified by the letters A, B, C, and D, starting at the center. The radius of the A zone is 4.8 km, the B zone 8 km, the C zone 16 km, and the D zone 24 km. The A and B zones are divided by the radii into 8 equal segments, respectively, and the C and D zones into 16 equal segments, respectively. The area of each segment in the A zone is 9.1 sq km, in the B zone 16.3 sq km, in the C zone 38.1 sq km, and in the D zone 63.4 sq km. The segments are numbered 1 through 16, starting at the top or north, and reading clockwise around the circumference. Each segment is identified by using the zone letter and the number of the segment. For example, the first segment in the inner circle of A zone reading clockwise from the north would be A-2, the first segment between the inner and the second circle would be B-2, the first segment between the second and third circles C-1, and the first segment between the third and fourth circles, D-1. The population of part of a city, in those cases where a city is divided into 2 or more segments, was determined by estimating the percentage of the population of the city for each segment into which it is divided, except for the city of Chicago. The population figures for that part of Chicago which falls within the 24 km radius were taken from an Official One Mile Intersection Map prepared by the Chicago Civil Defense Corps, which lists population per square mile as received from the U. S. Bureau of the Census.

The "Segment Area and Population Breakdown" as shown in Table III-B-3 was prepared as follows: The segments are listed numerically starting with the A zone, followed by the B, C, and D zones. For each segment, rural area and population, city area and population, and total area and population are listed. The rural area of each segment was measured with a planimeter and subtracted from the segment total to obtain the city area in those segments having both rural and city areas. The rural population was computed by multiplying the rural area by the population per sq km in the rural area. The city population per segment is the sum of the population figures of those cities within the segment.

c. Future Population

The projected population is based upon information received from, and consultation with the Northeastern Illinois Metropolitan Area Planning Commission, utility companies and local governmental agencies. Figure III-B-2 has been divided into four quadrants and three circles with center at the Laboratory and radii of 2, 5 and 7 km. The present, and estimated populations in 1970 and 1980 for each of the resulting twelve areas are as follows:

<u>Area Number</u>	<u>Off-Site Population</u>		
	<u>1960</u>	<u>1970</u>	<u>1980</u>
<u>0-2 km radius</u>			
A-1 (Northeast)	-	-	-
A-2 (Southeast)	-	-	-
A-3 (Southwest)	140	160	170
A-4 (Northwest)	495	620	795
Total	635	780	965

TABLE III-B-3

SEGMENT AREA AND POPULATION BREAKDOWN

Segment No.	Rural		City		Total	
	Area	Population	Area	Population	Area	Population
A-2	5.2	1,264	3.9	-	9.1	1,264
A-4	5.7	1,297	3.4	-	9.1	1,297
A-6	7.2	842	1.8	-	9.1	842
A-8	8.0	567	1.0	-	9.1	567
A-10	5.2	336	3.9	3,125	9.1	3,461
A-12	8.0	1,165	1.0	-	9.1	1,165
A-14	7.8	1,420	1.3	-	9.1	1,420
A-16	7.8	1,896	1.3	-	9.1	1,896
B-2	15.8	3,855	0.5	124	16.3	3,979
B-4	12.9	3,187	3.4	676	16.3	3,863
B-6	16.3	1,032	-	-	16.3	1,032
B-8	16.3	987	-	-	16.3	987
B-10	14.5	681	1.8	272	16.3	953
B-12	16.3	233	-	-	16.3	233
B-14	14.5	683	1.8	336	16.3	1,019
B-16	15.5	3,197	0.8	206	16.3	3,403
C-1	18.9	5,557	19.2	16,567	38.1	22,124
C-2	12.9	6,073	25.1	36,644	38.1	42,717
C-3	18.1	11,221	18.1	6,157	38.1	17,378
C-4	13.2	2,980	24.9	9,707	38.1	12,687
C-5	21.2	1,312	16.8	10,623	38.1	11,935
C-6	31.8	1,824	6.2	2,381	38.1	4,205
C-7	34.2	1,655	3.9	363	38.1	2,018
C-8	38.1	1,661	-	-	38.1	1,661
C-9	37.5	2,453	0.5	1,083	38.1	3,536
C-10	32.6	5,313	5.4	7,156	38.1	12,469
C-11	31.8	2,988	6.2	2,895	38.1	5,883
C-12	38.1	510	-	-	38.1	510
C-13	34.7	1,067	3.4	3,621	38.1	4,688
C-14	31.8	3,322	6.2	5,820	38.1	9,142
C-15	32.3	3,936	5.7	5,700	38.1	9,636
C-16	23.3	6,822	14.8	19,127	38.1	25,949
D-1	17.9	4,660	45.6	56,174	63.4	60,834
D-2	7.2	1,853	56.2	125,484	63.4	127,337
D-3	1.3	2,201	62.1	92,465	63.4	94,666
D-4	16.1	42,687	47.3	67,678	63.4	110,365
D-5	16.1	17,050	47.3	42,258	63.4	59,308
D-6	46.6	7,149	16.8	10,313	63.4	17,462
D-7	60.6	2,628	2.8	650	63.4	3,278
D-8	61.6	3,556	1.8	2,816	63.4	6,372
D-9	52.6	10,688	10.9	21,566	63.4	32,254
D-10	33.1	8,704	30.3	47,958	63.4	56,662
D-11	60.8	3,661	2.6	2,183	63.4	5,844
D-12	63.4	908	-	-	63.4	908
D-13	63.4	1,423	-	-	63.4	1,423
D-14	59.5	2,639	3.9	3,492	63.4	6,131
D-15	44.8	6,551	18.6	26,250	63.4	32,801
D-16	31.6	7,302	31.8	37,500	63.4	44,802
TOTAL					1827.8	874,366

FIG. III - B-2
 ARGONNE NATIONAL LABORATORY AND THE IMMEDIATE SURROUNDING AREA

<u>Area Number</u>	<u>Off-Site Population</u>		
	<u>1960</u>	<u>1970</u>	<u>1980</u>
<u>2-5 km radius</u>			
B-1 (Northeast)	2,561	3,329	4,328
B-2 (Southeast)	1,409	1,691	2,029
B-3 (Southwest)	4,626	5,551	6,661
B-4 (Northwest)	<u>3,316</u>	<u>4,642</u>	<u>6,499</u>
Total	11,912	15,213	19,517
<u>5-7 km radius</u>			
C-1 (Northeast)	7,520	13,723	24,315
C-2 (Southeast)	2,019	3,029	4,543
C-3 (Southwest)	1,186	1,660	2,324
C-4 (Northwest)	<u>4,150</u>	<u>7,075</u>	<u>11,320</u>
Total	14,875	25,487	42,502

C. Meteorology

The following is a summary of meteorological data collected at Argonne:

a. At 19 feet above ground, the daily average wind speed varies from 5 mph in August to 10 mph in March and gusts exceed 50 mph about once a year. At 150 feet above ground, this speed is exceeded about 15 times a year. No air measurements above 150 feet, obtained with the aid of balloons, are available at Argonne.

b. Daily temperatures average from the low 20's in the winter to the high 70's in the summer. There are about 6850 degree-days per year. Rarely are there maximum temperatures above 100°F, and the minimum temperature recorded is -19°F.

c. Precipitation averages about 33 inches per year. Mid-day humidity values average about 50% in October to 75% in January; however, night readings are about 80% to 90% all year.

d. Solar radiation, in Langleys on a horizontal surface, varies from about 150 in December to 600 per day in June.

e. Atmospheric pressure varies between a high of 30.00 in. Hg and a low of 28.30 in. Hg with a mean of about 29.20 in. Hg.

f. Soil temperatures below grade vary only from 50°F in July to 52°F in February. Temperature at depths of four inches vary from the 80's in the summer to just below freezing in mid-winter.

Annual climatological summaries based on measurements taken at the DuPage site of the Argonne National Laboratory have been prepared and published as reports designated as ANL-4538, ANL-4793, ANL-4928, ANL-5256,

and ANL-5592. These reports cover the period from July 1949 to June 1954. As a supplement to the Argonne data, weather summaries prepared by the United States Weather Bureau are included as appendices to these annual climatological reports. Further information on normals, means, and extremes for the Chicago Area, as furnished by the United States Weather Bureau, is shown in Table III-C-1.

D. Geology and Hydrology

1. General Topography

The topography of the vicinity in general is flat to slightly hilly with an elevation of between 700 feet and 750 feet above sea level for the major portion of the Argonne site. The land for several miles north and east has an elevation in this same range but drops slowly toward the southwest. An intermittent stream, Sawmill Creek, crosses Argonne and empties into the Des Plaines River which borders the southern edge of the site. The normal level of the Des Plaines River adjacent to Argonne is 583 feet above sea level, and its flood plain extends approximately 1/8 mile onto Argonne property. The land rises sharply from the edge of this flood plain, where outcroppings of the Niagara dolomite may be seen, to the top of the bluffs overlooking the Des Plaines valley. These bluffs are approximately 650 feet above sea level. From here the land slopes gradually upward for nearly a mile to the immediate area of the AARR project, which is at elevation 745 feet above sea level.

2. Bedrock

The average elevation of the bedrock directly beneath the project site is approximately 610 feet above sea level and almost 100 feet below the bottom of the Reactor Containment Building foundation mat. From cores taken at the site it has been determined that the top 10 feet of this bedrock is composed of hard, fine grained, gray dolomite with a slightly weathered to sound condition. The Argonne deep well No. 1, drilled to a depth of 1,595 ft, reveals stone formations and their thicknesses as depicted in Fig. III-D-1. Pre-glacial erosion has worn off bedrock of more recent origin and has cut deeply into the dolomitic top surface of the presently existing bedrock in the Des Plaines River valley. Glacial till of the Wisconsin ice sheet deposited by the Valparaiso moraine overlies this bedrock and partially fills the Des Plaines River valley.

3. Subsurface Soils

The soil residing above the bedrock is composed of dense and very tough silty clays and clayey silts. There are some traces of sand. These clays were laid down, as mentioned above, as the glacial drift of the Valparaiso moraine. From the log of the soil borings taken at the site, indications are that there are no soft spots or layers of soft materials. The general character of the soil is tough and hard, and averages over 50 blows per foot (standard penetration - 140# hammer falling 30 inches on 2 inch O.D. split spoon) in the area from the bottom of the Reactor Building foundation mat to bedrock (approximately 100 ft).

TABLE III-C-1

METEOROLOGY

1a

TEMPERATURES FOR CHICAGO AREA

Month	Temperature						
	Normal			Extremes			
	Daily Maximum, °C	Daily Minimum, °C	Monthly, °C	Record Highest, °C	Year	Record Lowest, °C	Year
(a)	(b)	(b)	(b)	17		17	
J	0.39	- 8.28	- 3.94	19.4	1950	-26.1	1951
F	1.67	- 6.78	- 2.56	20.6	1954	-26.1	1951
M	7.22	- 1.67	2.78	27.8	1945	-21.7	1943
A	14.2	3.67	8.94	28.9	1958	- 7.22	1954
M	20.9	9.28	15.1	34.4	1949	- 1.11	1947
J	26.7	14.9	20.8	40.0	1953	1.67	1945
J	29.6	17.7	23.7	39.4	1956	9.44	1947
A	28.3	16.7	22.6	38.3	1947	7.78	1950
S	24.4	12.9	18.7	38.3	1953	2.22	1943
O	18.0	6.61	12.3	32.8	1954	- 6.66	1948
N	8.67	- 1.389	4.17	27.2	1950	-18.9	1950
D	1.83	- 6.34	- 2.22	18.3	1951	-24.4	1951
Yr	15.2	4.89	10.1	40.0	June 1953	-26.1	Feb 1951

1b

PRECIPITATION FOR CHICAGO AREA

Month	Precipitation						
	Normal Total, cm	Maximum Monthly, cm	Year	Minimum Monthly, cm	Year	Maximum in 24 hr, cm	Year
	(b)	17		17		17	
J	4.67	9.58	1950	0.965	1956	3.51	1950
F	3.57	8.51	1950	0.838	1958	3.18	1949
M	7.24	12.7	1954	0.838	1958	6.35	1948
A	7.16	21.2	1947	3.25	1946	10.4	1947
M	9.30	19.3	1945	1.98	1950	7.44	1951
J	10.5	16.3	1958	1.98	1956	11.6	1959
J	6.94	22.8	1957	3.38	1945	15.8	1957
A	8.10	15.0	1954	2.54	1953	7.90	1955
S	8.20	15.3	1945	1.17	1956	6.48	1951
O	6.50	30.6	1954	0.762	1956	14.3	1954
N	5.92	9.50	1951	2.26	1949	4.67	1959
D	4.95	16.9	1949	0.864	1943	6.04	1949
Yr	83.1	30.6	Oct 1934	0.762	Oct 1956	15.8	July 1957

1c

PRECIPITATION FOR CHICAGO AREA

Month	Snow, Sleet				
	Mean Total, cm	Maximum Monthly, cm	Year	Maximum in 24 hr, cm	Year
	17	17		17	
J	19.6	52.8	1943	16.8	1957
F	16.5	42.2	1950	21.1	1956
M	15.2	50.6	1954	30.0	1954
A	1.27	6.10	1951	6.10	1951
M	T	0.51	1954	0.51	1954
J	0	0		0	
J	0	0		0	
A	0	0		0	
S	0	0		0	
O	76	7.62	1952	7.62	1952
N	9.14	36.3	1951	20.3	1951
D	25.9	84.6	1951	25.4	1951
Yr	88.4	84.6	Dec 1951	30.0	Mar 1954

1d

WIND FOR CHICAGO AREA

Month	Wind				
	Mean Hourly Speed (m/sec)	Prevailing Direction	Fastest Speed		
	17	15	17		
J	4.87	W	22.4	W	1950
F	5.00	W	20.1	W	1951
M	5.14	W	24.1	NW	1955
A	5.00	ENE	22.4	NW	1951
M	4.47	NE	24.1	S	1950
J	3.93	SSW	22.4	W	1953
J	3.44	SSW	20.6	NW	1959
A	3.35	SW	24.1	NW	1949
S	3.84	S	21.5	SW	1959
O	4.16	S	20.1	S	1949
N	5.05	W	26.8	SW	1952
D	4.83	W	22.4	SW	1948
Yr	4.43	SSW	26.8	SW	Nov 1952

(a) Length of record, years.

(b) Normal values are based on the period 1921-1950 and are means adjusted to represent observations taken at the present standard location.

FIG. III D-1
BEDROCK AT AARR SITE

4. Hydrology

Due to the tight nature of the soil, percolation of water through it is very slow. Therefore, drainage from the AARR site will be mostly by surface runoff. Potable water for the Laboratory is obtained from two principal aquifers, the Niagara dolomite and the deeply buried Ordovician and Cambrian sandstones. The Laboratory has five shallow supply wells that obtain water from the Niagara dolomite, and one deep well, cased and sealed through the Niagara dolomite, that obtains its water from the deep sandstones. Only the Niagara dolomite need be considered in connection with possible contamination from an accidental spill of radioactive liquid, as the deep aquifers are separated from the surface by hundreds of feet of rock formations (see Fig. III-D-1), of which the Maquoketa shale is particularly impermeable. In the event of such a spill circumventing protective measures provided, the contaminated liquid would move very slowly through the existing glacial till and would be diluted by the normal ground water descending with it. As the diluted liquid approached the level of the bedrock it would be further diluted by water that is moving across and down the fissured dolomite. From this point the hydraulic gradient of the Niagara dolomite, which is generally to the south-eastward, would be followed (see Fig. III-D-2).

Tests of water movement through the glacial till at the Laboratory were made in 1952 by the U. S. Geological Survey.¹ These studies determined that the downward rate of movement of water through the till is 0.1 inch per day. At that rate it would take approximately 26-1/2 years for contaminated water entering the saturated soil at the base of the Reactor Containment Building foundation mat to reach the bedrock layer of Niagara dolomite directly beneath the site.

At the present rate of pumping it is very unlikely that contaminated water introduced into the glacial till at the Reactor Containment Building foundation mat would find its way into the cones of depression in the piezometric surface caused by the pumping from ANL supply wells (see Fig. III-D-2), and thus into the supply wells themselves. It is likely that the contaminated liquid, seeping slowly to the south or southeast, would eventually be discharged into the Des Plaines River. Dilution, radioactive decay, and partial retention through leaching and/or ion exchange by the natural soil would certainly take place in the long interval of time between insertion at the AARR site and ultimate discharge into the Des Plaines River.

The Des Plaines River, the Chicago Sanitary and Ship Canal, the Illinois and Michigan Canal, and the Sag Channel have all been interconnected by the time they reach Joliet, which lies about 22 km southwest of Argonne National Laboratory. The water flow in the Des Plaines River just prior to Joliet varies between 12 and 340 m³/sec, depending upon the flood stage and season. When the river is in flood stage, excess water is diverted to the higher capacity Sanitary Canal at a point just south of Lemont, before reaching Joliet. The flow rate of the Sanitary Canal for 1944 and 1945 averaged about 96 m³/sec; of this about 45 m³/sec is water which has been diverted from Lake Michigan, and the balance is made up of domestic and industrial discharge from this district. Both the river and the canal flow

Modified from LeRoux and Drescher, 1959

Base compiled from maps prepared by Argonne National Laboratory and from U.S. Geological Survey topographic maps

EXPLANATION

--- 620 ---

Contour of the piezometric surface
Dashed where approximate; interval 5 feet.
Datum is mean sea level

637

Well

Showing altitude of piezometric surface

1000 0 1000 3000 5000 FEET

FIG. III D-2

PIEZOMETRIC SURFACE OF WATER IN THE NIAGARA DOLOMITE AT ARGONNE NATIONAL LABORATORY, ILL.,
JUNE 2-3, 1960.

a distance of approximately 32 km/day. Therefore, any surface drainage from the AARR site would be diluted first by the river bordering the Laboratory, and then further diluted by the canal, normally within 17 hours. If such drainage occurred during a wet period, it would be diluted even more extensively and sooner.

Canal water is used prior to and in Joliet for some industrial uses, such as hydro-electric generators and condensers, as well as for irrigation at the state prison. After leaving Joliet, the Des Plaines River flows southwest to a point approximately 23 km further, where the river is joined by the Kankakee River, forming the Illinois River. The first use made of this water for drinking purposes is at Alton, Illinois, which is about 442 km from Argonne National Laboratory and 16 km downstream from the point where the Illinois River merges with the Mississippi River.

E. Seismology

Realistic predictions regarding earthquakes, or earth shocks, as well as their frequency and severity, can only be based upon the seismic history of the area concerned and the statistical representation which history provides. Significant geological features, such as known slip-planes or faults, pertinent to an area play an important role in the seismic history of that area and are, therefore, to be considered as factors in the prediction of an earthquake.

There is one known fault zone in the area. This fault, known as the "Sandwich" fault (passing through Sandwich, Illinois), strikes approximately S 60° E and is located approximately 32 km from the site as measured along a line normal to the fault and bearing from the site approximately S 30° W. It is uncertain what the dip of this fault may be. It is estimated that the major motion along this fault occurred during the Paleozoic era and sometime before the Pennsylvanian period ended and after the Silurian period. (Some estimate the fault was formed due to the uplift of the LaSalle anticline in northern Illinois after the Mississippian period.) There may be other faults paralleling and associated with the Sandwich fault.² These fault indications, which lie about 3 km north of the known fault, would broaden the zone of faulting.

A small area (of about 64 sq km) of severe faulting is located approximately 35 km north of the site and has been mapped as the "Des Plaines Disturbance."³ Because of the number, severity and local nature of these faults (tilted blocks and broken rock to depths of about 500 km) conclusions made from detailed studies of this area are that this chaotic disturbance was the result of the focusing of regional forces or the result of the impact of a large meteorite.⁴

Figure III-E-1 is a map showing the locations of earthquakes with epi-centers within a 600 km radius of the AARR site and with a probable intensity of III or greater (as evaluated in terms of the Modified Mercalli Intensity Scale of 1931), felt at the AARR site.⁵ The following table gives the pertinent data for each of these earthquakes:

FIG. III E-1
AARR PROXIMAL EARTHQUAKES

Map Location Number	Date	Epi-Center	Extent of Tremor in Sq Miles	Maximum Intensity at Center	Estimated Probable Intensity at Site
1	8/20/1804	Chicago, Illinois	30,000	-	-
2	12/16/1811	New Madrid, Missouri	2,000,000	X	IV
2	1/23/1812	New Madrid, Missouri	2,000,000	X	IV
2	2/7/1812	New Madrid, Missouri	2,000,000	X	IV
3	10/31/1895	Charleston, Missouri	1,000,000	VIII	IV
4	5/26/1909	Aurora, Illinois	500,000	VIII	VII
5	7/18/1909	Central Illinois	40,000	VII	III
6	1/2/1912	Between Aurora and Morris, Illinois	40,000	VI*	VI
7	4/9/1917	DeSoto, Missouri	200,000	VI	IV
8	3/9/1937	Lima, Ohio	150,000	VII-VIII	III
9	11/23/1939	Waterloo, Illinois	150,000	V	II-III
9	11/23/1940	Waterloo, Illinois	Widespread	VI	III
10	8/9/1947	Battle Creek, Mich.	50,000	VI	III

* 0.07 g maximum for a period ranging between 0.3 to 0.6 second. (6)

None of the earthquakes shown in Fig. III-E-1 can be considered as having a damaging effect on structures of adequate design located at the site.

Other strong tremors felt in the area, but with epi-centers greater than 600 km away, are as follows:

Date	Epi-Center	Extent of Tremor in Sq Miles	Maximum Intensity at Center	Estimated Probable Intensity at Site
8/31/1886	Charleston, S.C.	2,000,000	X	III
2/28/1925	Quebec, Canada	2,000,000	VIII	III

References for Section III

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5. Earthquake History of the United States - Part I, U. S. Dept. of Commerce, Coast and Geodetic Survey - Washington.
6. Seismology Division, Coast & Geodetic Survey, U. S. Dept. of Commerce, Environmental Science Services Administration, Report on Seismicity of the Grundy County, Morris, Illinois Area, (Sept. 17, 1965). (Report compiled for construction permit and operating license for Dresden Nuclear Power Station Unit #2).

IV. DESCRIPTION OF DESIGN

The design description presented in this Section reflects the design of the AARR facility as of September 1, 1967. For purposes of this report, the core package is defined as the fuel elements, control plates, control plate drive mechanisms, core structure, and the beryllium reflector, all of which are described in Section IV-B. With the single exception of the outer cylinder of the beryllium reflector, in which the locations of beam tube pockets must be varied to accommodate the AARR experiment facilities, all of the components listed as comprising the core package are identical and interchangeable with corresponding components used in HFIR.

The design is, of course, subject to the changes which are normal during the development of detailed engineering design. However, it is emphasized that a continuing safety analysis will be conducted on the detailed design as it is developed, to assure continued conformance to the established safety and design criteria (Section II). A thorough analysis of the final AARR design will be presented in the Final Safety Analysis Report.*

A. Summary Description

1. General

The AARR with a HFIR core package is a beryllium-reflected, light-water cooled and moderated, aluminum-clad fuel plate, flux-trap type reactor which utilizes highly enriched U^{235} fuel. The design power level is 100 MW. Figure IV-1 is a cutaway view of the Reactor Containment Building, showing the major areas and important equipment items. The principal design parameters of the plant are presented in Table IV-A-1.

The AARR facility is to be located on the Argonne National Laboratory site in DuPage County, Illinois, near the geographic center of the property, at a point approximately 150 m northwest of the existing CP-5 reactor facility.

2. Purpose

The AARR is designed as a multipurpose research tool to provide Argonne and midwest university scientists with facilities necessary to conduct basic nuclear chemistry, low energy physics and material science research programs to levels of advancement considerably beyond those possible in the CP-5 and other lower power research reactors currently available at the Argonne site.

3. Reactor Core

The reactor core, illustrated in Figure IV-2, consists of a series of concentric annular regions, each approximately 2 ft high. A 5-in. diameter hole penetrates the center of the core. The Internal Thermal Column (ITC) thus formed contains a removable insert which carries the ITC hydraulic and gas operated rabbit and static irradiation thimbles. The core is identical to the HFIR core.

*Ed. note: See footnote, p. 198

FIG. IV-1 CUTAWAY OF AARR REACTOR CONTAINMENT AND ACTIVE MATERIAL HANDLING BUILDINGS

TABLE IV-A-1

AARR DESIGN PARAMETERS

I.	Reactor Power Levels, MW	
A.	Steady state, operating (fission power)	100
B.	Minimum steady-state burnout	138
II.	Core Life at Full Power, days	22.5 (a)
III.	Thermal Neutron Fluxes at 100 MW, 10^{15} neutrons cm ⁻² sec ⁻¹	
A.	Maximum unperturbed in ITC	5.5
B.	Maximum in typical ITC sample (b)	1.8 (c)
C.	Average in typical ITC sample (b)	1.5 (c)
D.	Maximum unperturbed in Be reflector	
	1. Beginning of fuel cycle	1.1
	2. End of fuel cycle	1.6
IV.	Reactor Materials	
A.	Fuel plate	U ₃ O ₈ -Al cermet, Al cladding
	1. U ²³⁵ enrichment, %	~ 93
	2. Total fuel loading of U ²³⁵ , kg	9.40
	3. U ₃ O ₈ concentration in cermet, wt%	
	Inner element	30.0
	Outer element	41.0
<hr/>		
(a)	First four HFIR cores operated to 2000-2300 MW-days at 50, 75, 90 and 100 MW continuously; subsequent full power cores have averaged close to 2300 MW-days.	
(b)	Rabbit samples in a fully loaded ITC (assumed to be 75% water and 25% aluminum with additional neutron absorption corresponding to expected samples).	
(c)	Values obtained by applying ANL computational factors to maximum unperturbed flux reported for HFIR by ORNL (ORNL estimates 2.0×10^{15} for average flux in typical ITC sample).	

TABLE IV-A-1 (continued)

4. Total B ¹⁰ loading, gm	2.8 ± 10% (d)
B. Coolant	H ₂ O
C. ITC moderator	H ₂ O
D. Radial reflector	
Removable	Be + 5% H ₂ O
Permanent	Be + 2% H ₂ O
E. Shim, safety and regulating plates	
Black	Eu ₂ O ₃ + Al
Gray	Ta + Al
White	Al

V. Heat Transfer and Primary Coolant Data

A. General

1. Design pressure, psig	
Pressure vessel	1000
Rest of primary system	875
2. Normal operating pressure (vessel inlet), psig	650
3. Design coolant flow rates, gpm @ 120°F	
a. Pump capacities (total)	
1. Main (2)	18,940
2. Purification (2-1 standby)	200
b. Through fuel channels	13,800
c. Control plate region	1,080
d. Through ITC region	320 (e)
e. Beryllium reflector	1,520 (e)
f. Reflector rabbits & baskets	570
g. Beam & through tubes	500
h. Reserve flow	1,350
4. System pressure drop, psi	167

(d) Inner fuel element plates only.

(e) Includes contingency for overall needs.

TABLE IV-A-1 (continued)

5. Coolant temperatures (at 100 MW), °F	
a. Vessel inlet	120
b. Vessel outlet	156
c. Maximum hot streak or hot spot at fuel outlet	246
d. Maximum surface (fuel plate-water interface)	377
B. Fuel region	
1. Geometry	Involute plate
2. Fuel element dimensions	
a. Height of active core, in.	20
b. Total fuel plate height, in.	24
c. Fuel plate thickness, in.	0.050
d. Cladding thickness (each surface), in.	0.010
e. Fuel plate cermet thickness, in.	0.030
f. Coolant channel thickness, in.	0.050
3. Number of fuel plates, total	540
4. Total heat transfer area, ft ²	428.8
5. Volume of active core, liters	50.5
6. Power density, MW/liter *	
a. Maximum	4.38
b. Average	1.93
7. Heat flux, 10 ⁶ Btu hr ⁻¹ ft ⁻² *	
a. Hot spot	1.99
b. Average	0.773
c. Burnout at end of cycle at operating pressure	2.72

* Based on ~ 97.5 MW heat release from fuel plates when operating at 100 MW fission power.

TABLE IV-A-1 (continued)

8. Coolant flow rate, gpm @ 120°F	13,800
9. Coolant velocity, ft/sec	51
10. Pressure drop across fuel element, psi	107
11. Temperatures, °F	
a. Water inlet	120
b. Water outlet (avg)	169
c. Oxide-water interface (max)	377
d. Oxide-water interface (avg)	188
e. Metal-oxide interface (max)	461
f. Metal-oxide interface (avg)	195
g. Fuel plate (max)	480
h. Fuel plate (avg)	203
i. Saturation temperature at hot spot	467
C. Control region	
1. Geometry	Two Concentric Cylinders
2. Number of plates	
a. Regulating - shim	1 (inner cylinder)
b. Shim-safety	4 (outer cylinder)
3. Dimensions, in.	
a. Overall region radii	
1. Inside	8.567
2. Outside	9.436
b. Height within core	
1. Black region	22
2. Gray region	5
3. White region	~ 20
c. Thickness of plates	0.250
d. Coolant channel thickness	
1. Inner	0.104
2. Middle	0.129
3. Outer	0.136

TABLE IV-A-1 (continued)

4. Heat flux (max), Btu hr ⁻¹ ft ⁻²	38,000	
5. Coolant flow rate, gpm	1080	
6. Coolant velocity, ft/sec	16	
7. Pressure drop across plates, psi	9	
D. Removable reflector region		
1. Geometry	Three concentric cylinders, inner cylinder lined with aluminum	
2. Beryllium dimensions, in.		
a. Overall i.d.	18.872	
b. Overall o.d.	23.756	
c. Height	24.0	
3. Heat flux (max), Btu hr ⁻¹ ft ⁻²	0.2 x 10 ⁶	
4. Coolant velocity, ft/sec	12.5	
5. Coolant flow rate, gpm	175	
6. Pressure drop, psi	~ 9	
E. Permanent and semipermanent reflector region (including control plate access plugs)		
1. Geometry	Two concentric cylinders, inner cylinder semipermanent	
2. Beryllium dimensions, in.		
a. I.D. (semipermanent)	23.884	
b. I.D. (interface)	26.250	
c. Overall o.d.	52.0	
d. Height	24.0	
e. Heat flux (max.) Btu hr ⁻¹ ft ⁻²	0.3 x 10 ⁶	
f. Coolant flow rate, gpm	910	
g. Coolant velocity, ft/sec	13	
h. Pressure drop, psi	~ 9	
VI. Experiment Facilities	No.	Diameter, in. (f)
A. Horizontal beam tubes		
1. Blind tubes	6	5
2. Through tubes	2	5 (g)

(f) For rabbit facilities, diameter refers to sample diameter; in other facilities, diameter refers to inside diameter of facility.

(g) Each through tube may be used as two blind tubes if desired.

TABLE IV-A-1 (continued)

B. Internal thermal column		
1. Hydraulic rabbits	5	1/2
2. Gas (cooled) rabbit	1	0.1
3. Gas (fast) rabbit	1	1/2
4. Static irradiation facilities	2	1/2
C. Permanent reflector irradiation facilities ^(h)		
1. Hydraulic rabbits	2	1
	2	1/2
2. Gas rabbits	2	1/2
3. Static irradiation facilities	1	1
	1	1/2
4. Spare holes for future use ⁽ⁱ⁾	4	1-1/2
	6	2-1/2

VII. Principal Primary System Components

A. Reactor vessel

1. Inside diameter, ft	8
2. Overall height, ft	28
3. Wall thickness in main cylindrical section, in.	4-7/8
4. Upper main closure free diameter, ft	8
5. Lower closure diameter, ft	4
6. Material	} Type 304L stainless steel
Shell	
Top & Bottom Heads	

B. Primary heat exchangers

1. Number of units installed	3
2. Thermal capacity per unit, MW	33-1/3
3. Type	Horizontal shell and U-tube
4. Tube-side fluid	Secondary coolant
5. Shell-side fluid	Primary coolant
6. Materials	Carbon steel shell clad inside with stainless steel; high nickel alloy tubes

^(h) All reflector hydraulic rabbits are installed in 1-1/2 in. diameter holes; reflector gas rabbits are installed in 3 in. holes; reflector static facilities are installed in 1-1/2 in. and 2 in. holes.

⁽ⁱ⁾ A total of 16 additional irradiation facilities are available in the removable and semi-permanent reflectors, over and above those tabulated.

TABLE IV-A-1 (continued)

C. Primary coolant circulation pumps		
1. Number of units installed		2
2. Units required for 100 MW operation		2
3. Head at design flow, ft of water		470
4. Coolant flow per unit, gpm @ 120°F		
a. Normal motor drive		9,470
b. Pony motor drive (d-c)		1,900
5. Motor horsepower per unit		
a. Normal motor drive		1,250
b. Pony motor drive (d-c)		15
6. Material (casing)		304 SST
VIII. Containment		
A. Geometry		Cylindrical building with ellipsoidal dome and flat bottom.
B. Type		Reinforced concrete, steel-lined.
C. Dimensions, inside, ft		
1. Diameter		120
2. Height		120
3. Wall thickness (for strength and radiation shielding)		
a. Side walls		24 in. Concrete
b. Dome		12 in. Concrete
D. Principal design parameters		
1. Design internal pressure (j), psig		5.4
2. Leakage, % per day at maximum accident pressure (5.4 psig)		
a. Design goal		0.085
b. Guaranteed		0.85

(j) Per ASME Boiler and Pressure Vessel Code, Section III - Nuclear Vessels - Class B Vessels.

FIG. IV-2
CORE, CONTROL PLATES
AND REFLECTOR

The fuel region is composed of two concentric fuel elements. The inner one, which contains 171 curved fuel plates, has an inside diameter of 5.067 in. and an outside diameter of 10.590 in. The outer fuel element contains 369 curved fuel plates and has inner and outer diameters of 11.250 and 17.134 in., respectively.

The fuel plates are 0.05 in. thick and are curved in the shape of an involute, thus providing a constant coolant channel width. The plates are of complex sandwich-type construction composed of U_3O_8 -Al cermet held between covers of Type 6061 aluminum. To minimize the radial peak-to-average-power density ratio, the fuel is nonuniformly loaded along the arc of the involute. A burnable poison is included in the inner fuel element to further flatten the neutron flux and to reduce the negative reactivity requirements of the control plates. A core loading is expected to include approximately 9.4 kg of U^{235} and 2.8 gm of B^{10} . This will provide a maximum of 13% available excess reactivity during the core lifetime, which is expected to be a minimum of 15 days.

4. Reflector

The fuel region is surrounded by a cylindrical beryllium reflector approximately 16.5 in. thick, and 2 ft high. The inner beryllium is 3.689 in. thick and is composed of four concentric cylinders. Three of the four cylinders (2.441 in. thick) are readily removable to permit replacement, when necessary, due to radiation damage or for other causes. The outer beryllium, in which are located the major experiment facilities, is 12.875 in. thick with an outside diameter of 52 inches.

5. Control Plates

The control plates, in the form of two thin poison-bearing concentric cylinders, are located in an annular region between the outer fuel element and the beryllium reflector. These plates are driven in opposite directions by drive mechanisms located beneath the reactor. The inner control cylinder has its poison arranged so that reactivity is increased by downward motion. This cylinder is used for shimming and regulation; it has no fast safety function. The outer control cylinder consists of four separate quadrants, each having an independent drive and safety release mechanism. Reactivity is increased as the outer plates are raised. All control plates have three regions of different poison content designed to minimize the axial peak-to-average power density ratio throughout the core lifetime.

6. Control and Instrumentation

The reactor control and instrumentation system design reflects a considerable emphasis on a very high degree of safety of operation. Redundant channels, connected in two-out-of-three scram logic, are provided for monitoring of scram variables. Each channel in a redundant group is physically and electrically separated from the others, insofar as possible. An extensive independent on-line testing system permits the safety functions of any one channel to be tested at any time during reactor operation. Wherever feasible, the primary sensors are also included in the tests.

Attention has also been given to preventing unnecessary scrams, while maintaining the reactor at the highest safe power level. Interlocks in the rod control circuits, by limiting the rate of change of reactor power, and by initiating automatic corrective action to limit the power to values commensurate with the existing rate of coolant flow, forestall possible scrams.

All controls and indicators needed to safely operate the reactor are located in the central control room where they are easily accessible to the operator. During normal startup the reactor power is raised to 10 MW automatically. However, manual control is also provided. In the power range, reactor power is automatically controlled by a three-channel servo system. This system maintains either a pre-set level or a level corresponding to the coolant flow rate, whichever is less.

Ten nuclear instrument channels are provided, the detectors for which are housed in individual thimbles. The detectors for three wide-range log channels are located inside the vessel. The detectors for the remaining channels are located in thimbles outside the vessel. Aluminum neutron windows inside the vessel are utilized to enhance the neutron flux at these latter detectors.

7. Reactor Vessel and Pools

The reactor core assembly is contained in an 8-ft inside diameter reactor pressure vessel, which is located in the 17-ft diameter Reactor Pool. The principal characteristics of the reactor vessel are given in Section IV-D. The top of the pressure vessel is 17 ft below the pool water level, with the reactor core centerline 29 ft below the pool level. Adjacent to and connected with the Reactor Pool is a Spent Core Storage Pool 33-1/2 ft deep, 18-1/2 ft wide, and 13-1/2 ft long, and adjacent to the Spent Core Storage Pool is the Core Parts Storage and Transfer Pool, 21-1/2 ft deep, 18-1/2 ft wide and 12 ft long. An underwater hatch is also provided for transferring materials to the Main Canal in the adjacent Active Material Handling Building.

8. Primary Coolant System

The primary coolant enters the pressure vessel through a 24-in. pipe above the core, passes through the core, and exits through a 24-in. pipe beneath the core. The design flow rate is about 19,140 gpm, of which approximately 13,800 gpm flows through the fuel region and the remainder through the reflector, Internal Thermal Column, control plate channels, beam tubes, irradiation facilities, structures and external auxiliary systems. The system is designed to operate at an inlet pressure up to 750 psi; however, it is anticipated that normal operation at 100 MW will require an inlet pressure of only 650 psi. Under these design conditions, the inlet temperature will be 120°F, the corresponding exit temperature will be 156°F, and the pressure drop through the core will be about 110 psi.

From the reactor the coolant flow is distributed to three horizontal heat exchangers and then, through a single manifold, to two primary pumps connected in parallel which return the coolant to the reactor vessel.

A pressure control valve discharges a rated 200 gpm of primary loop water to a low pressure degasification-purification circuit; two parallel pressurization pumps (one standby) return this water to the primary loop through a flow control valve. The heat exchangers and pumps which are described in Section IV-G are grouped and shielded as a whole, and are located 30 ft below the experiment floor on the service floor of the Reactor Containment Building.

A secondary coolant system removes heat from the primary system and transfers it to the atmosphere by passing water over a three-cell induced-draft cooling tower located adjacent to the Reactor Containment Building.

9. Experiment Facilities

The primary purpose of the AARR is to provide experimenters with horizontal beam tubes for the transmission of high-intensity collimated thermal neutron beams outside the reactor vessel for physics experiments, and vertical facilities in both the Internal Thermal Column and the reflector for both short and long term sample irradiation. A total of eight horizontal beam tube facilities (6 blind and 2 through tubes), and nine vertical ITC facilities (5 hydraulic rabbits, 2 gas rabbits, and 2 static sample facilities) are provided. Other experiment facilities for sample irradiations are also provided in the reflector. These include four hydraulic rabbits, two gas rabbits, and two static vertical facilities located in various flux regions of the permanent reflector. Ten spare holes are provided for later installation of additional facilities in the permanent reflector. An additional sixteen spare holes for irradiation facilities are located in the removable and semi-permanent reflectors.

10. Buildings

The reactor, and all of the equipment and facilities described above (except as noted), are housed in a 120-ft diameter Reactor Containment Building, which extends 90-1/2 ft above grade and 39 ft-4 in. (bottom of foundation mat) below grade. The containment structure is designed to withstand a 6 psig overpressure with less than 0.1% per day leakage and includes a 24-in. concrete radiation shield wall. During normal operation the bulk of the air used to ventilate the Reactor Containment Building is exhausted unfiltered through a 250-ft stack. In the event of a minor release of fission products or other activity to the building, the volume of exhausted air may be reduced and passed through absolute and charcoal-bed filters prior to stack release. Should the activity level exceed a maximum permissible instantaneous rate of stack discharge, the containment isolation valves close and the reduced flow filter system continues to operate, recirculating the air within the Reactor Containment Building to reduce the activity level therein. All suspect or contaminated liquid and solid waste is collected, monitored, and either discharged or removed for processing elsewhere at the ANL site, in accordance with established Laboratory procedures. The adjacent Active Material Handling Building houses an extension to the inner transfer canal and a rabbit handling pool, various mechanical equipment, and the normal air supply handling equipment for the Reactor Containment Building.

In addition to these buildings, a two-level Electrical Equipment Building averaging 58 ft x 64 ft houses the normal power switchgear and the emergency power systems. The Laboratory and Office Building, containing approximately 21,600 ft² of floor space, houses the reactor control room and special radiochemical laboratories associated with the rabbit experimental facilities. The buildings are located contiguous to the Reactor Containment and Active Material Handling Buildings.

11. Utilities

The site is supplied with 13.8 kV electrical power, 30 psig potable and fire protection water, 30 psig laboratory water, 55 psig cooling tower make-up water from the nearby Chicago Sanitary and Ship Canal, natural gas, and 200 psig steam. All other materials and utility services are shipped in bulk or generated at the site. Two diesel-electric generators and two sets each of 250 V and three sets of 64/32 V batteries are available to furnish emergency power when required.

B. Reactor Components

The arrangement of reactor components within the reactor vessel is shown in Figures IV-3 and IV-4. The fuel elements, control plates, ITC, and reflector are supported by concentric cylindrical structures called pedestals, which are bolted to a washer-shaped structure called the vessel adapter. This is in turn bolted to the side of the pressure vessel. A pair of cylindrical shrouds extends above the top surface of the fuel element. These shrouds enclose the control plates and provide the required coolant flow characteristics and the necessary protection from the turbulence of the vessel inlet coolant stream.

Since the reactor internal components are removable from the reactor pressure vessel a high degree of flexibility is provided for inspection or future design modifications.

1. Fuel Elements

The function of the core is to provide a controlled nuclear heat source capable of supplying an unperturbed flux of at least 5×10^{15} neutrons/cm²-sec to the Internal Thermal Column at 100 MW operation. The core assembly will provide the reactor system with fuel elements that can be handled safely in loading and unloading the reactor.

The core, as shown in Figure IV-5, consists of two concentric fuel elements. Exploded views of the core composite are shown in Figures IV-6 and IV-7.

The HFIR fuel elements that will be used in AARR design are described in ORNL-3572¹ as follows:

"The fuel region of the HFIR, shown in Figure IV-5 is composed of two concentric, cylindrical fuel elements containing vertically oriented curved plates extending in the radial direction. The inner element, which contains 171 plates, is initially loaded with 2.6 kg of U²³⁵ and 2.8 g of B¹⁰. The inner diameter is 5.067 in., and the outer diameter is 10.590 in. The outer element contains 369 plates and initially contains 6.8 kg of U²³⁵ but no burnable poison. Its inner diameter is 11.250 in., and the outer diameter is 17.134 in.

"The individual plates are of a sandwich-type construction composed of a fuel-bearing cermet hermetically sealed between covers of type 6061 aluminum. The fuel-bearing cermet is a mixture of U₃O₈ and aluminum, approximately 30 wt % U₃O₈ in the case of the outer annulus. Uranium containing at least 93% U²³⁵ is used as fuel. Initially each inner annulus plate contains 15.18 g $\pm$ 1.0% of U²³⁵; each outer annulus plate contains 18.44 g $\pm$ 1% of U²³⁵. In addition, each inner annulus plate initially contains 0.0164 g of B¹⁰. The finished plates are 0.050 in. thick and 2 ft long, with a nominal active length of 20 in.

FIG. IV-3
REACTOR VESSEL VERTICAL SECTION

REACTOR VESSEL HORIZONTAL SECTION A-A

FIG. IV-4

FIG. IV-5
CORE ASSEMBLY

FIG. IV-6
EXPLODED VIEW OF CORE

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FIG. IV-7
CORE AND REFLECTOR SUPPORT

The thickness of the fuel plate core is 0.030 in., and the cladding thickness is 0.010 in. The plates are curved in the form of an involute which provides a constant cooling channel width between plates; the width is 0.050 in. Before bending, the inner annulus plates are approximately 3.6 in. wide, and the outer annulus plates are approximately 3.2 in. wide. The fuel cores are centered to provide adequate edge cladding. Both inner and outer plates are fastened between cylindrical aluminum side plates by welding.

"To minimize the radial peak-to-average power density ratio, the fuel surface density in each plate is varied along the arc of the involute curve. The B^{10} included in the inner plates in the form of B_4C , is added to an aluminum filler piece, which together with the fuel-bearing cermet makes up a composite, rectangular fuel plate core, as shown schematically in Figures IV-8 and IV-9."

Originally, the HFIR core was expected to have a lifetime of 15 days at 100 MW. However, operation at 100 MW has shown that the reference HFIR core can be run safely at 100 MW for periods in excess of 22 days.

The ORNL mechanical, hydraulic and steady state heat transfer analyses, as presented in ORNL-3572¹ have been used as the basis for this report. Pertinent extracts from ORNL-3572, giving some of the results of those analyses, are reproduced below.

"Mechanical and Hydraulic Analysis"²⁻⁴

"The fuel plates in the HFIR are subjected primarily to two types of loading: (1) that resulting from hydraulically induced, lateral pressure differentials across the fuel plates and (2) that resulting from thermally induced, differential radial and longitudinal expansions between adjacent fuel plates and side plates.

"The hydraulically induced loads exist because small dimensional differences cause velocity and pressure differences between adjacent coolant channels. This phenomenon was integrated into the overall fuel element analysis by developing analytical expressions for the local velocity distribution, velocity pressure drop, and frictional pressure drop in channels distorted by gradual deviations...From these equations it was possible to calculate the lateral pressure differentials across a fuel plate.

"Deflections and stresses resulting from the above pressure differentials were studied both analytically and experimentally for the involute geometry fuel plate. Results of these studies indicated that when the involute was subjected to a uniformly distributed load,

FIG. IV-8
FUEL AND BURNABLE-POISON DISTRIBUTION
IN THE AS-BUILT ELEMENT

UNCLASSIFIED
ORNL-DWG 63-3672

FIG. IV -9
CROSS SECTION OF FUEL PLATE BEFORE BENDING

the deflected shape was in the form of an S, relative to the initial involute contour; that is, both positive and negative deflections existed... The advantage of this condition is that the total cross-sectional flow area of a channel tends to remain constant as adjacent plates deflect; this reduces further changes in the lateral pressure differentials and thus reduces the plate deflection. A further advantage of the above involute-curve deflection characteristic is that the critical velocity tends to be quite high as compared with a uniformly curved plate. The estimated critical velocity for buckling-type failure for the HFIR fuel elements is well above the maximum operating coolant velocity.

"Thermally induced deflections of the plates were also studied both analytically and experimentally. Although both radial and longitudinal expansion exist, the longitudinal expansions are by far the more significant. In this case the important temperature difference is that between the fuel plate and the two side plates. Since the fuel plate is hottest, it is in longitudinal compression and tends to buckle... This condition was predicted analytically and measured experimentally in a simulated test. The experimental data were used with the approximate analytical expressions to arrive at appropriate empirical formulae, which permitted the inclusion of this phenomenon in the overall analysis.

"In the above pressure and thermal deflection tests the creep behavior of the fuel plates at appropriate temperatures was also studied, and the results were incorporated into the fuel element analysis.

"Results of Heat-Removal Analyses

"Some of the results of the HFIR heat-removal calculations are presented in Table IV-B-1. Heat flux and operating-temperature values are listed for both beginning and end of cycle for the average fuel element conditions, the outlet from the maximum hot streak, and the maximum hot spot. As would be expected, buildup of the oxide film on the fuel-plates results in a decrease in incipient boiling power level (from 146 to 138 MW) as the fuel cycle progresses. The oxide buildup also results in an appreciable increase in plate temperatures as evidenced by the hot-spot metal temperature, which increases by about 100°F during the fuel cycle. Since plate strength is dependent upon the hot-plate temperatures (which are somewhat lower than the hot-streak values listed here), it appears from these data that, with the possible exception of very local hot spots, the temperatures are such that plate strengths can be expected to remain at reasonable values throughout the fuel cycle.

TABLE IV-B-1

RESULTS OF HFIR HEAT REMOVAL CALCULATIONS

Total reactor power, MW	100					
Power density (97.5 MW in fuel region), MW/liter						
Average	1.93					
Maximum	4.384					
Heat flux (97.5 MW conducted to coolant), Btu hr ⁻¹ ft ⁻²						
Average	7.76 x 10 ⁵					
Hot spot	1.97 x 10 ⁶					
Incipient boiling power level at steady state, MW (with 600-psi core inlet pressure)						
Beginning of fuel cycle	146					
End of 15-day fuel cycle	138					
Coolant velocity (average), fps	51					
Temperatures, °F (with 121°F inlet water temperature)						
	<u>Start of Cycle</u>			<u>End of 15-day Cycle</u>		
	<u>Fuel Average</u>	<u>Hot Streak Outlet</u>	<u>Hot Spot</u>	<u>Fuel Average</u>	<u>Hot Streak Outlet</u>	<u>Hot Spot</u>
Fuel plate temperature, °F	205	312	390	212	360	480
Metal-oxide interface temperature, °F	197	303	371	204	351	461
Oxide-water interface temperature, °F	197	303	371	197	303	377
Water outlet temperature, °F	178	246	246	178	246	246

"Buildup of the oxide film also produces another interesting result in that the arrangement of wide and narrow coolant channels that produces the lowest incipient boiling power level at the start of the cycle is not necessarily the one that is the nearest to boiling at the end of the cycle...

"The effect of coolant-channel tolerances on minimum calculated incipient boiling power level is shown in Fig. IV-10. These curves indicate that for a constant average channel thickness tolerance, a change in local channel thickness tolerance of 5 mils results in about a 9% change in incipient boiling power level. A 5 mil change in average channel thickness tolerance, with the local tolerance constant, would result in about a 11% change in incipient boiling power level. Studies such as this have been used to establish the required fuel element fabrication tolerances.

"The results of the fuel element design analysis indicate that a fuel element constructed to the specifications employed in the heat-removal calculations is capable of meeting the overall HFIR performance specifications."

2. Control Plates

The HFIR control plates and reactor control scheme to be used in the AARR reactor are described in ORNL-3572¹ as follows:

"The control plates are located in a 0.869-in.-thick annular region between the outlet fuel element and the removable beryllium. The control plates consist of two 1/4-in.-thick concentric cylinders. The coolant passages, listed radially outward, are 0.104-in., 0.129-in. and 0.136-in. The coolant water velocity in the control region is about 16 fps. The inner cylinder, which is a single piece, is used both as a shim and a regulating rod. The outer cylinder is divided into four quadrants, each used as a shim-safety rod and each having its own drive rod and scram mechanism (see Section IV-H for complete description of control action). The general arrangement is shown in Figures IV-3, IV-6, and IV-11 through IV-13.

"The chain reaction is controlled by altering the efficiency of the beryllium reflector as the control plates are moved vertically between the core and the reflector. The single inner cylinder is driven downward, thus driving the poison out of the reactor to increase reactivity, while the outer quadrants are driven upward out of the reactor to increase reactivity.

"In order to reduce axial variations in the power distribution and to prolong the life of the control elements

FIG. IV-10
EFFECT OF COOLANT-CHANNEL-THICKNESS
TOLERANCES ON BURNOUT POWER LEVELS

SOURCE: ORNL - DWG 65-2958

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ORNL-DWG 63-3580R

FIG. IV-11
CONTROL PLATES

FIG. IV-12
CONTROL PLATE POSITIONS
DURING A CYCLE

SCRAM PLATE MAGNET AND LATCH MECHANISM

FIG. IV-13

the cylinders are divided into three longitudinal sections, each of which has different neutron absorbing characteristics. The lower section of the inner cylinder and the upper section of the outer cylinder contain 33 vol % Eu_2O_3 dispersed in aluminum. These sections are highly neutron absorbing and are called the black regions. The central section of each plate contains 40 vol % tantalum dispersed in aluminum. They are less absorbing than the Eu_2O_3 sections and are called the gray regions. The upper section of the inner cylinder and the lower section of the outer cylinder are made of aluminum and are called the white regions. In normal operation criticality is achieved by driving in unison the inner cylinder down and the outer quadrants up. As shown in Figure IV-12, the two control cylinders are then gradually withdrawn to compensate for fuel burnup throughout the core life and are maintained symmetrical about the core midplane so as to maintain symmetry of the power density in the axial direction. A servo control system maintains constant power by moving the inner cylinder.

"In each control element the black region is 22 in. long, 3/16-in.-thick and is clad with 1/32-in.-thick aluminum. The gray regions are 5 in. long and have the same thickness of absorber and cladding as is the case in the black region. The white regions are of solid aluminum. In order to balance the hydraulic forces, a large number of 1/4-in. holes are drilled through the white and the gray regions of the five control elements. Corrosion studies indicate that edge cladding around these holes is not necessary in the gray region.

"The inner control cylinder has an outer radius of 8.921 in. and an inner radius of 8.671 in. The overall length is 68-3/8 in. The lower end of the black region is 14-5/16 in. above the lower edge of the cylinder, and the upper end of the white region is 7-1/16 in. below the upper edge of the cylinder. The outer control plates have an outer radius of 9.300 in. and an inner radius of 9.050 in. The overall width of each is 13-39/64 in., and the overall length is 66-3/16 in. The upper end of the black region is 8-5/8 in. below the upper end of the plate, and the lower end of white region is 10-9/16 in. above the lower end of the plates.

"The cylinder and plates are driven from beneath the reactor by drive rods which extend into the subpile room, where the drive mechanisms are located. These mechanisms are described in detail in Section IV-B-4. Guidance for each outer control plate is furnished by six ball bearings. Four bearings are attached to each outer control plate, two at each end, and are run in stationary tracks which extend above and below the core. The other two bearings are attached to the lower track assembly and bear

against the control plate. The inner control cylinder is guided by eight bearings, four of which are attached to the track assemblies at each end of the reflector and extend through the slots between the four outer plates."

A total flow rate of about 1100 gpm is provided for the cooling of the control plates. A pressure differential of about 9 psi, which is provided by orificing the control plate flow at inlet of the core, is adequate for a water velocity of 16 ft/sec over the plates.

The maximum heat flux is 3.8×10^4 B/hr ft² while the maximum surface temperature is about 205°F.

3. Reflector

The reflector as shown in Figure IV-3 provides a 16.5 in. thick cylindrical region of beryllium surrounding the core in the radial direction. The beryllium reflector extends 1 ft above and below the centerline of the active fuel region. No beryllium is placed directly above or below the core. Some reflection of neutrons is provided in the axial direction by the cooling water above and below the core. The design of the reflector follows closely that of the HFIR and the description of this segment of the core internals is reproduced from ORNL-3572¹:

"The inner portion of the reflector is readily removable to permit replacement, when necessary due to radiation damage or other causes, and to allow access to the control plate drives. The reflector is water cooled; but in order to avoid excess neutron absorption, the volume of water in the reflector is held to the minimum consistent with the cooling requirements.

"The inner portion of the reflector consists of a 3.689-in.-thick beryllium section composed of four concentric cylinders with cooling water flowing axially between them. The outer part of the reflector is "a 12.875-in.-thick" beryllium annulus provided with axial circular coolant channels. Three of the four inner cylinders, called the removable reflector, may be removed from the reactor as a unit, thus permitting access to the control plate shock absorbers. The fourth cylinder, called the semipermanent reflector, is removed only when necessary because of radiation damage. Four small pieces of the semipermanent beryllium, called control plate access plugs, are easily removable to facilitate access to the control drives. Although the thick outer annulus, called the permanent reflector, may be removed, it is anticipated that this would be done very infrequently.

"To minimize the chance, should the beryllium crack, of interference with the control plates which are located between the removable reflector and the fuel region, a 1/16-in.-thick aluminum liner covers the inner surface of the removable beryllium. Coolant for these removable reflector

cylinders is provided by water flowing through 1/8-in. axial grooves on the surface of these cylinders. A 1/16-in.-thick water gap between the removable and semipermanent reflectors facilitates removal and replacement of the removable reflector, furnishes cooling, and provides for expansion of the removable reflector, furnishes cooling, and provides for expansion of the removable reflector. Eight 1/2-in.-diam vertical facilities are provided in the middle removable reflector cylinder, and eight others are located in the removable control plate access plugs. They are provided with beryllium plugs which will be in place when the facilities are not in use."

The permanent reflector is cooled by water which flows axially through 1/8 in. diameter coolant holes. The number of holes and the pattern of the coolant holes will be similar to the HFIR design. The coolant holes will be spaced to maintain all portions of the beryllium at temperatures not to exceed 290°F.

It is expected that the helium gas generated in the beryllium due to irradiation will have very little effect on density or cause any swelling for neutron doses of approximately 1.4×10^{22} n/cm² fast (> 1 MeV). At AARR temperatures the volume increase is small since the threshold temperature for helium agglomeration is about 825°F.

Calculations indicate that the estimated lifetime of the beryllium in the various regions of the reflector will be approximately 0.6 yr for the removable reflector, 2.2 yr for the semipermanent and 6 yr for the permanent reflector.

The flow through the reflector is orificed at the top of the core to provide a pressure differential of about 9 psi for the 1/8 in. coolant holes. The resulting velocity for the 120°F inlet water is about 12.5 ft/sec in the removable reflector and requires a flow rate of 175 gpm. The maximum surface temperature is about 185°F for a maximum heat flux of 2×10^5 B/hr ft². The nominal water temperature from the reflector region is about 166°F.

The permanent reflector which is 4.5 inches larger in outer radius than the HFIR reflector, and the semipermanent reflector, will utilize the 9 psi pressure differential to provide a coolant velocity of about 13 ft/sec. For the 120°F inlet water temperature the maximum surface and internal beryllium temperatures are 230°F and 290°F, respectively. The maximum heat flux is about 3×10^5 B/hr-ft² for these conditions which results in an average outlet temperature of about 134°F. An estimated 910 gpm is required for the permanent reflector.

* A total flow rate of about 1500 gpm is allotted for the reflector. The excess (over the 1,085 gpm required) is a contingency for overall needs and is presented for consistency in engineering calculations.

The experiment facilities that penetrate the permanent reflector are discussed in Section IV-E.

4. Control Plate Drive Mechanisms

The function of the control plate drive mechanisms is to provide uniform vertical movement to the control plates during reactor operation and also to insert the outer control plates into the core when rapid shutdown is required.

The control drive mechanisms are located beneath the reactor and are attached to the bottom head of the reactor vessel. These mechanisms and their function are identical with those used in the HFIR. For convenience the following description of these mechanisms is reproduced from ORNL-3572¹.

"a. Introduction

"Because the control plate arrangement in the HFIR is somewhat different from that usually encountered, and because a knowledge of how the control plates function is essential in order to understand the workings of the control and safety systems, the control drive mechanisms will be described before embarking on a discussion of the control and safety systems.

"As previously described, control of the reactor is effected by the movement of two concentric cylinders located between the fuel and reflector regions of the reactor (Fig. IV-2). This configuration was chosen in order to minimize the perturbation of the power distribution caused by the withdrawal of the control plates as burnup of the core occurs (see Fig. IV-12). The control elements are moved by mechanisms located in the subpile room beneath the reactor and shielded from it by water, steel, and high-density concrete. The connecting elements between the drive mechanisms in the subpile room and the control elements in the reactor vessel are drive tubes of stainless steel which extend upward from the subpile room through seals in the reactor vessel lower head.

"The inner cylinder, the shim-regulating plate, moves downward to increase reactivity. Because this is the direction of gravitational, hydraulic, and flow forces, this control element is bolted securely to the drive tube and moves only when motor driven; it has no provision for fast scram action. When used for shimming, it will be referred to as the No. 5 shim plate, and when used for regulation, as the regulating rod. The outer control cylinder is divided into four quadrants, each of which is provided with a separate control drive mechanism and a quick-acting scram latch. These plates, shim-safety plates 1 to 4, move downward in the direction of the gravitational, hydraulic, and flow forces to decrease reactivity.

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"b. Shim-Safety Drives

"The safety release mechanisms are of the same general type as those used in the ORR and the ETR (i.e., the ball-latch type). This mechanism consists of a latch head located near the top of the drive tube, in the reactor vessel, with a latch push rod extending downward into the subpile room through the center of the drive tube; a seal between the push rod and the drive tube is provided at the lower end of the drive tube. This latch push rod is moved upward relative to the drive tube in order to engage the latch and is held in position against the force of a release spring by an electromagnet. Shutoff of the currents in the magnet results in release of the latch mechanism. The latch head consists of an outer cage with the same outer diameter as the drive tube. This cage has holes through which a circle of balls may be partially extended by the tapered surface of the latch plunger, which is located inside the cage and which is attached to the upper end of the latch push rod. When the balls are extended, the latch is "cocked" and is held in position by the electromagnet. A hydraulic cylinder assembly allows for thermal expansion of the push rod, but forms a solid connection when the push rod makes the rapid movement to initiate a scram (see Fig. IV-13).

"Concentric with the drive tube, but entirely within the reactor vessel, is a hollow control-plate extension tube to the top of which the bottom of the shim-safety quadrant is attached by means of a self-aligning coupling. Inside of this control plate extension tube, a shoulder is provided which has an inside diameter slightly larger than the drive tube but smaller than the diameter of the circle of extended balls. Since the latch is cocked while the balls are below the latch shoulder of the extension tube an upward movement of the drive tube will lift the extension tube and thus raise or withdraw the shim-safety quadrant. If, while the shim-safety quadrant is raised, the magnet is de-energized, the latch push rod is pushed downward by pressure forces and springs, the balls retract, and the control plate and its extension tube are free to move downward, reducing the reactivity.

"The lower end of the control plate extension tube forms a shock-absorber piston which enters a shock-absorber cylinder near the lower end of the stroke to decelerate the safety quadrant and extension tube following a scram.

"The design criteria specify that the shim-safety plates start moving within 0.01 sec after a safety parameter is exceeded. In addition, the travel of the safety plates following release shall be equal to, or

greater than, that which would be obtained if they had an initial acceleration of $4 \times g$ which decreased to $1 \times g$ at the end of 6 in. of travel.

"The scram latch mechanism is designed to be the coincidence element in the two-out-of-three scram circuit. The output of each safety channel is the controlled current in one of three electrically independent coils in each scram-latch magnet. These magnets are therefore the elements which determine the behavior of the scram latches on the basis of the outputs of the three safety channels. The first increment of excitation is provided by any one safety channel and produces flux only in the yoke and shunt, the air-gap reluctance being high enough so that the armature flux is negligible. As the excitation is increased, corresponding to the output of two safety channels, the yoke, armature, or both saturate. The resulting force on the armature is negligibly small when the shunt is unsaturated and rises rapidly (approximately as the square of the excitation) from nearly zero at the shunt-saturation point to full force at the armature-saturation point. By the proper choice of coil turns and current, it has therefore been possible to develop a magnet which exerts virtually no holding force if one coil is energized and exerts the full force if two or three coils are energized. This magnet is thus a two-out-of-three coincidence element and will release if any two of the safety channels are de-energized but will hold if only one is de-energized. One of the advantages of this system lies in its adaptability to on-stream testing.

"The drive mechanisms are of the type using a non-rotating lead screw of the acme thread type with a rotating nut. The lead screw angle is chosen so that the drive is self-locking; that is, a force applied to the lead screw will not result in rotation of the nut. The nut for each drive is driven through suitable gearing by a-c motors of the two* phase reversible type. All drives are designed for the same rate of travel since it is desired to operate them in unison. The rate of travel is 5.75 in./min with the drive motors operating steadily at their maximum unloaded speed.

"The scram plate drives are also provided with a nonreversible air-powered motor coupled through appropriate gearing to provide for unidirectional high-speed insertion of the drive following a scram or when needed during an a-c power outage. The design speed is approximately ten times the speed developed by the electric motors, or about 60 in./min.

* AARR will use three phase motors.

"All drives are provided with position-sensing and -transmitting equipment which is used to transmit information to the control room for use in the control system and for the operator's information. In addition, a variety of limit switches are provided as necessary to indicate and limit drive position, latch operator position, etc. Seat switches are also provided to indicate independently of the drive mechanisms when the shim-safety plate is fully inserted.

"c. Shim-Regulating Cylinder Drives

"The shim-regulating drive for the inner control cylinder does not have a safety release mechanism. The drive system for this control element consists of a drive tube which extends from the subpile room, through a seal in the vessel lower head, to a support ring, which is the lower part of the inner control cylinder. This support ring is fastened to the upper end of the drive tube by a coupling which allows for some misalignment between the control cylinder and the drive tube.

"The drive gearing provided for shim motion of this unit is similar to that for the four outer plates; however, no safety release mechanism or fast run-down air motor is provided. The design speed is the same as for the outer rods (i.e., 5.75 in./min). The system is provided with limit switches, position sensors, and transmitters similar to those for the outer drives.

"The inner cylinder has a regulating drive of limited stroke which operates independently of the shim drive to move the shim-regulating cylinder under automatic control. The regulating system is unusual in that three independent channels are provided, the output of each being the angular velocity of a motor shaft. The three channels are combined in a servo input gear box which has a single output shaft driving the nut of a ball-bearing lead screw which has a limited stroke. The servo input gear box contains two differential gear mechanisms which provide the velocity addition required to cause the shim-regulating plate to be moved at a velocity proportional to the algebraic sum of the velocities of each of the three servo motors. The input gearing for each motor is of the worm-worm wheel type and is self-locking so that a de-energized motor will not be rotated by the other two motors or by the load. The servo motors are of the low-voltage printed-circuit type and for normal rated output require 48 v dc at 16 amp.

"Designed speed of the regulating motion with all three motors driving in the same direction is 16 in./min with the stroke limited by mechanical stops in addition

to the limit switches provided for normal operation. The regulating drive position is transmitted to the control room in addition to the shim drive position in order to enable the operator to supervise the servo systems. The shim drive position transmitter always provides information on the plate position; however, the regulating drive position indicator only supplies the position of the drive mechanism.

"In addition to the drive gearing and motors for both shim and regulating functions, the central plate drive is provided with a "backup cylinder." This oil-filled hydraulic cylinder is attached to the lower end of the shim drive lead screw and has the primary function of limiting the rate of travel of the central drive rod in the unlikely event of complete failure of the rotating nut or regulating drive gearing. As a secondary function, the hydraulic cylinder is used to balance out most of the force caused by the coolant pressure in the reactor vessel being transmitted through the drive tube to the mechanisms. This force balancing serves to reduce the loading on the gears, thereby reducing the wear and also allowing better operation of the automatic power regulation system. The force balancing is accomplished by having the hydraulic cylinder piston rod approximately equal in area to the guide tube emerging from the reactor vessel and by operating the hydraulic cylinder at the same pressure as that in the reactor vessel. This pressure is obtained by interconnecting the cylinder and the reactor vessel through a double bladder-type accumulator system, which requires no instrumentation for operation and which will maintain the same pressure even if the accumulator bladders should leak. Two accumulators are provided in order to have a buffer zone of water between the reactor water and the hydraulic oil so as to minimize the possibility of contaminating the reactor system with hydraulic oil. The speed-limiting feature of the backup cylinder does not depend on having reactor pressure in the hydraulic cylinder since this is accomplished by selection of the clearance between the cylinder wall and the piston.

"The hydraulic piston rod area is somewhat smaller than the guide tube area in order that a net downward force be present in normal operation. The downward force is chosen to be sufficient to keep the gears of the gear drives loaded and thus minimize backlash for good servo operation."

C. Fuel Handling System

The fuel handling system is similar to that now used for handling HFIR cores. All movements of active elements are done visually and under water. The active cores will be stored in a pool of water inside the Reactor Containment Building.

1. Spent Core Handling

After reactor shutdown and before core removal is started, the spent core is cooled by forced circulation until the remaining decay heat can be removed by natural circulation. The primary cooling system pressure is reduced to atmospheric pressure.

In preparation for core removal, the first operation is to disconnect remotely and remove all external connections to the Internal Thermal Column (ITC) facility. The ITC insert is released from the top plug by rotating a breech lock, removed from the reactor vessel and stored in the Reactor Pool. The 30-in. diameter quick-opening closure in the vessel head is then removed and stored in the Reactor Pool. The removal of this closure permits direct viewing of the top of the core from the personnel bridge.

After the necessary hold-down devices are removed, the handling fixture is secured to the inner element. The inner element is then slowly lifted out of the reactor vessel and transported to the Spent Core Storage Pool located adjacent to the Reactor Pool in the Reactor Containment Building, and placed temporarily in one of the special storage racks provided. The length of the handling tools and the hoist lifting height are designed to prevent the fuel elements from being lifted above elevation 769'-6" (i.e., there will always be at least 10 ft of water above the active fuel). The storage racks are elevated to permit free natural circulation of pool water through the core to assure continuous removal of decay heat, and are equipped with an upward-projecting poison-bearing material rod or tube at the center which fills the ITC space in the center of the core when the core is in the rack. The procedure is repeated to remove the outer fuel element from the reactor. The inner fuel element is then moved from its temporary location and mounted within the outer element (with the center penetrated by the poison center rod) for long-term decay storage. A grating is placed over the cores in the storage facility to prevent any large objects from falling on them. Figure IV-14 shows the HFIR storage rack details. The AARR design will be similar.

After proper decay time has expired, the fuel elements and core parts are transported individually from the Reactor Containment Building to the Active Material Handling Building through an underwater hatch in the containment wall. In this building the fuel elements are placed in a shipping cask located in the Main Canal. The cask is removed from the canal, decontaminated, and transported from the building by truck.

2. New Core Handling

The new fuel elements will be shipped to ANL as units by the fuel supplier. Prior to transfer to AARR, the new elements will be inspected

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FIG. IV-14
IN-POOL STORAGE RACK

for damage and may be subjected to physics measurements in another ANL facility to assure compliance with specifications. If not immediately needed at AARR, the fuel elements will be dried, placed in a shipping container and stored elsewhere.

Upon arrival at AARR, the vehicle carrying the fuel elements in their container is backed into the truck aisle in the Active Material Handling Building through a 12 ft x 12 ft door. The fuel elements are then removed from the container, examined, and stored in the Active Material Handling Building Special Material Store Room. Prior to refueling, a new core (two elements) will be brought into the Reactor Containment Building on a special hand-cart through air lock No. 2. This air lock is adjacent to the Special Material Store Room and is at approximately the same elevation as the walkway around the top of the pool. Each element is transferred from the hand-cart to the top of the Reactor Pool and placed on a special platform in the pool by means of special fuel element handling tools. During actual refueling the personnel bridge and the rigid fuel handling tools (similar to those used in HFIR) are used to lower each element into place in the reactor vessel.

Safety and subcriticality measures similar to those employed in the HFIR will be used in the AARR.

After the fuel elements are installed, the personnel bridge crane is used to insert the fuel hold-down devices, other structures, and the breech plug. The ITC insert is then installed, after which connection of the rabbit lines is effected. The final operation is a pressure test to assure proper closure of the vessel prior to resuming operation.

3. Ruptured Fuel Handling

In case of a ruptured or damaged core leaking fission products, the handling procedures differ from those used for normal spent core handling. The reactor primary water is circulated and its radioactivity reduced by natural decay and by removal of fission products by means of the primary purification system. When the hazards are reduced to a tolerable level, the core is removed from the reactor in the same manner as a normal spent core, except that each element is transported directly to a defective element storage tank shown in Figure IV-15. During the decay period the elements remain in these tanks, and further contamination of the canal system is thus prevented. After decay, leaky or ruptured fuel elements are inserted into separate sealed containers and transferred to the Main Canal for loading into shielded casks provided with adequate cooling means for shipment from the site. The defective element storage tank and auxiliary systems are discussed in Section IV-G below.

FIG. IV-15
DEFECTIVE ELEMENT STORAGE TANK

D. Reactor Vessel

1. General Description

As shown in Figure IV-16, the reactor vessel has an overall height of about 28 ft and an inside diameter of 8 ft except in the bottom cylindrical section where the inside diameter is 47 in. The vessel has a full-opening bolted flat head top closure and a bolted flat head bottom closure which provides full access to the smaller diameter bottom extension. The single coolant inlet and outlet nozzles, located above and below the core respectively, are 24-in. nominal diameter. The vessel wall is about 4-7/8 in. in thickness throughout the midsection in the region of the beam tube nozzles.

The top head also incorporates a 30-in. clear diameter quick-opening closure for the removal and replacement of reactor cores. Numerous penetrations for experiment and instrumentation facilities are located in the top and bottom heads. Control rod penetrations are located in the bottom head. The quick-opening closure plug has a centrally located penetration for the insertion of the Internal Thermal Column irradiation facility.

The central core region of the vessel shell is penetrated by ten nozzles for the horizontal beam tube facilities. These penetrations are aligned with openings in the Reactor Pool shield tank and biological shield. A plan view of the beam tube nozzle orientation is shown in Figures IV-4 and IV-17.

The reactor vessel is completely submerged in a reactor pool approximately 38 ft deep which is enclosed by a stainless-steel shield tank forming the inside surface of the concrete biological shield. The 17 ft of water in the pool above the top head of the vessel serves as a biological shield during reactor operation and fuel transfer.

The vessel is supported by lugs (located below the core centerline) which rest on a support structure. The supports are placed close to the core mid-plane in order to minimize vertical movement at the point where the beam tubes penetrate the vessel. Hence, the expansion of the vessel causes little vertical displacement of the beam tube nozzles. A bellows connection between the biological shield penetration and beam tube allows some differential movement between the reactor vessel and beam tube.

The reactor vessel contains high purity primary coolant water (at a pH of 5.0 to 5.3) under normal operating conditions and is entirely submerged in a pool of demineralized water at a pH of 7.0 which is normally open to the Reactor Containment Building atmosphere. Coolant water enters through the top inlet nozzle, passes through the core and reflector, and exits through the bottom outlet nozzle. The portion of the vessel above the pedestal support ring, including the top head, is exposed to water at 120°F. The portion of the reactor vessel below the pedestal support ring (below core) and above the smaller cylindrical extension is exposed to 156°F water at 100 MW operation.

FIG. IV-16
REACTOR VESSEL

PENETRATION SIZES
SHOWN VIEW A-A

2 $\frac{1}{4}$ DIA. HOLE	2 $\frac{1}{2}$ DIA. HOLE
H1-1 H2-1	H1-2
H1-6 H2-2	H1-3
H1-8 H2-3	H1-4
H1-9 H2-4	H1-5
H2-5	H1-7
	H1-10
5 $\frac{1}{4}$ DIA. HOLE	5 $\frac{3}{4}$ DIA. HOLE
H3-1	H4-1
H3-2	H4-2
H3-3	H4-3
	H4-4
	H4-5

FIG. IV-17
REACTOR VESSEL CROSS SECTIONS

2. Design Requirements

The reactor vessel and appurtenances will be designed for 1000 psig and 250°F and fabricated, inspected, and tested in accordance with all the rules of Section III of the ASME Boiler and Pressure Vessel Code for Class A (reactor) vessels, all applicable Code Cases and all applicable case interpretations, except that stress intensity limits as given in Section III of the ASME Code may be reduced in the region of the beam tube penetrations. The reactor vessel shell and closure heads will be constructed entirely of Type 304L stainless steel.

In general, the design, fabrication and inspection procedures established for pressurized water and boiling water power reactors are applicable to the AARR pressure vessel. However, certain features of the AARR reactor system place added emphasis on the design and fabrication of the reactor vessel. These are: (1) numerous non-radial nozzle connections for beam tubes located approximately on the core centerline, (2) numerous penetrations through the top and bottom heads, and (3) substantial loadings of the vessel wall imposed by the high pressure-drop across the high-power-density core. Table IV-D-1 gives a summary of reactor vessel design parameters.

The reactor vessel consists of a 96-in. inside diameter cylindrical shell section, a top closure flange, a flat head top closure, a 47-in. inside diameter cylindrical extension piece, a hemispherical transition piece between the 96- and 47-in. cylinders, bottom closure flange, and a flat bottom head. The overall height of the cylindrical section measured from the gasket face of the top closure flange to the gasket face of the bottom closure flange is approximately 25 ft-4 in. Fourteen nozzles (two flow, two poison injection,* and ten beam tube) are provided in the vessel shell. The inlet and outlet primary coolant nozzles are 24 in. nominal outside diameter and the two emergency poison injection nozzles are 2-1/2 in. nominal pipe diameter.

The midsection of the vessel, which is 8 ft in inside diameter, provides a circumference that leaves adequate ligament area between the ten horizontal beam tube nozzle penetrations. The arrangement and number of penetrations represents the maximum number of facilities that can be accommodated by the vessel within this region without exceeding permissible stress intensities. Beam tube and through tube nozzles will be accurately installed so that the centerline of the penetrations will be properly aligned with their respective holes in the reflector.

Two types of beam tube nozzles are provided in the vessel shell; one is for the blind tubes and the other for the through tubes. All blind tube nozzles (6) are alike and all through tube nozzles (4) are alike. The blind tube nozzles penetrate the reactor vessel at an angle of approximately 16° with the normal, with the axis tangent to a circle (center at vessel center) 26 in. in diameter. The through tube nozzles enter the vessel at an angle of about 24° with the normal, with the axis passing within 19.5 in. of the core center. The planes of the axes of the beam tube nozzles are

* Tentative, pending final design of the poison injection system.

TABLE IV-D-1

REACTOR VESSEL DESIGN PARAMETERS

Pressure	
Operating, psig	650
Design, psig	1000
Temperature	
Operating, °F	
Vessel Inlet	120
Vessel Outlet	156
Design, °F	250
Dimensions	
Inside diameters	
Main cylindrical section	8 ft
Lower cylindrical extension	47 in.
Length of main cylindrical section (shell) (approx.)	15 ft - 2 in.
Length of lower cylindrical section (shell) (approx.)	7 ft
Thickness of top head (approx.)	19-7/8 in.
Thickness of bottom head (approx.)	12-7/16 in.
Length of hemispherical transition (approx.)	3 ft - 2 in.
Wall thickness in main cylindrical section (approx.)	4-7/8 in.
Nozzles (shell)	
Inlet i.d., in. (1 total)	22
Outlet i.d., in. (1 total)	22
Beam tube i.d., in. (10 total)	8-1/2
Poison injection, nominal pipe size, in. (2 total)*	2-1/2
Top head penetrations, i.d., in.	
Vertical facilities (10 total)	2-1/4
Vertical facilities (5 total)	2-1/2
Instrument (3 total)	5-1/4
Vessel instrumentation, viewing ports, spares (5 total)	5-3/4
Quick-opening closure (1 total)	30 (Clear)
ITC (1 total) (approx.)	5-1/2
Instrument (4 total)	1-1/2
Bottom head penetrations, i.d., in.	
Control rods (5 total)	1.938
Gas facilities (2 total)	2-1/4
Nitrogen-16 probes (4 total)	1-1/8
Material (shell and closure heads)	Type 304L Stainless Steel

* Tentative, pending final design of the poison injection system

located 4.5 in. from the core mid-plane, with one above and one below the mid-plane; each plane contains three blind tubes and one through tube. All beam tube nozzles have an 8.5-in. i.d. and are designed for a "Grayloc" seal using a clamp-type fastening (as manufactured by the Gray Tool Company). Operation of the "Grayloc" seal will be performed remotely from an overhead position through approximately 30 ft of water. The design includes a captive device to hold the "Grayloc" assembly on the vessel nozzle and a single drive-screw to open and close the clamp remotely for connecting or disconnecting the beam tube adapter.

The flat head top closure is of the bolted and double-gasketed closure type. Two gaskets with an inter-gasket leak-off groove for monitoring leakage are provided. Metallic O-rings will be used for the gasket material.

The top head closure design will provide a means for retaining the studs in the vessel flange and to prevent rotation of the studs when the tensioning nuts are being stressed or removed. Bolt stud locating guides will be provided for installing the top head in place. Thread protectors will be provided to shield stud bolt threads during reflanging operations, and spherical washers will be employed to minimize stud bending. Bolt studs will be mechanically replaceable, if necessary.

The top head, shown in Figures IV-16 and IV-17, has a 30-in. clear central hole to accommodate the quick-opening closure plug which is removed during refueling operations. Removal of the top head will provide unimpeded access to the full 8-ft internal diameter of the reactor vessel to service or replace reactor internals other than the core. For example, the top head must be removed to replace the permanent beryllium reflector and its shroud, a fairly infrequent operation.

Penetrations for vertical facility experiments and instrumentation are located in the top head as shown in Figure IV-17. Three instrument penetrations for wide-range logarithmic nuclear channels and fifteen vertical facility penetrations are provided. In addition, five penetrations are provided for vessel and internals instrumentation, viewing ports and as spare holes.

Primarily, the quick-opening closure provides access for the replacement of the reactor core. The quick-opening closure consists of a plug retained in place against the pressure by a breech lock. Sealing of the plug is obtained by two O-rings producing a piston-type seal. The design of the closure mechanism will permit operation with long-handled tools manipulated by personnel standing on the personnel bridge platform located approximately 20 ft above the top head. The mechanism which operates the seal and rotates the breech plug will be designed to effect release or engagement within five minutes. All of the breech plug assembly parts are of a larger diameter than the vessel head opening, thereby preventing their loss to the interior of the reactor vessel. The quick-opening closure provides a clear circular opening of 30 in. diameter. The removal plug of the quick-opening closure assembly will be keyed to the vessel top head to prevent rotation of the plug. An Internal Thermal Column (ITC) circular opening and four 1-1/2-in. penetrations for in-core instrumentation will be provided in the quick-opening closure plug.

The O-ring seal for both the quick-opening closure and ITC plug closure will be an elastomer; tentatively, the material selected is Buna-Synthetic rubber. It has been estimated that the gamma radiation dose rate at the seals (approximately 12 feet above the active core) will be about 8×10^5 mr/hr at 100 MW reactor operation. Under these conditions the estimated exposure over a core lifetime of 22.5 days is about 4×10^5 R, which is well below the expected damage-producing exposure of 10^9 R for Buna-Synthetic rubber.

The bottom head, also shown in Figures IV-16 and IV-17, is a single-piece flat head forging bolted to the bottom cylindrical extension. Removal of the bottom head provides unimpeded access to the full diameter of the bottom vessel extension piece. Penetrations for five control rods, two gas facilities, and four N^{16} probes are provided in the bottom head. All control rod penetrations are 1.938 in. in diameter except for the counterbores which are 2.375 and 3.500 in. in diameter. The two gas facility penetrations are 2-1/4 in. in diameter and are in line with their mating penetrations in the beryllium reflector. The four N^{16} probe penetrations are 1-1/8 in. in diameter.

The vessel bottom closure is of the bolted- and double-gasketed type. Two gaskets with an inter-gasket leak-off groove for monitoring leakage are provided. Gaskets will be of the same material and construction as those used for the top closure. The design provides a means for retaining the studs in the extension piece flange and preventing rotation of the studs during bolt tensioning or releasing. Bolt studs are mechanically replaceable, if necessary.

3. Preliminary Stress Analysis*

a. Vessel Shell

A computer program, SEAL-SHELL-2⁵ is now available for the stress analysis of a shell of revolution with axisymmetric pressures, temperatures, and distributed loads, and was used to determine the stresses in the shell. The program is based on the finite-element method of calculation, in which a shell is approximated by a series of straight or curved segments. Each segment may have different elastic properties and pressures, and may vary linearly in thickness. Each segment may also have a temperature distribution varying arbitrarily throughout the thickness, but varying linearly along the segment. Since the segments are assumed to be interconnected at the nodal points, the essential elastic characteristics of a segment are represented by the relationship between forces applied to the nodal points and the deflections resulting therefrom. In the SEAL-SHELL-2 program, the force deflection relationship is obtained in the following manner:

- (1) For each segment a displacement field is assumed and the strain energy is determined.

* Statements and conclusions in this section are based on analyses performed for the solid Inconel vessel which was described in the original draft version of the PSAR. Similar analyses have not been performed in such detail for the solid stainless steel vessel.

- (2) By application of the principle of virtual work to the strain energy of each segment, the forces on each segment are determined in terms of deflections. Once the force-deflection relationship is known for each segment, the forces at the nodal points can be computed.

It should be noted that the approximation in the finite-element method is of a physical nature, i.e., a shell is substituted by a series of straight or curved segments. Also, the finite-element solution will converge to the true solution if the segment size is successively reduced.

The AARR vessel shell was approximated by 200 segments with an average length of approximately 2 inches. The stiffness matrix was calculated from the strain energy through the principle of virtual work. Mechanical and thermal stresses based on the theory of thick shells were calculated at the inner-, mid- and outer-surfaces of the vessel wall. Because of the numerous nonradial nozzle connections and neutron irradiation, lower stress intensity limits than are permitted by the ASME Code, Section III, "Nuclear Vessels" were adopted for the design of the reactor vessel.

- (1) The primary general membrane stress produced by mechanical loads was limited to S_m .
- (2) The primary local membrane stress produced by mechanical loads was also limited to S_m .
- (3) The combined primary membrane, primary bending, and secondary bending produced by mechanical loads, discontinuities, or thermal expansion was limited to 90% of the yield strength of the material.
- (4) The peak stress intensity was limited to S_a .

Figure IV-18 shows the primary, secondary, and thermal stresses throughout the vessel obtained in the preliminary analysis. (Note that the stresses shown for the top head are for a domed head and not for the type of head shown in Figures IV-16 and IV-17.)

b. Beam Tube Nozzle Penetrations

Special computations were included in the preliminary design to determine the maximum number and optimum placement of the beam tube nozzles. The multiplicity of non-radial beam tube nozzles in the cylindrical shell introduces stress concentrations around the penetrations. Because of the proximity of adjacent nozzles around the vessel circumference, the time-honored area replacement method of the ASME vessel codes was not expected to yield reliable evaluation of the stress state. Junction stresses were computed by Eringen's method,⁶ which assumes that the beam tube nozzles are membrane closures. The values of the stress concentration factor (S_c) around a typical penetration are tabulated below:

Angle between radial line and long. axis of shell	S_c at upper surface	S_c at middle surface	S_c at lower surface
$\phi = 0$	3.308	3.286	3.263
$\phi = 22.5$	3.068	2.892	2.716
$\phi = 45$	2.469	1.904	1.340
$\phi = 67.5$	1.842	0.863	-0.117
$\phi = 90$	1.574	0.415	-0.745

The computed values of the stresses are in good agreement with test results obtained by Taylor and Lind at the University of Illinois.⁷

c. Top Head

The flat head top closure contains the following openings: (1) a 30 in. clear central hole for the quick-opening closure, (2) fifteen penetrations for vertical irradiation facilities, (3) three openings for instrumentation, and (4) five openings for miscellaneous uses. Several penetration arrangements were analyzed to yield the optimum arrangement for experimental needs within the limitations imposed by stresses. In the stress analyses, the top head was treated as a perforated plate and effective elastic constants were used in the computation of the equivalent stresses and deflections. The peak stress intensity due to any loading was limited to S_a to assure against fatigue failures in the top head. To prevent gross plastic deformation, the average stress intensity across the minimum ligament section through the thickness of the plate was limited to S_m for the mechanical loads; the average stress intensity across the minimum ligament width (transverse to the thickness) of the plate was limited to $1.5 S_m$ for the mechanical loads. For the combined mechanical and thermal loads, the stress intensity across the minimum ligament was limited to $3 S_m$.

d. Bottom Head

The bottom head, which is a single piece flat head forging bolted to the bottom cylindrical extension, has penetrations to provide for five control rods, two gas facilities, and four N^{16} probes. Again numerous configurations were analyzed to arrive at an optimum design at acceptable stress levels and deflections.

e. Coolant Nozzle Loads

The Bijlaard theory⁸ was used to determine local stresses in the vessel wall at the two 24-in. nominal diameter inlet and outlet nozzle-to-shell junctions. External forces and moments were first resolved in the three principal directions and then the vessel wall membrane, bending, and shear stresses were calculated at the nozzle intersection. Because of the limitation of applicability of Bijlaard theory to relatively large nozzles,

the calculated stresses were compared with the available existing experimental results. Wherever a discrepancy existed between the theoretical and measured values, the larger value was used in the design as the local stress due to external loading. For a particular loading of 31,000 ft-lb on the inlet nozzle a maximum additional shell stress was found. A uniform increase in top shell course plate thickness accommodates these piping stresses.

4. Radiation Damage and Material Surveillance

Although the possibility of significant radiation damage to the vessel appears remote because of the use of Type 304L stainless steel for the vessel material, a surveillance program will nevertheless be instituted. The program will consist of an accelerated irradiation of specimens, at approximately one order of magnitude greater than expected at the vessel wall adjacent to the neutron window. Irradiation surveillance samples will be prepared from various locations in the vessel: (1) from plugs removed from the vessel for the beam tube nozzles; (2) from cutouts for the large inlet and outlet nozzles; (3) from various prolongation samples from vessel forgings; and (4) from vessel shell test plates used in conjunction with welding procedure qualification.

E. Experiment Facilities

1. Horizontal Beam Tubes

Two types of horizontal beam tube facilities are provided in AARR; blind tubes and through tubes. Six blind tubes, each of which requires only one reactor vessel penetration, terminate within the permanent reflector; all blind tubes enter the reactor vessel at an angle of approximately 16° with the normal and with their axes tangent to a circle approximately 26 in. in diameter. The two through tubes each require two vessel penetrations, and pass through the permanent reflector; through tubes enter the reactor vessel at an angle of approximately 24° with the normal, and their axes pass within 19.5 in. of the core center at the closest point. The beam tubes are shown in Figures IV-19 and IV-20. A plan view of the beam tube orientation is shown in Figure IV-4. Three blind tubes and one through tube are provided on each of two horizontal planes. The planes of the axes of the beam tubes are located 4.5 in. above and below the core mid-plane.

The beam tubes penetrate the reactor biological shield, Reactor Pool, reactor pressure vessel and reflector shroud, and either terminate in the permanent beryllium reflector or pass through it. They do not interfere with disassembly of the removable or semi-permanent beryllium reflectors. They must, however, all be removed to replace the permanent reflector.

The penetrations in the biological shield are stepped to reduce radiation streaming to tolerable levels, and to provide cone-shaped high intensity neutron beams approximately 13 in. in diameter at the shield face. The width of the beam is sufficient to permit performing two experiments simultaneously by splitting the beam.

Since each beam tube constitutes an inward protrusion of the vessel wall, the tubes must be capable of withstanding the reactor system pressure, temperature, and neutronic environment. Conditions within the reactor vessel require that the beam tubes be designed to withstand: (1) a collapsing pressure of 875 psig, (2) the thermal stresses created by gamma and neutron heating, and (3) the irradiation damage from the sustained exposure to the neutron flux. In order to ensure a high transmission of neutrons, the tube material must also possess a low total cross section for thermal neutrons. To fulfill these requirements, aluminum 6061-T6 was chosen as the most suitable material combining both structural strength and acceptable neutronic characteristics. The advantages of aluminum are: (1) it has a high thermal conductivity resulting in small temperature gradients across the beam tube wall and consequently, low thermal stresses, (2) it will not undergo severe changes in physical properties over a reasonable beam tube life because of irradiation damage, and (3) the total neutron cross-section is acceptable.

FIG. IV-19
BLIND BEAM TUBE ASSEMBLY

FIG. IV-20
THROUGH BEAM TUBE ASSEMBLY

The blind beam tube assembly and one end of the through tube assembly consist of five major subassemblies: (1) aluminum blind beam tube or through tube, (2) stainless steel beam tube adapter, (3) stainless steel biological shield alignment liner, (4) bolt-breech connectors, and (5) pool (bellows) seal. The assembly and subassemblies perform the following major functions within the facility: (1) provide for alignment of the reactor vessel nozzle with the beam tube port in the biological shield: (2) seal and confine the pool water by means of a bellows that allows for the relative movement between the reactor vessel and the biological shield: (3) make up the high pressure seal needed between the beam tube and the reactor vessel nozzle: and (4) provide the experimenter with an opening through which to collimate neutrons to suit his experiment.

The blind beam tube consists of an aluminum pressure tube encased in a thin-walled aluminum jacket. Both jacket and tube are inserted into an aluminum liner which lines the opening in the reflector. The liner is sealed to the reflector shroud. Primary system water within the reactor vessel is used for cooling the beam tubes. The water enters through the openings in the aluminum jacket at the beam tube tip and flows through the annulus formed by the tube and jacket toward the beam tube seal end and the flow reverses, enters the annulus formed by the jacket and vessel nozzle inside diameter, and discharges through the nozzle cooling outlet line. Coolant flow through the through tube is basically the same except that the coolant enters slots in the aluminum jacket at the opposite end of the tube.

Each beam tube nozzle coolant outlet line, equipped with an orifice sized for proper flow is connected to a manifold below the reactor vessel. For one end of the through-tube nozzle, the coolant outlet will be closed with a blank orifice plate.

A bolt-breech mechanism is used to connect the beam tube to the beam tube adapter. The bolt-breech connector mechanism is employed to facilitate remote assembly or disassembly of the beam tube for repair or replacement. Removal of the bolt-breech is accomplished by loosening the bolts until the breech can be rotated sufficiently to be removed. Thus the bolt and internal threads which are most likely to be damaged are part of the removable assembly and can easily be replaced. A summary of some of the beam tube design parameters is given in Table IV-E-1.

A high pressure seal at the reactor nozzle is formed between the beam tube adapter and the beam tube. The beam tube adapter, in turn, is sealed to the vessel nozzle. The purpose of the stainless steel beam tube adapter is to serve as a transition between the reactor nozzle and the beam tube. It extends from the vessel nozzle to the experimental face of the biological shield and forms a support housing for the internal shielding and the collimators. The collimators and shield plugs are gasketed and bolted to the beam tube adapter rear flange. The beam tube adapter, collimator, or shield plug form the secondary seal and are designed to take full reactor pressure should the beam tube rupture. The adapter, after initial installation, is semipermanent, and removal is not intended unless damage to sealing surfaces between it and the beam tubes occurs. Connection of the beam tube to the adapter rather than to the nozzle is done to minimize the possibility of damage to the vessel nozzle during beam tube replacement.

TABLE IV-E-1BEAM TUBE DESIGN PARAMETERS

Operating external pressure, psi	650
Design external pressure, psi	875
Operating temperature of coolant, °F	120
Design temperature (max. metal), °F	300
Materials	
Beam tube and front jacket	aluminum 6061-T6
All other components	stainless steel
Dimensions, in.	
Beam tube	
Inside diameter	5
Cylinder wall thickness	0.450
Hemisphere wall thickness at tip (blind tube only)	0.225
Front Jacket	
Inside diameter	6.15
Wall thickness	0.0625
Annulus (between tube and jacket)	0.125

A damaged seat on the beam tube adapter is easily remedied by replacement of the adapter.

The design of the opposite end of the through beam tube assembly is modified from that described above in order to provide a bolt-breech connector for securing and sealing the end of the through beam tube which has been inserted through the vessel. The bolt-breech connector features a high pressure bellows seal which also allows for axial thermal expansion of the through beam tube. To prevent catastrophic leakage in the event of a rupture in the high pressure bellows, the gasketed and bolted closure at the rear of the adapter serves as a back up high-pressure seal.

Except for connection between the reactor nozzle and beam tube adapter, all assembly and disassembly operations are performed through the beam port in the biological shield. The connections at the reactor vessel beam tube nozzles are accessible vertically through approximately 25 ft of water for remote fastening or disconnecting. The clamping device used for the connection is equipped with a single drive screw for tightening or loosening of the clamp. When the clamp is opened, it allows the hubs (reactor vessel nozzle and the beam tube adapter) to be separated or disconnected. The clamp is designed to remain captive on the reactor vessel nozzle. The drive screw, which has a hexagonal head and is accessible in the Reactor Pool from a position above, is operated from above by a long-handled tool. The seal gasket is fastened in place on the beam tube adapter hub by a screw before installation of the beam tube adapter.

The entire beam tube facility inherently lends itself to easy piloting of assembly tools or components because of its stepped circular design. All connections not easily accessible are designed on the breech-lock principle to facilitate removal by remotely-operated tools. The beam tube has been especially designed to be piloted into position during its final phase of assembly.

Any work requiring access to the area between the biological shield face and core can only be performed when the reactor is shut down and the core removed. This work includes: (1) replacement of collimators, (2) plugging of beam tubes, and (3) removal or replacement of shielding within the biological shield liner.

Insertion or removal of beam tubes is performed under "dry" conditions. That is, the reactor is shut down, the core is removed and the vessel is drained below the level of the beam tubes prior to their removal. Because of the void within the beam tube, the reactor water cannot contribute significantly to shielding during tube removal. Therefore, removal of the reactor water should not increase the radiation dose rate at the exit of the beam tube. However, if needed to ensure safe radiation levels, a plug can be inserted into the beam tube prior to replacement operations. External shielding at the beam tube port can also be furnished as necessary.

Beam tubes are disconnected by remotely-operated tools and guided into a shielded cask. The cask is sized to hold one blind tube, which is approximately 6-1/2 ft long. Removal of a through tube is accomplished by cutting the tube (approximately 16-1/2 ft long) into three equal sections

about 5-1/2 ft long during insertion into the cask. The cask is equipped with a cutting device, as well as an elevating device for positioning at the two beam tube levels. Cask cooling is not required.

At present, two types of experiments are anticipated. The neutron measurement and diffraction experiments will require that various types of collimating and/or reflecting or absorbing devices be inserted in the beam tube. The other type will involve sample irradiation experiments that will require insertion of a support device and a transfer mechanism into the through beam tube. It is expected that the collimator seal will take the form of an aluminum diaphragm at the reactor end of the collimator and a flange seal at the outer end. The sample transfer mechanism adapter will likewise have a small diameter seal around the transfer actuator rod plus the large diameter flange seal, both at the outer end. Shield plugs will require only the large diameter flange seal at the outer end.

Shield plugs will be inserted in all beam tubes not being used for particular experiments. Shielding so provided will be equivalent to the biological shielding and require appropriate cooling.

Taps are provided for monitoring beam tube pressure, drawing a vacuum, or introducing helium. Lines from these taps plus those from the collimators and shield plugs are brought out to a rectangular recess in the face of the biological shield surrounding the beam port, where connections to necessary service lines can be made.

Service lines provide for helium, water, vacuum and off-gas at the beam tube experiment facilities and are located on the Reactor Pool wall above the biological shield instrument balcony as shown in Figure IV-21. Lines connected to the beam tube experiment facilities extend upward along the cavity face and through penetrations in the top of the shield cavity into the biological shield instrument balcony floor trench and to a panel mounted over the beam tube cavity on the pool wall and finally to the stubs of the main service lines on the pool wall. The valves, flowmeters, temperature indicator gauges, and necessary electrical gear are mounted to the panel. Appropriate drain lines are routed to the hot drain at the bottom of the shield cavity.

Coolant flow to each of the beam tube jackets and nozzles, and to each of the collimators is monitored, and an alarm is initiated in the event of loss-of-coolant-flow. Pressure within the beam tubes is also monitored and an unusual increase in pressure sounds an alarm. The need to scram the reactor because of loss-of-coolant-flow or an abnormally high pressure in a beam tube will be established during Title-II design.

Under normal operating conditions a flow rate of 50 gpm at 120°F is required for cooling each of the four blind beam tubes closest to the core. When appropriate safety allowances for heating rates, flow rates, and film coefficients are considered, calculations indicate that the maximum beam tube temperature will not exceed 250°F. This result also includes allowance for the buildup of 3 mils of corrosion product on the tube surface, a thickness which would probably require over 1-1/2 years to develop and which is in excess of the 2 mil level at which spallation is probable⁹. At 50 gpm

FIG. IV-21
SERVICE FACILITIES FOR AARR
HORIZONTAL BEAM TUBE EXPERIMENTERS

the temperature in the through tubes and the remaining two blind tubes will be considerably lower. However, preliminary calculations indicate that in the case of loss-of-coolant-flow the tube metal will not exceed 350°F for at least ten minutes after scram from full power.

Until recently there has been very little quantitative information on long-term ($>10^{22}$ nvt) neutron irradiation effects on the beam tube material (6061-T6 aluminum). Some information was obtained from the ETR in August 1963. One of the ETR control rod guide tubes, which was fabricated from type 6061-T6 aluminum, had been exposed to an integrated fast neutron (>1 MeV) dose of 2.6×10^{22} nvt. Twelve hardness measurements were made and ranged from R_B 69.5 to R_B 75.5 with an average hardness of R_B 72. This data indicated that there was no significant change in the hardness of the material. Short term irradiation results were also very encouraging. Investigations had been conducted that showed that aluminum alloy 6061 in the fully-annealed, cold-worked, and age-hardened conditions undergoes very small changes in the tensile properties as a result of neutron irradiation to doses on the order of 10^{19} fast (>2.9 MeV) nvt and 10^{20} thermal nvt.¹⁰ At irradiation temperatures above 115°C (239°F), no changes in properties were observed. Small changes observed when the material was irradiated at 43°C (109.4°F) were mostly annealed out after 1/2 hour at 200°C (392°F). In general, beam tube temperatures in the high neutron flux will range from at least 150°F at the outside surface to 250°F on the inside surface.

The most recent investigations by ORNL¹¹ and by Idaho Nuclear Corporation¹² have shown that at doses above 5×10^{21} nvt (>0.8 MeV) loss of ductility in aluminum and several aluminum alloys (A1-1100, X-8001, and 6061) became apparent. Unfortunately most of the current information is fragmentary and thus no complete behavior pattern is known, especially as it pertains to the effect of irradiation temperature on net loss in ductility. However, it is anticipated that by the time AARR starts operation, the relationship between changes in physical properties with irradiation at temperature should be reasonably well understood.

The AARR surveillance program will include periodic removal of a beam tube for destructive testing. Tubes will be removed according to schedule (to be established at a later date) and physical properties of the beam tube tip will be obtained. Plans have been made in the HFIR facility to expose surveillance specimens to a higher flux than the beam tubes and to evaluate the results periodically to determine if any significant changes in mechanical properties have occurred. Further, it is expected that at least one HFIR beam tube will have been removed for examination and possibly destructively tested before AARR starts operating. These results will be available to the AARR.

The portion of the tube nearest the core will be exposed to neutron fluxes and doses shown in the table below:

Energy Group	AARR Beam Tube Exposure	
	Flux (n/cm ² -sec)	10-yr Exposure (nvt)
>0.9 MeV	8.5×10^{13}	2.8×10^{22}
>0.1 MeV	2.35×10^{14}	7.4×10^{22}

2. Internal Thermal Column Irradiation Facility

The Internal Thermal Column (ITC), located centrally in the core, is a region of high neutron flux made available for irradiation of samples. The ITC facility is a device contained within the Internal Thermal Column for the purpose of irradiating encapsulated samples. The facility, which penetrates the top of the reactor vessel, provides for various durations of sample irradiation ranging from a few milliseconds to several core cycles. Long irradiations are performed in static irradiation facilities (basket-type) which are inaccessible during reactor operation. Short term irradiations of a few milliseconds are performed in the fast gas rabbit, and short to long (seconds to core life) irradiations are performed in the hydraulic rabbits and in the gas-cooled rabbit.

The ITC facility is a two-piece assembly consisting of the ITC reactor core assembly (Figure IV-22) and the ITC top assembly (Figure IV-23). The ITC reactor core assembly, sometimes referred to elsewhere in this report as the ITC insert, extends downward about 13 ft from the top of the reactor vessel and terminates just below the core. The tubular assembly housing, 5 in. in outside diameter, is supported vertically by the quick-opening closure plug and laterally by guides above and below the core. The ITC reactor core assembly, which contains all the irradiation facilities, is installed and removed as a one-piece unit during refueling operations. The ITC top assembly is connected to the ITC reactor core assembly at the quick-opening closure plug, and is fastened to the plug by a breech lock. Sealing is obtained by synthetic rubber O-rings. Connection of the various hydraulic and gas rabbit lines is made to the ITC top assembly.

Initial irradiation facilities installed in the ITC facility are as follows: (1) five-1/2 in. hydraulic rabbits, (2) one gas (helium)-cooled rabbit, (3) one gas fast rabbit (helium operated), and (4) two static irradiation facilities. All rabbit samples enter and exit the reactor at the quick-opening closure plug and, with the appropriate equipment provided, can be routed to various other areas in the plant, as discussed in Section 4 below.

A summary listing of the various ITC facilities follows:

(1) Hydraulic Rabbits

No. of facilities	5
Sample Size	1/2 in. dia x 3/4 to 3 in. long
Quantity of samples per irradiation	21 in. accrued length (max)
Sample transfer interval (reactor to load-unload station)	30 sec

FIG. IV-22
ITC REACTOR CORE ASSEMBLY

FIG. IV-23
ITC TOP ASSEMBLY

(2) Gas Rabbit (Fast)

No. of facilities	1
Sample Size	1/2 in. dia x 3/4 in. long
No. of samples per irradiation	1
Sample exposure time	~1 msec minimum
Sample transfer interval (reactor to load-unload station)	50 msec - ~1 sec

(3) Gas-Cooled Rabbit

No. of facilities	1
Sample Size	0.1 in. dia x 3/4 in. long
No. of samples per irradiation	1
Sample exposure time	1 sec minimum
Sample transfer interval (reactor to load-unload station)	2 sec minimum

(4) Static Facilities

No. of facilities	2
Sample Size	1/2 in. dia x 18 in. accrued length (max)

Primary coolant flow is maintained through the ITC reactor core assembly, static (basket) facilities, and hydraulic rabbit facilities by utilizing the reactor core pressure drop. Cooling water above the core enters the flow tube of each irradiation facility at the top, passes through the annulus between the sample and the rabbit tube, and is discharged below the core. The ITC reactor core assembly housing is cooled on the outside surface by coolant passing through the annulus formed by the housing and ITC shroud. The bulk ITC water in the volume within the housing and external to the irradiation facility tubes is used to cool the facilities. The velocity through the bulk ITC volume is kept intentionally low in order to prevent fast introduction of voids in this area. Coolant enters the bulk ITC volume through ports in the ITC shroud, flows upward to the top of the ITC reactor core assembly housing, down through the housing, and exits through holes in the bottom guide plate for the irradiation tubes. Approximately 110 psi pressure differential is available over the ITC facility.

Cooling of the gas-cooled rabbit tube, gas operated fast rabbit tube, and samples therein is done with helium.

Cooling requirements of the gas systems have been determined from the heat generated by (1) the in-core facility hardware, and (2) the samples within the facilities. The cooling requirements of the ITC Gas Operated Fast Rabbit Facility are minimal for the following reasons:

- 1) Sample irradiation is of the order of 5 milliseconds.
- 2) The heat generated in the in-core facility hardware is transferred to the primary reactor water, and the facility need not rely on internal gas flow for heat removal.

- 3) The water in the pools and canals acts as a heat sink in conjunction with part of the flow system tubing.

Gas flow is needed to cool the ITC Gas-Cooled Rabbit Facility only when a sample is being irradiated. The heat generated by the in-core facility hardware is transferred to the primary reactor water. In the event of a failure in the flow system, apparent through the reduction of either coolant flow or pressure, a standby pressurized tank will be exhausted to remove the sample.

The gas pumps used in both flow systems have been sized to provide the required sample delivery times after irradiation, thereby greatly exceeding the facility cooling requirements.

Reliability of the gas pumps and other flow system components is the subject of research and development studies.

Further information on the operation of rabbit systems will be found in Section 4 below.

3. Reflector Irradiation Facilities

The permanent reflector is penetrated by 18 vertical holes, which extend completely through the beryllium. Each penetration is equipped with a locking mechanism and an aluminum liner. In addition 16 smaller but unlined penetrations are provided in the removable and semi-permanent reflectors. The orientation and location of these 34 reflector irradiation positions are shown in Figure IV-24. Penetration diameters, distances from core center, and contemplated uses are listed in Table IV-E-2.

Access to the reflector penetrations is generally made through the upper vessel head 30-in. opening. In addition, fifteen smaller penetrations in the upper vessel head and two in the lower vessel head are provided for use in conjunction with the reflector penetrations. The sizes and locations of these penetrations are shown in Figure IV-17.

The types of irradiation facilities contemplated for initial installation into the reflector penetrations are static irradiation facilities, hydraulic rabbit facilities, and gas rabbit facilities. Those penetrations not occupied by a specific type of irradiation facility will be filled with a beryllium insert. A summary listing of the various reflector facilities follows:

(1) Hydraulic Rabbits

(a) 1/2 in.

No. of facilities	2
Sample Size	1/2 in. dia x 3/4 to 3 in. long
Quantity of samples per irradiation	21 in. accrued length (max)

TABLE IV-E-2REFLECTOR PENETRATIONS

<u>PENETRATION</u>	<u>DIA. (in.)</u>	<u>RADIAL LOCATION FROM CORE CENTER (in.)</u>	<u>FACILITY USE</u>
V2-1 thru V2-8	1/2	12-7/8	Be Filler
V2-9 thru V2-16	1/2	11-3/8	Be Filler
V3-1	1-1/2	15-1/2	1/2" Hydraulic Rabbit
V3-2	1-1/2	15-1/2	1" Static Basket
V3-3	1-1/2	15-1/2	Be Filler
V3-4	1-1/2	15-1/2	Be Filler
V3-5	1-1/2	15-1/2	1/2" Hydraulic Rabbit
V3-6	1-1/2	15-1/2	Be Filler
V3-7	1-1/2	15-1/2	Be Filler
V4-1	3	19-1/2	1" Gas Rabbit
V4-2	1-1/2	19-1/2	1" Hydraulic Rabbit
V4-3	2	19-1/2	1-1/2" Static Rabbit
V4-4	1-1/2	19-1/2	1" Hydraulic Rabbit
V4-5	3	19-1/2	1/2" Gas Rabbit
V5-1	2-1/2	23-1/2	Be Filler
V5-2	2-1/2	23-1/2	Be Filler
V5-3	2-1/2	23-1/2	Be Filler
V5-4	2-1/2	23-1/2	Be Filler
V5-5	2-1/2	23-1/2	Be Filler
V5-6	2-1/2	23-1/2	Be Filler

Sample transfer interval (reactor to load-unload station)	30 sec
--	--------

(b) 1 in.

No. of facilities	2
Sample Size	1 in. dia x 3 in. long
Quantity of samples per irradiation	21 in. accrued length (max)
Sample transfer interval (reactor to load-unload station)	30 sec

(2) Gas-Cooled Rabbits

No. of facilities	2
Sample Size	1/2 in. dia x 3 in. long
No. of samples per irradiation	1
Sample exposure time	1 sec minimum
Sample transfer interval (reactor to load-unload station)	2 sec minimum

(Note that sample motion can only be achieved in one of these facilities at a time)

(3) Static Facilities

No. of facilities	2
Sample Size	1 - 1 in. dia x 18 in. accrued length (max) 1 - 1-1/2 in. dia x 18 in. accrued length (max)

Primary coolant is used to cool the static irradiation facilities in the reflector. Proper coolant flow is obtained by orificing the flow at the entrance and exit of the facility. The pressure drop over the core is available to drive the primary coolant through the static (basket) facilities, and through the annular space between the rabbit facilities and the penetration wall.

Figure IV-25 shows two typical reflector static irradiation facilities, each consisting of a sample basket extending through the reflector, a locking mechanism securing the samples in the basket, and a hold-down device to fasten the assembly to the top structure of the reflector. Cooling water enters the top of the basket tube through the orifice, passes through the annulus between the tube and the sample, and discharges from the bottom orifice to the region below the core. In the event that a facility is not used for irradiation, a beryllium filler insert is substituted for the sample basket. A hold-down device and locking mechanism are also provided for the filler insert.

FIG. IV-25
REFLECTOR STATIC IRRADIATION FACILITY

Hydraulic and gas-cooled rabbit facilities are cooled internally and operated by reactor water and by helium, respectively. Further information on these systems will be found in Section 4 below.

4. Rabbit Facilities

a. Hydraulic Facilities

The hydraulic rabbit facility provides a means for transporting samples from a load-unload station to either the reflector or Internal Thermal Column, water cooling the samples during irradiation, and returning them to the load-unload station. All hydraulic rabbit tubes enter the reactor through the top head of the vessel.

Figure IV-26 shows a typical reflector hydraulic rabbit assembly, and Figure IV-27 shows a typical hydraulic rabbit flow system. The system consists of three major areas: (1) the load-unload station, (2) the transfer station (located between valves BV20A-7 and BV20A-8), and (3) the reactor (ITC or reflector). Load-unload stations for all hydraulic rabbit facilities (ITC and reflector) are located in the Active Material Handling Building Rabbit Pool. All transfer stations are located in the Reactor Pool. The portion of the system between the load-unload station and the transfer station is a low pressure circuit independent of the reactor, whereas the portion of the system between the transfer station and the reactor is a part of the high pressure reactor primary coolant system.

Operation of the system is as follows:

(1) Sample Loading

Samples to be irradiated are charged into the load-unload station (one station is provided for each hydraulic facility), and by a proper setting of valves are transported to the transfer station by the low pressure system. The low pressure isolation valves at the transfer station are then closed, and high pressure transit valves are opened, allowing primary system water supplied by the primary coolant system pumps at about 650 psig pressure to force the samples into the reactor (ITC or reflector). The samples are positioned in the reactor by a stop at the bottom of the rabbit tube. The water used for sample transport discharges to the core outlet plenum below the reactor (about 540 psig pressure).

(2) Irradiation Period

Immediately after the samples have been inserted, the high pressure reversing valve is closed, and the coolant control valve immediately above the reactor top head (see Figure IV-27) is opened to permit water from the inlet plenum in the top of the reactor vessel (650 psig pressure) to pass through the rabbit tube and cool the samples. The water is discharged to the core outlet plenum below the reactor.

(3) Sample Unloading

When sample unloading is desired, the coolant control valve is closed, the high pressure flow reversing valve is opened, and the transfer station is vented to the primary system degasifier tank. The pressure differential between the core outlet plenum (about 540 psig) and the degasifier tank (about 1.5 psia) is used to reverse the coolant flow in the rabbit tube and transport the samples back to the transfer station. The valving is then changed so that the low pressure system transports the samples from the transfer station to the load-unload station.

b. Gas-Cooled Facilities

A gas (helium)-cooled rabbit facility provides a means for transporting a single sample from a load-unload station located in a Laboratory and Office Building radiochemical laboratory directly into a rabbit tube in the ITC facility or in the reflector, cooling the sample with gas during irradiation, and rapidly returning the sample to the load-unload station in the radiochemical laboratory. The facility provides sample exposure times from about one second upward, and a post-irradiation transit time of 4-5 seconds from the reactor to the load-unload station. The ITC gas-cooled rabbit tubes are a part of the ITC facility which enters the reactor through the top head of the vessel.

Figure IV-28 shows a flow diagram of a typical gas-cooled rabbit facility in the reflector. Figure IV-29 is a flow diagram of the ITC system. Referring to either flow diagram, when a sample is to be irradiated, it is placed in the load-unload station, the valves listed in the insertion mode are opened, and the sample is transported by high pressure helium into the reactor (ITC or reflector) to a sample stop at the reactor mid-plane. The sample is cooled by high pressure gas during irradiation. At the conclusion of irradiation the gas flow is reversed to return the sample quickly to the load-unload station.

The ITC gas-cooled rabbit system is entirely independent of the reflector system. The ITC system employs two gas pumps, both of which are used for continuous operation. The reflector systems are similar in operation.

In both the ITC and reflector systems, the operating gas pressure does not exceed 400 psia at any point. Therefore, leaks developing between the rabbit tubes and the reactor water will not cause the entry of gas into the reactor. Should a failure in the flow system of either facility occur, a pressurized tank will be used to eject the sample from the system. The heat generated by either ITC or reflector facility hardware within the irradiation zone is transferred to the reactor water. The effectiveness and reliability of this heat transfer is being determined through research and development studies.

FIG. IV-28
FLOW DIAGRAM
REFLECTOR GAS COOLED RABBIT
FACILITY

FIG. IV-29
ITC GAS COOLED RABBIT
FACILITY-FLOW DIAGRAM

c. Fast Gas Operated Facility

The fast gas (helium) rabbit facility provides a means for transporting a single sample from a load station located either in a Laboratory and Office Building radiochemical laboratory or on the Experiment Floor, directly to and through the rabbit tube in the ITC facility, to an unload station located either in a Laboratory and Office Building radiochemical laboratory or on the Experiment Floor. Figure IV-30 is a flow diagram for the system. When a sample is to be irradiated it is placed in a load station, the valves listed in the appropriate irradiation mode are operated in sequence, and the sample is propelled by high pressure helium from the gas pump toward the reactor core. When the sample has passed the pressurized holdup tank, flow from the holdup tank commences while that portion of the flow system containing the gas pump is valved off, allowing the gas pump to continue operating through the flow system bypass. At the same time, the sample has been propelled to and through the rabbit tube in the ITC facility and on to an unload station at the desired destination.

The operating gas pressure in the system does not exceed 400 psia at any point. Therefore, a leak developing between the rabbit tube and the reactor water will not cause entry of gas into the reactor.

FIG. IV-30
 ITC GAS OPERATED FAST RABBIT
 FACILITY-FLOW DIAGRAM

F. Radiation Shielding

1. Introduction

To the extent feasible, the intense sources of biologically harmful radiation in the AARR are located within the Reactor Containment Building (RCB). The RCB houses: the reactor pressure vessel; the entire primary coolant system, including piping, pumps, heat exchangers, and water-purification equipment; the off-gas system; the Reactor Pool, in which the reactor pressure vessel is submerged; storage pools for spent fuel, used control rods, and other "hot" vessel internals; recirculation-system filters for air cleaning; beam-tube and through-tube experiment facilities; and portions of the rabbit systems for research irradiations.

The Active Material Handling Building contains: storage facilities for radioactive solid and liquid wastes; the Main Canal, where spent fuel is loaded into a shipping cask after cooling in the RCB for approximately 8 months; and the Rabbit Pool, through which irradiated rabbit samples must pass en route to experimenters' facilities, which are in the Laboratory and Office Building.

The normal sources of radiation in the Laboratory and Office Building are the irradiated targets in the radiochemical laboratories and in the Radiochemical Service Room.

The radiation shielding must provide adequate biological protection of personnel working within the various buildings of the AARR complex. Some of the personnel will be engaged in plant operations. Others will be concerned with operation of the various research experiment facilities in the RCB and in the Laboratory and Office Building. The shield design provides both for established shielding for well-defined operations, and for shielding tailored to the needs of certain research facilities, typically beam-tube and through-tube experiment facilities.

The radiation shielding is designed to provide biological protection against radiations: from the reactor during operation at power levels up to 100 MW; from the primary coolant system and the associated auxiliary equipment such as filters, demineralizers and from solid, liquid, and gaseous wastes; from hot fuel elements, control rods, beryllium-reflector sections, and other vessel internals, during transfer and in storage; and from experiment facilities.

Special consideration is given to design of radiation shielding to provide biological protection in the event of a malfunction which might result in increases in radiation intensity in areas where personnel must work. One example of this is the design of shielding of the load-unload station for rabbit facilities, to provide personnel protection if a malfunction resulted in a highly activated target becoming stuck at the load-unload station.

Another basis for design of the radiation shielding is the requirement for low radiation background at certain experiment facilities, to minimize interference with the measurements being made.

Finally, the design of the radiation shielding is guided by the goal of reducing the administrative problems of personnel assignment to operations in various radiation areas.

Sections below summarize the criteria for radiation dose rates established to govern the designs of the various radiation shields in the AARR complex.

2. Summary Description of the Radiation Shields

The major radiation-shielding components for conditions of normal plant operation are: the reactor biological shields (radial and axial); shields against radiations from the primary coolant system; shields against radiations from target materials and from radiation streaming from beam tubes and through tubes; and the pools and canals which provide shielding against radiations from the operating core and from activated materials in storage or in transit in these pools and canals, such as spent fuel and control rods.

The reactor vessel is submerged in a pool of water (Reactor Pool) which is an effective shield against neutron and gamma radiation. The walls and floors of the Reactor Pool and of the other pools and canals are concrete (ordinary or magnetite), with thicknesses determined by radiation-shielding requirements to meet the dose-rate criteria for personnel in adjacent zones. Where feasible to do so, the storage facilities for hot materials in the various pools are so arranged that a portion of the radiation attenuation will be provided by pool water. There will be allowance for the anticipated variations from the paths established for transfer of hot components to storage.

The radial bulk biological shield includes pool water and a 7-ft-thick zone of magnetite concrete approximately up to the elevation of the top flange of the reactor vessel. In addition, the beryllium reflector, the primary coolant in the reactor pressure vessel, a steel thermal shield inside the vessel, and the pressure vessel itself all assist in the attenuation of radiation from the operating reactor core.

Axially downward from the reactor core, the bulk shielding includes water, a steel shield plug, and the bottom head of the reactor pressure vessel (see Figure IV-3). Radiation proceeding in an oblique downward direction is attenuated by the 4 to 6 ft thick magnetite concrete floor of the Reactor Pool. Axially upward from the reactor core, the bulk shielding includes primary coolant water, the thick top head of the reactor pressure vessel, and the water in the Reactor Pool.

The RCB houses the piping runs and the primary heat exchangers and pumps, and process equipment for the primary coolant, such as demineralizers and filters. This primary equipment is located on the Service Floor, and the units are segregated into areas to which access will be prohibited or limited, as appropriate, during reactor operation. In general, the Service Floor is divided into areas of limited access and an area where access is prohibited during normal reactor operation at power. The radiation shielding on the Service Floor is provided by a combination of geometric

distance and permanent or removable shielding walls.

Personnel on the Experiment Floor of the RCB are afforded protection against radiations from the primary coolant equipment on the Service Floor below, by the Experiment Floor slab. This slab is reinforced, ordinary concrete, 6 ft thick, with special shielding where needed to compensate for local shallow trenches in that slab for experimenters' facilities.

Materials irradiated in the rabbit facilities are shielded by pool water and then by concrete and steel shields as these materials are transported to the hot facilities when they are to be handled. Special design provisions will be included to protect personnel from radiations in the event that rabbit facility equipment should malfunction and interrupt the scheduled transport of a hot target.

Demineralizers and filters will be isolated and shielded to protect personnel even in the event of an accident resulting in melting of some fuel and dispersal of fission products in the primary coolant system. The demineralizer beds for the primary coolant system and for the pools will be designed to permit remotely controlled removal of contaminated resin and its replacement by fresh resin. This flexibility is reflected in the dose-rate criteria tabulated in Section 5 below.

In the event of contamination inside the RCB, personnel outside would be afforded protection against direct gamma-ray radiation by the 2-ft-thick reinforced concrete building walls, and by the 1-ft-thick lightweight concrete dome. Personnel on the first floor of the Laboratory and Office Building are protected from direct radiation from the RCB by an additional wall with a gamma-ray-shield effectiveness equal to that of 1 ft of ordinary concrete. The Laboratory and Office Building also has a concrete roof.

3. Remarks on Radiations Determining Shield Design

Preliminary calculations have indicated that the radial and axial bulk biological reactor shield thicknesses are determined principally by the requirements of attenuating gamma-ray radiation from the reactor fuel zone and its immediate vicinity. Special problems of radiation streaming exist because a principal purpose of the AARR is to supply streaming ducts for physics experiments. These ducts are oriented so as to be tangential to the fuel zone and thus reduce the intensity of streaming of gamma rays and high energy neutrons. Nevertheless, substantial local shielding will be required to reduce radiation levels from these experiment facilities, and space is provided for such shielding which will be designed to fit the particular facility.

The principal radiation from the primary coolant consists of high-energy (6-7 MeV) gamma rays arising from decay of nitrogen-16 (N^{16}). N^{16} is produced in the water passing through zones of intense high-energy neutron flux, by the $O^{16}(n,p)N^{16}$ reaction. Conservative values of the N^{16} production rate have been specified for shield-design analyses.

4. Shielding Criteria for Normal Reactor Operations

The criteria for design of shielding for the plant and for the experiment facilities are set so that the radiation doses to personnel will

not exceed the limitations specified in 10 CFR 20 in the operating plant. In general, the shield design criteria are established to limit biological doses in areas of normal, unrestricted access to levels which would provide a margin for average annual radiation dose so that personnel also would be able to perform operations in limited-access areas without exceeding the 10 CFR 20 dose limitations.

For limited-access areas, the dose-rate criteria are determined by the anticipated exposure times of personnel. It is expected that limited access areas will be of two types: (1) plant-operation areas, for example part of the Service Floor in the RCB; and (2) areas in the immediate vicinity of certain experiment facilities, typically those utilizing beams of neutrons.

In each case, the criterion dose rate includes contributions from all sources of radiation. At dose points on the Experiment Floor, the principal normal contributors include: (1) N^{16} activity in primary heat exchangers and piping runs; (2) scattered radiation from beam-tube facilities; and (3) radiation penetrating the radial biological reactor shield. Thus, to make allowance for scattered, incompletely shielded radiations from the experiment facilities, the criterion for the total radiation dose rate from all other sources is that this total not exceed 1/4 mrem/hr during normal reactor operation at 100 MW. The reference dose-point height is 3 ft above the floor slab. Present indications are that the bulk of the contribution to the dose rate on the Experiment Floor, other than from scattered radiation from experiments, will come from the Service Floor rather than from the gamma-rays penetrating the bulk biological reactor shield.

Shielding and water depths in the pools and canals are based primarily upon radiation intensities from the operating reactor, spent fuel elements, used control rods and other vessel internals. Once the activated components are in their storage locations in the pools, there is a depth of water above them such that the dose rate contribution at the top surface of the pool water is negligible. The criterion dose rate at the top surface of pools and canals from hot material in storage is 1/4 mrem/hr at the closest normal approach of personnel. During transfer of hot material, however, the depth of the water cover is reduced. The criterion on the dose rate during transfer of spent fuel is that, for a fuel element that has been operated for (conservatively) 30 days and cooled for six hours, the dose rate at the personnel bridge is not to exceed 5 mrem/hour. More generally, the criterion dose rate is 5 mrem/hr at an elevation of 3 ft above the surfaces of pools and canals with spent fuel in transit. At the surface of the Reactor Pool the criterion dose rate is 5 mrem/hr at the closest normal personnel approach for the case of normal reactor operation at 100 MW.

At dose points on the stairwell from the Experiment Floor to the Service Floor of the RCB, and at limited-access dose points on the Service Floor, the criterion general background dose-rate is 5 mrem/hr, for normal reactor operation at 100 MW.

In the Rod Drive Access Room and in the Electronic Equipment Room in the RCB, the criterion dose rate is 25 mrem/hr during normal reactor

operation at 100 MW. The sources of radiation are primary-coolant-pipe runs near the outside surfaces of the concrete walls of these rooms. Typically, exposure times of personnel in these rooms will be short.

Irradiated targets in rabbit facilities will be shielded throughout their transfer path by water (in the Rabbit Pool) and by concrete and steel shields. Upon irradiation, these targets will be transferred to shielded radiochemical laboratories in the Laboratory and Office Building.

5. Shielding Criteria for Abnormal Conditions

The Pool and Canal Demineralizers and the Primary Coolant Demineralizers are isolated in shielded cubicles on the Service Floor of the RCB. Dose-rate criteria have been established at dose points near the outside surfaces of the shielded cubicles. The source specified for shield design for the Pool and Canal demineralizers is a total of 0.05% of the fission products in the reactor, after 30 days of operation followed by 24 hours of cooling. For the primary coolant demineralizers, the corresponding source is specified to be 0.5% of the total reactor fission products. The designs of demineralizer systems include provision for remotely controlled replacement of the resin with fresh resin by sluicing.

To protect personnel in the Laboratory and Office Building in the event of a major reactor accident which contaminated the RCB, the roof and the wall adjacent to the RCB are designed to afford additional attenuation of direct gamma radiation from the RCB. A criterion for the design of the Laboratory and Office Building is that under MCA conditions personnel in that building would not receive a whole body dose in excess of 25 rem, from direct radiation from the (assumed) uniformly contaminated RCB, in the first 15 minutes following the initiation of the MCA.

G. Heat Removal and Auxiliary Systems

1. Primary Coolant System

a. General Description

The primary coolant system provides a means for removing thermal energy produced in the reactor and transferring it to the secondary coolant system for subsequent dissipation to the environment. A simplified flow diagram showing the various heat removal and auxiliary systems in schematic form is shown in Figure IV-31. A complete primary system flow diagram, which indicates all main valves, is shown in Figure IV-32.

b. Primary Coolant Loop

The main loop consists of the reactor pressure vessel, three parallel-connected heat exchangers each sized for 33-1/3 MW heat removal capacity, two circulation pumps, each with a capacity of 9470 gpm, a full-flow strainer located immediately upstream of the reactor vessel, and connecting 24-in. piping with associated valves.

Under normal conditions, a flow of 19,140 gpm at 650 psig pressure and 120°F enters at the upper portion of the reactor vessel and flows downward through the core, beryllium reflector, and other internals. The main portion of the flow, 13,800 gpm, passes through the core at a velocity of approximately 51 feet per second, and exits through the 24-in. outlet nozzle below the core. Reactor outlet pressure and temperature are approximately 523 psig and 156°F. The coolant leaving the reactor vessel is circulated through the three main heat exchangers and, after diversion of a 200-gpm stream to the low-pressure clean-up system, enters the main circulation pumps. The combined flow from these pumps and one of the two pressurization pumps, which returns the clean-up flow, passes through the full-flow strainer into the reactor vessel. The distribution of flow through the reactor vessel and through portions of the primary loop and related sub-systems is shown in Figure IV-33. The flows shown in Figure IV-33 are approximate and have not been corrected for differences in coolant temperature in various parts of the system.

The primary heat exchangers are of the horizontal shell-and-U-tube type with a stainless steel clad carbon steel shell and nickel alloy tubes, with welded channel and bolted channel cover, constructed in accordance with Section III, Class A of the ASME Boiler and Pressure Vessel Code. The specification includes such requirements as special analyses and all necessary quality control and inspection methods and procedures. Nickel alloy has been selected as the heat exchanger tubing material because of the high concentration of chloride ions in the secondary coolant.

The design of the heat exchangers is unconventional in that they are designed for circulation of high pressure primary coolant on the shell side and low pressure secondary cooling water on the tube side. The continuous addition of airborne organic contaminants into the cooling tower basin, plus the fouling impurities contained in the treated Chicago Sanitary and Ship Canal water used for cooling tower make-up, make this arrangement desirable, as it permits maintenance of a secondary coolant flow of

FIG. IV-31
SIMPLIFIED OVERALL FLOW DIAGRAM

FIG. IV-33
PRIMARY SYSTEM FLOW DISTRIBUTION DIAGRAM

sufficient velocity to reduce significantly the extent of surface fouling. In addition, a uniform flow pattern, and the absence of pockets and hideouts permits the use of a much lower design fouling factor than would be possible with secondary coolant on the shell side (0.0015 vs. 0.0025). The net result is a reduced surface requirement with improved operational reliability.

The heat exchanger shell is designed for 250°F and 875 psig, and the tubes for an external pressure of 875 psig. Each heat exchanger has a capacity of 33-1/3 MW at a primary coolant flowrate of about 3.2×10^6 lbs/hr (approx. 6,400 gpm at 120°F) and a secondary coolant flowrate of 5.17×10^6 lbs/hr (approx. 10,330 gpm at 83°F). The primary coolant inlet temperature is approximately 156°F and the outlet 120°F. Secondary cooling water is furnished at 83°F and is raised to 105°F at the heat exchanger outlet.

The primary pumps are vertical, stainless steel, single suction, mechanical-type shaft seal, centrifugal pumps with direct-connected water-cooled motors, and d-c pony motors. Each pump is rated at 4.69×10^6 lbs/hr (9,470 gpm) at 120°F and 470 ft head. The drive motors are rated at 1250 HP. The casing design temperature is 250°F and design pressure is based on 875 psig system pressure plus shut-off head. Pressure parts are designed to conform to Section III, Class A, of the ASME Boiler and Pressure Vessel Code.

Primary system piping material is stainless steel Type 304 and will conform to piping standards prepared by ANL. These standards will be more stringent than the current USAS B31.1 (Power Piping) Standards and will be based on the proposed USAS B31.7 Standards (Nuclear Piping). All joints are welded and the pipe is designed for a temperature and pressure of 250°F and 875 psig except in those sections where valve closure can impose pump shut-off head in addition to system pressure.

c. Primary Emergency and Shutdown Cooling System

This system utilizes the same components as the primary coolant loop, with the exception that motive power for the main circulation pumps is provided by the d-c pony motors mounted on the shafts of the pumps in tandem with the main motors. Under emergency operation, or after reactor shutdown, these pony motors operate the pumps at reduced speed to remove decay heat from the reactor core. Either pump operated by its pony motor is capable of providing the 1900-gpm flow required under these conditions. The pony motors are connected to the Class-I (uninterrupted) power bus.

d. Degasification System

The degasification system is part of the low pressure clean-up and pressurization branch of the main loop. This system is provided to remove fission product gases from the primary coolant and to discharge these gases to the off-gas system for disposal. During normal operation, some oxygen and hydrogen produced by radiolytic decomposition of the coolant will also be removed by this system; however, radiolytic oxygen and hydrogen in the primary loop will reach an equilibrium concentration which is little-influenced by the operation of the degasification system.

A portion of the primary flow (200 gpm) is withdrawn from the primary loop downstream of the heat exchangers and discharged into the degasifier through the pressure control valve. A vacuum is maintained in the degasifier by two vacuum pumps which discharge to the off-gas system. The gases removed from the primary coolant are diluted with air after the vacuum pumps in order to maintain a hydrogen concentration below the lower explosion limit in air.

In the off-gas system (see Section IV-M), gas extracted from the degasifier passes through a moisture separator and filter system, and discharges with the Reactor Containment Building exhaust air through the 250 ft stack. The off-gas filter system is designed to remove particulate matter and halogens which may be released from the degasifier during normal operation.

Normally the amount of activity in the effluent gases from the degasifier vent system will be sufficiently small to permit release to the surroundings after mixing with the other exhaust system effluents. If the activity in the gases leaving the degasifier exceeds a preset limit, the off-gas line and the inlet line to the degasifier will automatically be valved off.

e. Purification System

The purification system is also part of the low pressure clean-up and pressurization branch of the primary loop. Water from the degasifier is pumped through the purification system by the transfer pumps to the make-up and water storage tank. The purification system consists of a double bank of separate bed demineralizers equipped with prefilters and a common post-filter.

The ion-exchangers are sized for a normal continuous throughput of 200 gpm; however, for special cleanup purposes during shutdown the ion-exchangers can be valved in parallel to accommodate a flow of 400 gpm. The transfer pumps are also sized for this flow. The ion-exchangers are equipped for regeneration in place and are furnished with sluicing facilities for fresh and exhausted resins.

f. Pressurization System

The primary coolant loop operating pressure of 650 psig is maintained by either of the two parallel-connected pressurization pumps, each with a capacity of 200 gpm. The pumps take suction from the make-up storage tank and discharge to the main loop at a point upstream of the reactor vessel through a flow control valve. Pressure is controlled by the pressure control valve in the letdown line to the degasifier.

A d-c driven (Class-I power) emergency pressurization pump of 30-gpm capacity is also provided in order to maintain the system pressure in the event of utility power (Class-III) failure.

g. Soluble Poison Injection System

This system serves as a backup for the normal reactivity control system, and provides an additional means of shutting the reactor down in the unlikely event that control and safety rods should become inoperable. The system is manually initiated and consists of a storage tank for cadmium nitrate solution piped to the suction header of the pressurization pumps. The stored poison concentration is below saturation at room temperature. An electrical resistance heating coil maintains a slight circulation in the storage tank.

h. Irradiation Facilities Cooling System

Approximately 500 gpm of primary coolant are used for cooling of the beam tube facilities and their associated reactor vessel nozzles. This water is taken from the inlet plenum in the reactor vessel and passed through the beam tube thimble jackets and nozzles and is returned to the reactor coolant outlet line via a manifold arrangement.

2. Secondary Coolant System

a. General Description

The secondary coolant system provides a means for removing thermal energy from the primary system and transferring it to the cooling tower for subsequent dissipation to the atmosphere. The cooling tower also serves as the heat sink for auxiliaries such as the shield plug cooling system, pump coolers, and water chillers. The pool and canal cooling system and the air conditioning units are cooled by the chilled water system.

b. Secondary Coolant Loop

Secondary cooling water is circulated through the tube side of the primary and auxiliary heat exchangers by two secondary circulation pumps (each 50% capacity). Primary coolant temperature to the reactor is controlled by regulating the flow of secondary coolant through the primary heat exchangers. Heat absorbed by the secondary coolant is rejected to the cooling tower located due north of the Reactor Containment Building. The two secondary circulation pumps are supplemented by two emergency and shut-down circulation pumps which have special motor windings so that they can be operated at rated capacity on Class III service or at reduced capacity on Class II service. Make-up water for the cooling tower is obtained from the nearby Chicago Sanitary and Ship canal and treated in the Water Treatment Building.

3. Auxiliary Coolant Systems

a. Pool and Canal Cooling System

The pool and canal cooling system is designed to remove the heat generated in the Reactor Pool during operation of the reactor as well as the decay heat evolved from stored spent cores.

The pool and canal system consists of five main compartments. These are: the Reactor Pool, the Spent Core Storage Pool and Transfer Canal, the Core Parts Storage and Transfer Pool, the Main Canal, and the Rabbit Pool. The two latter compartments are located outside the Reactor Containment Building. The Reactor Pool contains facilities for storage of two defective cores.

The Reactor Pool with the defective core storage facilities, and the spent core and core parts storage areas are furnished with separate cooling circuits each circulating 200 gpm of pool water and sized to dissipate 0.4 MW of heat. This capacity is sufficient to handle the heat losses to the pool from the reactor operating at 100 MW as well as to remove afterheat from a full core storage area with the spent cores in various stages of decay. The secondary side of both of the pool and canal heat exchangers is connected to the chilled water system to which the heat is rejected. The two cooling loops are interchangeable.

The facilities for storage of defective cores have a thermosyphon cooling loop submerged in the Reactor Pool. Provision is made for attachment of a small vacuum operated degasser to the defective core storage enclosure. All gases so removed are discharged into the main off-gas system upstream of the radioactivity monitoring station.

The pool and canal system located in the Reactor Containment Building is furnished with a separate bed 200-gpm demineralizer for clean-up purposes. Portions of the cooling loop flows can be bypassed through these ion-exchangers as required. Continuous filtration is provided through the separate 60-gpm sweepflow purification system.

Purification of the Main Canal and the Rabbit Pool located outside the Reactor Containment Building (in the Active Material Handling Building) is accomplished with a separate 60-gpm purification loop using filters and a mixed-bed demineralizer.

b. Shield Plug and Collimator Cooling System

The shield plug and collimator cooling system is designed to remove the heat generated in the beam tube shield plug or beam tube collimator during reactor operation. The cooling circuit consists of low pressure piping, heat exchanger, circulation pumps, and a water purification system. Cooled demineralized water is circulated to all beam tube facilities. The heat exchanger rejects heat to the secondary coolant system. A small slipstream is continuously removed and directed to a filter and demineralizer unit from which it is returned to the main stream.

c. Primary and Pool Water Storage System

The primary coolant loop and all pool and canal compartments are piped to an 80,000-gallon storage tank located outside the Reactor Containment Building. This tank is sized to accommodate enough water to lower the level in the Reactor Pool to the reactor vessel top flange, or to completely drain any one of the other compartments. The primary coolant can be drained either directly or through the primary purification ion-exchanger into this tank. Inadvertent draining of any of the pools is

prevented by avoiding bottom pipeline penetrations, insofar as is possible.

A 2,000-gpm pump is provided for the transport of water to and from the storage tank. A steam-fed heating coil is utilized to prevent freezing of the tank and pipelines.

H. Instrumentation and Controls

The AARR will be provided with instrumentation and control systems to permit safe operation of the facility under normal and abnormal conditions. Redundant electrical systems will ensure that electrical power is available for operation of the essential instrumentation and control systems under all credible conditions.*

The instrumentation system can be divided into three principal categories: nuclear, process, and radiation monitoring.

1. Nuclear Instrumentation

The nuclear instrumentation system will provide (1) continuous monitoring of the neutron flux from source level to approximately 300% of full power, (2) protection against excessively short period and excessive flux level, and (3) signals for the automatic flux control system. Detectors for the wide-range log channels will be located inside the reactor vessel in thimbles that extend to positions just outside the permanent reflector. Detectors for all other channels will be situated in thimbles immediately outside the vessel.

a. Channel Description

(1) Wide-Range Log Channels

Three redundant channels will be employed to provide continuous indication of reactor flux level and period from source level to approximately 300% of full power. Through period interlocks in the shim-rod control circuits, these channels will also impose limits on the rate of power rise, until the power level at which the power range protective channels become effective is reached. Conventional dual-range fission chambers/counters are utilized as detectors to cover approximately ten decades of neutron flux measurement. In the lower five decades conventional log counting circuits are employed. In the remaining range the statistical current fluctuation technique referred to as the "Campbell," or "Mean Square Voltage" system is utilized to provide logarithmic indication of the neutron flux level.

(2) Power Range Protective Channels

Three redundant linear channels will be included to provide protection against both excessively high level and excessively high rate of rise of neutron flux (see Figure IV-34). The effective range of these channels will be from approximately 5% to 150% of reactor full power. Uncompensated ion chambers will be utilized as detectors. Detector output signals will be fed into linear current amplifiers to yield an output for indication, recording and bi-stable trip operation. Rate of change of flux is derived from this signal in a simple R-C differentiator.

* See Section IV-I.

FIG. IV-34
POWER RANGE PROTECTIVE CHANNEL
(THREE REDUNDANT CHANNELS ARE PROVIDED)

(3) Automatic Reactor Power Control Channels

Three redundant linear channels, utilizing uncompensated ion chambers as detectors, will provide signals proportional to the neutron flux to the automatic reactor power control system. This system will maintain either a preset power level, or a level commensurate with the coolant flow rate, whichever is less, in the range from approximately 10% to 100% of full power.

(4) Wide-Range Linear Channel

A linear channel with range switching will be included to provide sensitive indication of the neutron flux level in the intermediate and power ranges. A compensated ion chamber will be utilized as a detector in conjunction with a current amplifier.

b. Safety and Serviceability Features

For maximum safety and serviceability, the following features will be incorporated:

(1) Power Supplies

The power for nuclear instrumentation will be supplied from three separate Class-I (uninterrupted) d-c power sources. It is expected that d-c to d-c converters will be employed to supply all necessary voltages. Each channel that is employed in a two-out-of-three configuration will be supplied from a separate source.

(2) Redundancy

Redundant nuclear channels are provided as indicated above to increase safety, permit on-line testing, and reduce the number of false scrams. These are connected in a two-out-of-three logic configuration. A channel trip signal in one channel, whether generated by component failure, channel calibration interlocks, or apparent off-normal values of measured neutron flux, will always leave the remaining channels in a one-out-of-two configuration. To further reduce the likelihood of simultaneous failures from a common cause the redundant channels will be isolated physically as much as practicable.

(3) Fail-Safe Features

The fail-safe principle will be followed in the design of the nuclear instrumentation. Wherever possible, direct channel trips will be provided through interlocks for all conditions that would disable the normal channel operation, e.g., interruption of channel continuity for calibration. Similarly all component chassis that are associated with alarm or scram functions will be provided with interlocks so that the removal of one of these chassis will automatically generate the alarm or initiate the protective action associated with that channel.

(4) Testing and Calibration

The nuclear instrumentation will be provided with built-in circuits for the signals listed below. These are individual, so that isolation of redundant channels is maintained during testing. Where feasible, for testing of high level trip circuits, the test signal will be superimposed on the existing reactor-derived signal. For calibration, the test signal only will be applied but with automatic concurrent tripping of the affected channel. The test signals will be injected at such a point as to include as much of the channel as possible.

(a) Wide-Range Log Channels

Signal source for checking both the log count rate and the log mean square circuit steady-state calibration and level trip circuits. Ramp signal for testing the period circuits and trips.

(b) Power Range Protective Channels and Automatic Control Channels

Current source for testing steady-state calibration and level trips. Ramp signal for testing power rate or change circuits and trips.

2. Process Instrumentation

Significant temperatures, flow rates, liquid levels and pressures will be measured at various appropriate points in the reactor process systems. At certain points conductivity and pH will also be measured. Signal transmission will, in general, be electric. Valve actuators will generally be pneumatic, but will utilize electric control signals. Measurement signals will, in general, be transmitted to the central control room.

Transmitters that are vulnerable to the radiation levels existent at the point of measurement will be installed as two-part units; the part susceptible to radiation will be mounted in an acceptably low radiation environment.

Redundant channels will be used for process measurements that furnish signals to the reactor safety system. In certain instances, where repair or replacement of defective components is not possible when the reactor is running, spare transmitters and sensors will be installed. If the corresponding units in an operating channel should fail, the spare sensor and transmitter can be quickly connected as substitutes. The same additional considerations with respect to redundancy and fail-safe features that were described for nuclear instruments apply to the process instrumentation.

3. Radiation Monitoring Instrumentation

The radiation monitoring instrumentation will comprise three subsystems. They are: health physics monitors, process monitors, and plant

effluent monitors. This equipment will be located as required in the facility to provide information for the protection of plant personnel and to assist in maintaining radiation levels within the requirements of the AEC Manual, Chapter 0524, "Standards for Radiation Protection."

Radiation level and alarm signals will be transitted to and displayed in the control room. Where required, the monitors will be equipped with local annunciation. The annunciation will consist of distinctive audible and visible signals which will be uniform for all such applications in the facility. To the maximum extent practicable, the equipment will be supplied for Class-I (uninterrupted) or Class-II (local prime mover-generator) electrical power as required.

Health physics monitors will be placed in such locations as the Reactor Containment Building Service and Experiment Floors; the Active Material Handling Building gas rabbit equipment room, filter rooms, waste disposal area, and pool area; and the Laboratory and Office Building Service Floor. Included are area monitors and particulate monitors. In general, gamma-beta activity will be monitored; however, neutron level will also be monitored where applicable.

Process liquid monitors will indicate the activity of the primary and secondary reactor coolant and that of the Reactor Pool, and the Spent Core Storage Pool. The gross activity of the primary coolant will be monitored immediately after it leaves the reactor in order to provide an early indication of any major failure of fuel element cladding. In addition, monitoring of the specific fission product activity in the primary coolant will be utilized for a more sensitive indication of fuel element failure. For this purpose, a sidestream will be taken from the reactor primary coolant outlet. A delay introduced between the sampling point and the monitor will permit decay of N^{16} activity, while ion-exchange beds will isolate fission product iodine for monitoring. Detection of activity in the secondary system will result in automatic stoppage of the blowdown and the make-up water supply.

Plant effluent activity will be monitored in the Reactor Containment Building exhaust system, the off-gas system, and the combined plant effluent leaving through the stack. The Reactor Containment Building exhaust monitors will actuate an alarm at a preset activity level somewhat above the normal operating level but in no case exceeding the maximum permissible level for continuous release. At a higher level, not more than the maximum permissible level for instantaneous release, these monitors will automatically trip the Reactor Containment Building isolation system (see Section IV-L). The off-gas system monitor will automatically stop the release of off-gas if the activity exceeds a preset limit. The stack effluent monitor will continuously sample stack gas in a sidestream drawn from a point downstream of all plant effluent connections to the stack. This monitor will be a combination particulate, gas, and iodine monitor and will measure the activity of these components separately.

4. Reactor Control and Safety System

The AARR reactor control and safety system will be designed to collect information from the various signal sources, determine when an

abnormal condition exists and automatically and/or upon operator instruction take appropriate action.

The information collection and display will be centralized in a control room situated outside the containment shell. The control room will contain all indicating and recording instruments and controls necessary to maintain the reactor in a safe status. This equipment will be mounted in a centrally located control console and separate instrument panels. The console will be arranged so that the operator can readily reach and manipulate all necessary controls. The instrument panels will be a graphic or semi-graphic display arranged so as to provide maximum unobstructed visibility of panel-mounted indicators for the operator at the console.

In addition to normal process and nuclear indicators and recorders the control room will house the plant annunciator. The annunciator will provide audible and visible alarms for significant off-normal plant conditions. The portion of the annunciator devoted to indication of reactor scram signals will incorporate a distinctive identification of the first scram signal in a sequence of several such signals.

A reliable reactor safety system will be provided to shut down the reactor automatically and rapidly, by releasing the shim safety rods, to prevent safety limits from being exceeded. In general, physically and electrically independent redundant channels will be used for each of the scram parameters in conjunction with two-out-of-three trip logic. Possible exceptions are simple direct-acting devices, such as a scram push-button. Provision will be made for on-line testing of the safety system. To the maximum extent practicable such tests will include an entire instrumentation channel from the primary sensor to final control contact. The system will be designed so that the failure or removal from service of a single component or channel will not negate safety system action when required.* Conditions resulting in automatic reactor scram will include:

1. Ratio of neutron flux to primary coolant flow - high
2. Rate of rise of neutron flux - high
3. Primary coolant pressure - low or high
4. Primary coolant flow - low
5. Primary coolant activity - high
6. Reactor vessel inlet temperature - high
7. Seismic disturbance

Control of reactor power will be based on reactivity changes effected by movement of the control rods. A total of five such rods are included as described in Section IV-B. Interlocks in the rod drive circuits will impose limitations as required for proper and safe operation of the reactor throughout the entire operating range.

For the normal operating mode there are two different operating conditions. In the "Start" condition all five shim rods will be moved simultaneously with withdrawal limitations automatically imposed, mainly by the reactor period. In the "Run" condition, which covers the range from

* The basic arrangement of rod scram circuits is shown in Figure IV-35.

- NOTES:
- 1. P1, P2 & P3 INDICATE SEPARATE SOURCES OF POWER.
 - 2. (A) - CURRENT INDICATOR
 - 3. K_{1A}, K_{1B} & K_{1C} ARE CHANNEL TRIP RELAY CONTACTS FROM A GROUP OF THREE REDUNDANT SAFETY INSTRUMENTATION CHANNELS; K_{2A}, K_{2B} & K_{2C} ARE FROM A SECOND GROUP, ETC.
 - 4. [F.A.S.] IS A FAST ACTING SWITCH CONTROLLED BY "FAST SCRAM" VARIABLES
 - 5. R - RESISTOR FOR TESTING

FIG. IV-35
ROD SCRAM CIRCUITS
BASIC ARRANGEMENT

approximately 10% to 100% of full power, flux control will be accomplished normally by the three-channel servo drive of the shim-regulating rod. Because of the limited stroke of the regulating rod drive, readjustment of the shim rods will also be required occasionally in the "Run" condition. Automatic shim insertion will be employed, but only manual withdrawal of the shim rods will be permitted.

In addition to the normal operating mode, other modes may be provided for operation of the reactor without pressurization and possibly without forced flow of the primary coolant. Operation in these modes would be limited to low power levels and would be utilized for the initial determination of reactor constants and for checkout after installation of a new core. Group movement of the rods would not be permitted in these modes, and only manual control would be employed.

Because of the restart problems arising from the poison buildup during low power operation following a reactor scram, it is most desirable to avoid scram whenever possible and to maintain the power at the highest safe level consistent with the available coolant flow. To cope with abnormal conditions, which if not corrected could possibly result in a reactor scram, several corrective actions are incorporated in the control system. These corrective actions, listed in order of increasing severity are: (1) "Rod Withdrawal Inhibit" - prevention of withdrawal of any shim rod, (2) "Setback" - a reduction in the automatic control power demand signal resulting in the servo insertion of the regulating rod, (3) "Reverse" - an electric-motor-driven insertion of all shim rods, and (4) "Fast Reverse" - an insertion of the shim-safety rods by the air motor drives.

No unique requirements are foreseen at this time for control of the process variables. Conventional methods and equipment are to be used.

I. Electrical Power Distribution System

1. Incoming Power

Electrical power is furnished to ANL by the Commonwealth Edison Company by means of two separate 138-kV lines. The Commonwealth Edison Company lines feed through circuit breakers to a common bus at ANL Facility 543. The lines are protected by means of high speed relaying, and Argonne has suffered only two power interruptions in eighteen years. The first power interruption occurred on April 23, 1966, and lasted for one hour and forty-five minutes. The second interruption occurred on February 9, 1967 and lasted only fifteen seconds.

Facility 543 is the primary substation which supplies electrical power to ANL distribution substations known as Facility 544 and Facility 549A. Power is supplied to Facility 544 by means of two 13-kV overhead lines which are fed from two 10/13.3 MVA, 132-12.5 kV transformers at Facility 543. The ZGS facility receives its normal electrical service from Facility 549A which is supplied from Facility 543 by means of one 138-kV overhead line. (See Figure IV-36.)

Facility 544 distributes electrical power to all ANL buildings and facilities except ZGS. Interconnections between Facilities 549A and 544 are provided by means of lines R1, R2 and Zr5 (LR3). Transformer capacities at Facilities 549A and 549B are sufficient to carry the load at Facility 544 in addition to their own requirements. At the present time, spare capacity at Facility 544 is adequate to carry the existing occupancy type loads of ZGS and High Energy Physics plus an additional 8 MVA of power available for AARR and future loads.

With the addition of AARR to the system, another 132-12.5 kV transformer will be installed at Facility 549A. One AARR feeder will be supplied from the new transformer, while the second feeder obtains power from existing transformer T3 at Facility 549A. Between the secondary buses of the two transformers a normally open circuit breaker is installed. In case one transformer must be removed from service the circuit breaker is closed, allowing the second transformer to supply power over either of the two AARR lines.

Should Facility 549A lose its primary power line, then AARR would share the surplus capacity of Facility 544 through feeders R1 and R2. As long as there is any utility electrical power available at ANL and either of the two AARR supply lines is still intact, electrical power can be supplied to the AARR site.

2. AARR Electrical Distribution System

a. General

The power distribution system for the AARR Facility (see Figure IV-37) provides three levels of reliability, Class-I, Class-II or Class-III as follows:

- (1) Class-I power serves equipment such as the emergency pony

FIG. IV-37
ONE LINE DIAGRAM OVERALL FACILITY
DISTRIBUTION - ELECTRICAL

motors for the primary circulation pumps, the emergency pressurizer pump, and reactor controls and instrumentation, which cannot tolerate any interruption of power. This emergency power supply consists of two storage battery banks for power and three storage battery banks for instrumentation and control equipment.

(2) Class-II power serves selected equipment required to operate during loss of incoming power but which can tolerate the brief interruption of service during the starting of the emergency generators (see Figure IV-38).

(3) Class-III power provides normal service to AARR from the two separate incoming 13-kV lines A1 and A2.

b. Class III - Normal Power

A modified primary-selective circuit arrangement is utilized to distribute power at the facility. Electrical service is delivered to AARR by means of two separate 13-kV overhead lines. Each line is capable of supplying all required electrical power for AARR.

At the AARR Facility the two 13-kV overhead pole lines A1 and A2 terminate at the west end of the AARR outdoor substation. All power stepdown transformers are supplied from this 13-kV substation. The two 13.2 - 4.16 kV transformers supply power to the large motors of the facility, and the 13.2 - 0.48 kV transformers provide power for the smaller motors and other building services.

The secondary terminals of the 5000-kVA, 13.2 - 4.16 kV, transformers are connected by enclosed bus duct to the 4160-V switchgear sections A1-1 and A2-1 located in the Electrical Equipment Building. The 4160-V switchgear controls the power to the primary pumps, the secondary cooling pumps, and the air conditioning and process system chiller.

Building service power for the major AARR buildings is supplied from two 480-V substations located on the Switchgear Floor of the Electrical Equipment Building. These substations supply power to motor control centers which are strategically located in the building complex. A third 480-V substation is located in the Water Treatment Building. This substation supplies power to the cooling tower fans and the water treatment equipment. All three substations are interconnected by normally open circuits such that if one transformer is taken out of service, the other two can supply the entire 480-V system load.

c. Class II Power - Prime Mover Generator Supply

Two standby generators each rated at 200 kW for 480-V, 3-phase, 60-cycle service are located on the Battery and Generator Floor of the Electrical Equipment Building. Both generators are diesel-driven and either generator is capable of carrying the entire emergency load. The two generators are connected to 480-V emergency switchgear sections EA-1 and EA-2, respectively. These switchgear units in turn supply power to motor control centers which distribute power to those auxiliaries which require emergency

FIG. IV-38
ONE LINE DIAGRAM EMERGENCY POWER
SYSTEM (CLASS-II)

power during loss of the Class-III power.

Provision is made on the switchgear units EA-1 and EA-2 for automatic starting of the standby generators in the event of failure of either source of normal power. The method being considered for automatic starting consists of four 15-second cranking cycles with four 15-second periods of rest. In the event either unit fails to start after these four cycles the starting cycle is locked out and an alarm actuated in the control room.

In the event of loss of normal power to either Class-II switchgear section, the associated feeder circuit breaker trips and the bus tie circuit breaker will be closed. Simultaneously the associated generator is started automatically and comes up to synchronous speed and rated voltage. Power is supplied to both EA-1 and EA-2 from the energized substation, and the operating generator is available to supply additional power if needed. If utility power is lost to both switchgear EA-1 and EA-2, the tie breaker does not close, or if it should be closed it trips open. Both generators are started automatically and as they reach rated speed and voltage, the generator breakers close automatically connecting each generator to its associated switchgear. During a power outage, the generators are operated independently. However, either generator has the capacity to furnish sufficient Class-II power to maintain the facility in a safe condition.

Upon restoration of normal power, the generators are manually synchronized to the utility power, and the load is manually transferred back to Class-III power. The generators are then disconnected from the switchgear and shut down.

d. Class I Power - Battery Power Supply

In order to ensure a continuous supply of electric power to vital equipment, the inherent reliability of batteries is utilized. Class-I power for the emergency motors consists of two 250-V battery banks. Each battery bank supplies a power panel which distributes power to the primary pony motors and other emergency loads. Normally open tie breakers are installed between the two power panels. The control and instrumentation equipment Class I power system consists of three separate 64/32 V battery banks. Each battery bank is kept separate from the others so that the two-out-of-three redundancy philosophy employed throughout the instrumentation and control circuits is maintained in the Class-I power system as well.

During normal operation, all Class-I equipment is supplied from the Class-III system through battery chargers and conversion equipment. The batteries are always floating on the line. Upon loss of normal power, each bank, which is always connected to its respective Class-I bus, immediately provides the energy to operate the connected emergency equipment.

Each 250-V battery system is sized to supply the total required Class-I emergency motor load for one-half hour. The instrumentation (64/32 V) battery banks are sized to carry their individual loads for one-half hour. Before the one-half hour has elapsed, at least one of the two prime-mover-driven emergency generators will have started. Either of the two generators can supply the total emergency load, including recharging one 250-V battery bank and the total 64/32 V instrumentation battery bank. Therefore in the event of an extended power outage the AARR Facility can be maintained in a safe condition.

3. Lighting System

In general, fluorescent lighting fixtures are furnished throughout the facility in offices, laboratories, and service areas. In the reactor area high bay, color-corrected mercury fixtures are used, with auxiliary fluorescent lighting under the instrument balcony.

Emergency lighting units consisting of combination battery-operated lights and battery charging equipment are mounted on shelves throughout the facility. These automatically provide emergency lighting in the event of power failure.

Illumination levels are 75 to 100 foot-candles for offices, laboratories, the control room, and the Reactor Containment Building Experiment Floor and balcony areas, and about 30 foot-candles in service and equipment areas.

4. Communication and Alarms

A public address system is provided for general paging. The paging system is zoned, and microphones are located at the receptionist's desk, in the control room, and in other strategic locations. The control room microphone has priority over all other microphones in the system.

Fire protection is provided by means of a fire detection and sprinkler system, and a smoke detection system. Actuation of any of these systems energizes controls in zone panels and fire alarm annunciators, and sends signals to the site fire station. A main fire alarm annunciator is located in a corridor of the Laboratory and Office Building. Auxiliary fire alarm annunciators are located in the control room, outside the Electrical Equipment Building, and outside the Active Material Handling Building.

A fire fog spray system is located around the 5000-kVA transformers in the outdoor switchyard.

5. Experiment and Laboratory Power

Power for the use of experimenters is provided by two 480-V, three-phase, and two 120/208-V, three-phase power distribution panels at two locations in the Reactor Containment Building. These panels provide central distribution points from which the 480-V, 208-V or 120-V power may be run to individual experiments via conduit and cable trays around the biological shield instrument balcony.

Electrical power is provided in each laboratory by a 120/208-V, three-phase panel with 100-Ampere mains and branch circuit breakers in each panel.

J. Reactor Containment Building

The Reactor Containment Building is a steel-lined reinforced concrete cylindrical pressure vessel with a flat bottom and an ellipsoidal dome. The containment vessel will have a 120 ft inside diameter and be approximately 120 ft in overall height. The internal volume is approximately 1.08×10^6 cubic feet. (See Figs. IV-39 through IV-43.)

The building walls are 2'-0" in thickness above the Experiment Floor and 2'-6" thick below this level. The dome is 1'-0" thick lightweight concrete and is covered with a roof coating. The limits for the primary stresses for this structure have been set so as to ensure that the normal working stresses of the materials used will not be exceeded while under the maximum loading condition of a MCA, plus the normal earthquake load, plus the wind load, plus live loads, plus dead load and that the yield strength of the materials will not be exceeded even under MCA conditions plus three times the normal earthquake load, plus the wind load, plus live loads, plus dead load. This latter loading condition is considered to impart to the structure the maximum stresses that can be developed from any credible combination of the following loads:

1. Load due to an internal positive pressure of 6.0 psig.
2. Load due to a barometric swing of ± 0.5 psig.
3. Load due to an earthquake having a horizontal acceleration of 0.075 g and a vertical acceleration of 0.055 g.
4. Wind loading due to a wind of 90 mph, simultaneously with earthquake loading.
5. Internal liner peak temperature of 123°F.

The spirit of Section III - Nuclear Vessels - of the ASME Boiler and Pressure Vessel Code will be followed for the design and combinations of stresses for the liner. The ACI Building Code (ACI-318-63) will be followed for the design of the reinforced concrete portion of the structure.

The Reactor Containment Building and associated penetrations will be designed to permit no more leakage than 0.085 %/day of the total free volume at an overpressure of 5.4 psi. The building contains a Service Floor, Experiment Floor, Instrument Balcony, and a service platform located around the periphery of the reactor biological shield, associated pool shield walls, and part of the building wall. Normal access to the Reactor Containment Building is by means of air locks located on the east side of the building. These air locks connect with the Laboratory and Office Building at the Experiment Floor (El. 746'-0") and with the Active Material Handling Building at the service platform (El. 781'-8"). There is an additional emergency air lock located on the northwest side of the Experiment Floor that permits egress to grade (El. 745'-6"). Two equipment access containment bulkhead doors are used to transfer equipment and/or components into or out of the building. A 12 ft x 12 ft opening at the Experiment Floor level (El. 746'-0") permits truck access to the Experiment Floor from the first floor of the Active Material Handling Building. A rectangular hinged and dogged containment bulkhead door seals this opening during reactor operation. A double inflatable seal arrangement is used to maintain containment integrity. It is anticipated that this door may be opened during each reactor shutdown so that experimenters' equipment may be transferred

FIG. IV-39
ACTIVE MATERIAL HANDLING AND
REACTOR BUILDING
BUILDING SECTION-1

FIG. IV-41
REACTOR BUILDING EXPERIMENT FLOOR PLAN

FIG. IV-42
REACTOR BUILDING INSTRUMENT BALCONY
PLAN

FIG. IV-43
REACTOR BUILDING PLAN AT TOP
OF REACTOR POOL WALL

to and from the Reactor Containment Building.

An 11'-0" diameter opening is provided at the northwest side of the Service Floor (El. 715'-0") for transfer of reactor equipment and plant components to and from that level through an equipment areaway. A circular, hinged and bolted, containment bulkhead door with a compression seal covers this opening during normal operation. It is anticipated that this door will be opened only during periods of extended shutdown of the reactor. An underwater hatch is provided to permit the transfer of radioactive materials between the canals in the Reactor Containment Building and the Active Material Handling Building.

The Service Floor (El. 715'-0") houses the primary coolant system except for the reactor, the primary strainer and piping leading thereto, and other associated equipment. Access to this floor is by means of a shielded stairway from the Experiment Floor above, located at the center-line of the building adjacent to the exterior south wall. There are two platforms above the Service Floor main level. One is for access to the control rod drive mechanisms below the reactor, and the other is a general service platform located at elevation 728'-0". Access to the service platform is provided by means of the shielded stairway.

The reactor, Reactor Pool, Spent Core Storage Pool, Core Parts Storage and Transfer Pool, Transfer Canal, and experimenters' equipment used with the various beam and through tubes are located on the Experiment Floor. This floor connects directly with the first floor of the Laboratory and Office Building through a personnel air lock. An instrument balcony is located approximately 16 ft above the Experiment Floor for experimenters' controls and instrumentation. Access to this balcony is by means of open riser steel stairs from the Experiment Floor and from the service platform located above in the southeast portion of the building. A service platform is located approximately 36 ft above the Experiment Floor around the periphery of the reactor biological shield and transfer canal walls, and extends along the building wall to the personnel air lock leading to the third floor of the Active Material Handling Building.

A radial crane with 30- and 15-ton hooks is provided for movement of experiment equipment, shielding and the beam tube and plug coffin. Separate fuel handling facilities are provided above the pools and canals. The Reactor Pool, Spent Core Storage Pool, Core Parts Storage and Transfer Pool, and Transfer Canal will be stainless-steel lined. The top 8-in. of concrete of the Experiment Floor slab are not required for structural purposes. This permits routing of utility trenches through it.

K. Plant Layout and Buildings

1. General

The AARR facility will be constructed at Argonne National Laboratory, DuPage County, Illinois, on a site located northwest of existing Building 330, which houses the CP-5 reactor (see Figures IV-44 through IV-51). Access to the AARR site will be by means of an extension to the existing east-west road located north of and parallel to Building 330 and by a new north-south road extending southward to Bluff Road. The principal buildings and structures other than the Reactor Containment Building are as follows:

- Active Material Handling Building
- Laboratory and Office Building
- Electrical Equipment Building and Outdoor Switchyard
- Water Treatment Building
- Cooling Tower
- Exhaust Stack

2. Active Material Handling Building

The Active Material Handling Building is located south of and adjacent to the Reactor Containment Building. It is a multi-level building extending three stories above grade and two stories below grade. The above-grade portion of the building is a steel framed structure with exterior walls of insulated metal wall panels. Access to the first floor level of the building (El. 746'-0") is provided from the Laboratory and Office Building, and from the outside, through personnel and truck doors located on the south side of the building. Access to below-grade levels of this facility is by means of an enclosed stairway, an elevator and an equipment area-way. Access to the third floor (El. 780'-6") from the Reactor Containment Building is provided by means of an air lock.

In addition, an 8 ft x 10 ft combined passenger and freight elevator of 10,000 lb capacity serves the basement and the three above-grade floors of the Active Material Handling Building and both floors of the Laboratory and Office Building (Els. 731'-0" and 746'-0"). Stairs located on the east and west sides of the Active Material Handling Building also provide access to the first, second, and third floors.

The Basement Floor (El. 731'-0") provides shielded space for the waste resin area and the liquid waste storage area. The function of the liquid waste storage system is to collect all liquid wastes originating in the Reactor Containment and Active Material Handling Buildings, to neutralize them, sample or monitor them, and then discharge them. The storage of liquid wastes is provided by two 6,000 gallon tanks. The spent resin disposal system provides collection tankage for all AARR disposable resins, provision for water sluicing of the resins to 30- or 55-gallon drums, and subsequent dewatering, sealing and preparation of drums for storage or shipping.

In addition there is a sampling and chemical addition room, a filter room which houses the fans, filters and other associated air handling

FIG. IV-44
SITE PLAN

FIG. IV-45
BUILDING ELEVATIONS EAST AND SOUTH

FIG. IV-46
BUILDING ELEVATIONS WEST AND NORTH

SECTION 2
SCALE 1/8" = 1' 0"

FIG. IV-47
LABORATORY AND OFFICE BUILDING
BUILDING SECTION-2

FIG. IV-51

ACTIVE MATERIAL HANDLING BUILD
SECOND AND THIRD FLOOR PLAN

equipment for the Active Material Handling Building exhaust systems, and the Radiochemical Service Room.

The concrete laboratory waste retention tank vault (El. 715'-0") is located below a portion of the basement floor area.

The first floor (El. 746'-0") provides space for the truck aisle, maintenance shop area, shielded gas rabbit equipment room and a large (approx. 3,000 sq. ft.) mock-up and equipment assembly area. A vestibule leading to the Laboratory and Office Building and a storage room are also on the first floor.

The second floor has several work area levels. The mechanical equipment area is at El. 762'-0". The 18 ft x 25 ft open bay to the first floor is also at this elevation. The floor of the Main Canal and the Rabbit Pool and the floor of the work area adjacent to and east of the base of the Rabbit Pool are at El. 758'-0". The work area west of the Main Canal and the 7-ft wide corridor connecting the two work areas are at El. 764'-0".

The third floor (El. 780'-6") provides space for the working area around the Main Canal and Rabbit Pool and for pot and small cask storage. A 10-ft square area southwest of the main canal is designated as a decontamination area. An 8-ft wide door in the west exterior wall is provided for the removal of irradiated material and spent core shipping casks. A 35-ton capacity crane with a 3-ton auxiliary hook traverses the entire length of the third floor level and passes out through an opening in the west wall for 15 ft for the handling of casks, pots and shipping containers. The 15-ft extension to the west allows the crane to transport loads from the 780'-6" elevation to an area outside the building immediately above the truck loading access area at grade level.

A special material store room and the personnel air lock between the Reactor Containment Building and the Active Material Handling Building are also located on the third floor.

The third floor is a confinement area and all doors and hatches leading from this area are gasketed to allow a slight negative pressure to be drawn on this volume. The Main Canal and Rabbit Pool are separated by a water-tight gate and are lined with stainless steel.

3. Laboratory and Office Building

The Laboratory and Office Building is a one story and basement structure located east of the Reactor Containment Building.

The building is of composite construction with exterior walls of masonry, except that the solid wall facing the reactor shell is of concrete, 12-in. thick. Windows are provided in the south, east and north exterior walls. Access to the building is provided by the main personnel entrance located at grade level on the east side of the building, by interior doors from the Active Material Handling Building on the first floor, and through a personnel air lock from the Reactor Containment Building located on the Experiment Floor.

A combined passenger-freight elevator, which is located in the Active Material Handling Building, has an 8 ft x 10 ft platform and a capacity of approximately 10,000 pounds and serves both floors of this building. Stairs are located along the west and east walls.

The control room, relay rack room and cable penetration room are located on the Laboratory and Office Building Service (basement) Floor and are protected from high winds, negative pressure and missiles by reinforced concrete walls and the first floor beam and slab system. Also located on the Laboratory and Office Building Service Floor are the radiochemical laboratories (which will receive experimenter-irradiated samples), four experimenter counting rooms, and the operations shift supervisors' office.

The first floor of the building houses the offices and associated spaces required for operation of the facility. Also provided are experimenter service and research support spaces required to assist the experimenter activities within the Reactor Containment Building.

4. Electrical Equipment Building and Outdoor Switchyard

The Electrical Equipment Building is a one story and basement structure located north of and adjacent to the Reactor Containment Building and west of and adjacent to the Laboratory and Office Building. The building is of composite construction with exterior walls of masonry. Personnel access to the building is provided from the Laboratory and Office Building, and from the outside through a personnel door located in the west wall. A steel roll-up truck door is provided at grade level on the west side of the building.

The Service Floor (basement), located at the same level as the Service Floor of the Laboratory and Office Building, provides space for the battery rooms, the emergency switchgear, motor control centers and associated electrical equipment, and the two emergency diesel-generators. The ceiling of the Service Floor will be designed to preclude damage by tornadoes to emergency electrical equipment located on the Service Floor.

The first floor, located at the same elevation as the first floor of the Laboratory and Office Building, provides space for the 4160-V switchgear, 480-V substation and motor control centers. Removable floor hatches, located along the west side of the building are provided for equipment access to and from the basement floor. A monorail with a 3-ton capacity manually operated hoist is provided above the hatches for handling of equipment. A 14-ft wide by 12-ft high steel roll-up door located in the west wall provides truck access.

The outdoor AARR Electrical Substation is located north of and adjacent to the building within a fenced-in area.

5. Water Treatment Building

The Water Treatment Building is a one-story structure located approximately 275 ft north of the Laboratory and Office Building and south-east of and adjacent to the cooling tower. The building is a concrete block structure with steel roof joists and built-up roofing. An 8 ft x 10 ft

steel roll-up truck access door is provided.

This building provides space for the cooling tower water treatment pumps; slipstream filtering pumps; other miscellaneous pumps; carbon filters; cation, anion and mixed bed purification units; other water-treating equipment; electrical switchgear and other associated electrical panels and equipment. The secondary cooling pumps, because of their size, are located outdoors, immediately adjacent to and south-west of the cooling tower.

6. Cooling Tower

The induced-draft cooling tower is located north-west of the Water Treatment Building, and consists of three cells. The tower is supported on a reinforced concrete basin.

7. Exhaust Stack

The stack required for the exhausts from the Reactor Containment Building and the Active Material Handling Building is of steel and 250 ft in height with an inside diameter at the top of 4 ft. Exhaust fans located above grade will discharge into the stack at its base. Accessories will include guys and anchorage, aircraft warning lights, lightning rods, and an outside ladder with cage and rest platforms.

L. Reactor Containment Building Ventilation

Ventilation of the Reactor Containment Building is accomplished by a once-through conditioned air supply of approximately 20,000 cfm and recirculating cooling units above the Experiment Floor, and a combination of indirect supply from the Experiment Floor and recirculating cooling units below the Experiment Floor. Figure IV-52 shows a schematic diagram of the Reactor Containment Building ventilation and exhaust system.

The once-through conditioned air supply originates in the mechanical equipment area on the second floor of the Active Material Handling Building, which is adjacent to the Reactor Containment Building. Air enters the Reactor Containment Building after passing through high-efficiency filters which remove atmospheric dusts, heating and cooling coils, and two isolation valves.

The air supply and exhaust system is consistent with the internal design pressure of the building, i.e., all openings are closed automatically and remain gastight at the design overpressure in the event of an emergency. During normal operation, the Reactor Containment Building Experiment Floor operates under a slight negative pressure relative to the atmosphere, and the Service Floor is maintained slightly negative relative to the Experiment Floor.

The total air volume of the building dome above the experiment floor is changed approximately 1-1/2 times per hour. However, air is drawn from the top and sides of the biological shield to provide the equivalent of ten air changes per hour in a circumscribed volume 10 ft above and 10 ft radially outward from the face of the biological shield.

A total of approximately 13,000 cfm of room air is drawn through intakes around the periphery of the Reactor Containment Building pool complex (just above the water surface), around the upper face of the biological shield, at the top of the building dome, and from the movable shield plug cavities. Of the 13,000 cfm, 4,000 cfm (the 2,000 cfm from the dome and the 2,000 cfm from the Reactor Pool) is diverted through two parallel filter streams and fans. The total air quantity is delivered to the stack fans by the Reactor Containment Building exhaust fans.

The air exhausted from the building is monitored by three redundant instruments having one alarm level and one trip level (see Section IV-H). For the trip function the instrument output circuits are arranged in a two-out-of-three configuration. Upon reaching the alarm level, a control room annunciator is actuated. The operator takes corrective action depending on the level, its rate of change, and other factors. He may manually actuate controls that shut down the 15,500 cfm building exhaust fan and reset the speed of the supply fan to produce a total flow of 4,500 cfm, under which condition all building air flows through filter streams before exhausting to the stack.

If the exhaust air trip level is reached, the isolation valves on the supply and exhaust ducts are closed within 3-4 sec, all supply and general exhaust fans are shut down (if they have not already been shut down by a

FIG. IV-52

REACTOR BUILDING HEATING,
VENTILATING, AIR CONDITIONING AND
EXHAUST SYSTEMS FLOW DIAGRAM

preceding manual trip), and a by-pass valve opens to allow the 4,000 cfm clean up exhaust system air passing through the filter streams to recirculate within the Reactor Containment Building. The by-pass valve is interlocked with the inner exhaust isolation valve to ensure opening whenever the isolation valve is closed, and an electrical interlock assures closing of the inner isolation valve (to actuate the by-pass valve through the interlock) if only the outer isolation valve closes as a result of a malfunction.

Restoration of normal operation from either manual or automatic actions must be accomplished manually.

To ensure continuity of filtration at all times, two independent filter streams, each with its own fan, are provided. Each filter stream includes the following (proceeding downstream):

1. Moisture separator to remove water particles entrained in the gas stream should vaporized water be released.
2. High-efficiency particulate air (HEPA) filter to remove radioactive particles.
3. Two beds of impregnated charcoal filters in series to remove radioactive halogen vapors.
4. High-efficiency particulate air (HEPA) filter to remove radioactive dust from downstream side of the impregnated charcoal filters.

The recirculating cooling units which form a part of the normal Reactor Containment Building air conditioning system are supplied with chilled water from the central chilled water system.

M. Radioactive Waste Handling

1. Gaseous Waste

a. Off-Gas System

The off-gas system provides for the disposal of all radioactive, or potentially radioactive, waste gases from the primary system degasifier.

The operation is as follows:

(1) Normal Cycle

During the normal cycle, the waste off-gases pass through a vent condenser and a liquid trap before being diluted with air. The total gas flow then passes through a high-efficiency particulate filter, and the impregnated charcoal halogen vapor filters, radiation monitoring station, and exhaust blower. The gases are discharged by the 25 cfm off-gas vent system blower into the 250-ft high stack via the building exhaust system. The suction side of the off-gas system is maintained at a partial vacuum during operation, to prevent out leakage of active gas.

(2) High Activity Cycle

In the event that the level of activity reaches a point which prohibits continuous discharge to the atmosphere, the radiation monitoring equipment automatically isolates the degasifier and all gases are retained within the primary system.

b. Tank Vent System

A separate ventilation system is provided to remove radioactive products released from tank vents. Room air is drawn into the tanks and evacuated through high-efficiency particulate filters, and discharged to the 250-ft high stack via the common exhaust plenum located in the Active Material Handling Building.

2. Liquid Waste

Liquid wastes generated within the AARR plant, are disposed of as follows:

a. Effluents from suspect usage, such as laboratory drains, are collected in suspect retention tanks and monitored, by evaporation of samples, to determine the activity level. If the level is below prescribed limits,* the liquid is released for drainage to the Laboratory Waste Treatment Plant. There, the liquid waste is chemically neutralized, disinfected and counted prior to release to the out-fall sewer. If the liquid waste is above prescribed activity limits, it is pumped to tank trucks for transfer

* Gross count (alpha plus beta) < 100 disintegrations/min-milliliter (d/min-ml), or gross count (alpha plus beta) < 1000 d/min-ml with alpha < 5 d/min-ml and Sr^{90} < 10 d/min-ml.

to the Reclamation Building, where the activity is reduced to a solid state by evaporation, filtration, ion exchange and/or flocculation prior to interim storage and/or shipment to a burial ground.

b. Effluent from the primary system including equipment leakage and drains, ion exchange regeneration solution, and from all other known active sources whether inside or outside containment will be collected in the active waste storage tanks, pumped to tank trucks and transported to the Reclamation Building, where the activity is reduced to a solid state by evaporation, filtration, ion exchange and/or flocculation prior to interim storage and/or shipment to a burial ground.

c. Fission-product-contaminated primary system and pool water, which may arise in the event of a fuel element failure, will be processed by a feed-and-bleed method of decontamination to minimize site liquid waste storage requirements. The waste will be processed either in a temporary facility brought in from off-site, or by utilizing existing ANL site facilities. The processed liquid can either be returned to the primary system and pools, discharged to the Laboratory Waste Treatment Plant if treatment reduces the activity to an adequately low level, or pumped into tank trucks for additional processing in the Laboratory Reclamation Building.

3. Solid Waste

Solid wastes are considered those resins within ion-exchange columns which become radioactive as a consequence of cleaning up the primary and pool systems during normal operation, or after a fuel element failure. The spent resins are sluiced from their respective vessels to a collection tank using water as the conveying fluid. From the collection tank, which is in a shielded subsurface room in the Active Material Handling Building, the resins are sluiced to shielded 30- or 55-gallon drums. The resin-filled shielded drums are dewatered, sealed and transported to the ANL interim storage vaults for future shipment off-site for burial.

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V. REACTOR FACILITY OPERATION

The AARR will be operated in a manner which best serves the research needs of the experimenters, and yet guarantees the necessary high degree of safety demanded by the facility. This will require a definite assignment of the responsibility and authority for all activities concerning the AARR facility.

A. Organization

The safe operation of the reactor and facility as well as all associated operational activities will be the responsibility of the AARR Facility Manager.

The AARR Facility Manager will be assisted in the discharge of these responsibilities by an Operations Staff. These personnel will be functionally organized to ensure effective performance and safe, well-regulated operation of the reactor and facilities. Although details of the administrative groupings have not yet been finally determined, the functional organization presently envisaged is shown in the organization chart, Figure V-1. Efficient operation may require that certain functions be divided between administrative groups or that a single administrative group discharge more than one function. The major activities of the Operations Staff are described below according to functional group.

1. Operations Group - The Operations Group will be responsible for all operational functions on the reactor and reactor plant such as (but not limited to) operation of the reactor, fuel handling and surveillance of the reactor and its various systems.

2. Evaluation and Analysis Group - This group will be responsible for analyzing and evaluating reactor performance to ensure maintenance of the facility at a high state of safety, efficiency and utility. This group will also be responsible for evaluating and delineating operational parameters for new cores prior to their use in the reactor proper.

3. Engineering and Maintenance Group - This group will be responsible for planning, engineering and executing reactor and plant modifications which affect the reactor or reactor-associated equipment. A preventive maintenance program will be developed and carried out to maintain the reactor and plant in excellent operating condition.

4. Experimenter Services Group - This group will provide assistance to experimenters in the design and operation of their equipment. In addition, they will review the safety aspects of the experiments and experiment equipment; plan, schedule and supervise the installation of the equipment in the reactor; and maintain safety surveillance over operation of the equipment. All safety reviews will be subject to final approval by the AARR Facility Manager.

FIG. V-1
AARR FUNCTIONAL OPERATIONS ORGANIZATION

B. Training

A formal training program for AARR operators will be established within the framework of the Laboratory's existing Reactor Operator Training Program. For the initial training of the operating staff, a specific program will be prepared and will include instructions on the design basis of the plant, physical arrangements and functions of plant components and proposed operating procedures. Final certification of qualified reactor operators will be made by the Laboratory Director on the basis of recommendations and evaluation of the training records.

C. Procedures, Manuals and Guides

The AARR will be operated through the use of carefully prepared, written procedures. These procedures will be designed to ensure that the operation of the reactor is carried on in a safe well-regulated manner. All procedures including those for startup, normal operation, shutdown and emergency actions for all parts of the plant will be assembled into a Standard Operating Practices Manual.

Written operating procedures (approved by the AARR Facility Manager) will be required for those experiments or pieces of experiment equipment, the malfunction of which might possibly jeopardize any aspect of safe operation of the reactor.

D. Internal Safety Reviews

An AARR Facility Review Committee will be established to review, from a safety standpoint, proposed experiments to be performed in the reactor, major modifications to experiments already installed in the reactor and modifications to the reactor plant. In addition, the Laboratory's standard procedures for review by the Reactor Safety Review Committee will be adhered to.

E. Fuel Management

Rigorous quality control procedures will govern all stages of the manufacture of AARR cores in order to ensure full conformity to specification of delivered cores. Surveillance of quality control procedures and acceptance testing will be the responsibility of the Special Materials and Services Division of the Laboratory. Final verification of core suitability before installation in AARR will be the responsibility of the Evaluation and Analysis Group.

VI. SAFETY ANALYSIS AND EVALUATION

A. Introduction

The safety analysis presented herein is based to a large extent upon the final safety analysis performed by Oak Ridge National Laboratory on the High Flux Isotope Reactor (HFIR), which is now in operation at its design power level of 100 MWt. As indicated in Section IV above, the core, control plates, control drive mechanisms, removable and semi-permanent beryllium reflector, and much of the reactor internal structure to be used in AARR will be identical and interchangeable with the corresponding HFIR components. Operating conditions of inlet temperature, pressure, coolant flow through the core and power level will also be identical with HFIR conditions. Also, identical coolant flow rates will be used in all zones of the reactor except for the Internal Thermal Column (ITC), wherein a substantial reduction in flow rate will be employed in AARR. The major effect of this change upon reactor safety is the beneficial one of reducing the severity of the maximum reactivity accident considered credible, viz., voiding of the bulk coolant in the ITC. The reduced coolant flow rate reduces the reactivity addition rate associated with voids swept into the ITC with the bulk coolant and thereby provides more time for effective safety system action. A detailed discussion of this accident for AARR is presented in Section VI-B. Therefore, with the exception of the foregoing, it is expected that the mechanisms whereby accidents might occur in AARR, and subsequent events within the reactor system, will be for the most part the same as those existing in HFIR. However, since AARR is constructed in a containment building at a different site, the external consequences of accidents will differ from those for HFIR. These consequences, with respect to the maximum credible accident (MCA), are discussed in Section VII below.

B. Accidents Involving Reactivity Transients

1. Introduction

As is the case with all reactors, there are various ways in which reactivity might be increased. Some of these are associated with normal operation of the reactor, while others constitute accidents. Analytical consideration is given in this Section to those accidents that are physically reasonable from the standpoint of available mechanisms for producing the accident, whether or not such an accident is considered to be credible. Appendix C provides the basis for the theoretical model used in analog studies, the generalized results of those studies, and a comparison of the results with experimental data gained from the SPERT program.

An experimental transient heat transfer program now in progress (described in Appendix D) will provide further checks upon calculated results for transients initiated at high reactor power levels.

The analyses of the accidents presented in this report have been restricted to the normal operating mode (Mode 1) of the reactor, as described in Appendix C. Although other operating modes are expected to be provided for the AARR, i.e., without pressurization and possibly without forced flow of primary coolant, detailed definitions of actual operating conditions and associated protective systems have not yet been specified for these modes. Further, previous analog studies for HFIR indicate that the consequences of the postulated accidents are more severe when initiated under normal operating mode conditions. Continued study of reactivity accidents for all operating conditions will accompany the reactor design and operational plans as they develop in the future. Results of these studies will be included in the final safety analysis report for the AARR.*

Much of the analog transient investigation reported herein was performed by Oak Ridge National Laboratory and reported in the final safety report documents for the HFIR (see references 5 through 8 of Section I of this report). In cases where the HFIR analyses are equally applicable to AARR, material from the referenced HFIR safety analysis documents has been reproduced verbatim (except for table, figure and reference numbers) in this report. Such reproduced material is identified in the text by indentation and quotation marks.

Four general types of reactivity accidents, categorized according to their mechanisms for occurrence, were considered. They are as follows:

- (1) Reactivity additions associated with movement of control plates.
- (2) Reactivity additions associated with voids displacing some of the water in the ITC.
- (3) Reactivity additions associated with cold water insertion.

* Ed. note: Due to Project cancellation, there will be no final safety analysis report; references to such a report in this document should be disregarded.

(4) Reactivity additions associated with experiment facilities.

A summary of the more pertinent analog calculations and results pertaining to accidental reactivity additions is presented in Table VI-B-1. It is emphasized that the results in this table are strictly calculational and have not been normalized in accordance with experimental data. All evidence available indicates that the calculations are quite conservative in that they predict higher temperatures, power, and energy releases than would actually be expected.

Typical calculated transient characteristics of the HFIR when subjected to a ramp reactivity addition during high power operation are shown in Fig. VI-B-1. The particular curves shown represent Case 4 of Table VI-B-1. These curves illustrate how the fluid film heat transfer coefficient is set equal to zero when the hot spot heat flux reaches the steady state burnout heat flux. The coefficient is presumed to return to its original value after the power is substantially reduced. Although this effect is not included in the analog calculation, depicted by the solid curves of Fig. VI-B-1, it is presumed to occur when the transient power level drops to approximately one-half the steady-state burnout level (~ 65 MW in this case). The dashed branch of the plate hot spot temperature curve indicates its expected temperature behavior as normal convection cooling is resumed.

Also, the analog calculation does not account for the heat of fusion of the metal, and hence a smooth temperature rise through the melting point (1210°F) is shown in Fig. VI-B-1. However, it is assumed that melting does not occur until the complete heat of fusion has been added which is equivalent to a non-melting temperature increase of 740°F . Thus, hot spot melting does not occur for calculated temperatures less than 1950°F , and the actual temperature curves would remain constant at 1210°F until the calculated value reaches 1950°F .

Several of the cases listed in Table VI-B-1 are not discussed in detail herein. These cases were calculated for the purpose of investigating the sensitivity of pertinent temperatures and energy releases to anticipated variations in parameters such as reactivity insertion rate, control element differential worth, initial power level, etc. No strong sensitivities were observed that might tend to alter the conclusions arrived at after analyzing the specific cases discussed in the following paragraphs.

Much of the ORNL material presented herein was developed to examine the consequences of an ITC voiding accident as conceived for the HFIR. It should be noted that such an accident in AARR would differ considerably from the HFIR concept because of major differences in ITC design and function. While a greater reactivity gain is possible in the more lightly loaded AARR ITC than in the HFIR, the lower bulk coolant flow in the AARR ITC results in a greatly reduced reactivity addition rate, and provides a greater net safety margin for effective safety plate action to limit possible excursions. The reported ORNL analog results for the HFIR ITC voiding accident ($1.3\% \Delta k/k$ in ~ 30 msec) are therefore very conservative if applied to the AARR case. This is discussed in greater detail in Appendix C below.

A summary description of reactivity accidents portulated for the AARR, in terms of assumed mechanisms for occurrence, the resulting maximum reactivity rates and total reactivity additions, are presented in Table VI-B-2.

TABLE VI-B-1
SUMMARY OF PERTINENT ANALOG RESULTS FOR REACTIVITY ADDITION ACCIDENTS

CASE NO.	TIME IN CYCLE (days)	FUEL COOLANT VELOCITY (ft/sec)	INITIAL STEADY STATE POWER (MW)	SAFETY SYSTEM		REACTIVITY ACCIDENT		PEAK POWER (MW)	HEAT FLUX PEAK HOT SPOT (Btu hr ⁻¹ ft ⁻²)	PLATE (a) TEMPERATURE PEAK HOT SPOT (°F)	PLATE TEMPERATURE PEAK AVERAGE (°F)	BULK WATER TEMPERATURE PEAK AVERAGE (°F)	TIME (b) INTEGRATED POWER FOR TRANSIENT (MW-sec)	PLATE SURFACE (c) TEMPERATURE NOMINAL MAX. AT PEAK POWER (°F)
				INITIAL DIFFERENTIAL WORTH ($\Delta k/k$ PER in.)	SCRAM SIGNAL (RATE OR LEVEL)	REACTIVITY ($\Delta k/k$)	RAMP (msec)							
1	0	42.0	100.0	0.00058	20 MW/sec	0.013	30.0	1175	$>2.47 \times 10^6$	2100	590	250	34	598
2	0	42.0	100.0	0.00058	20	0.013	60.0	550	$>2.47 \times 10^6$	1800	450	220	30	
3	0	42.0	100.0	0.00058	20	0.015	75.0	500	$>2.47 \times 10^6$	1900	440	215	30	
4	0	42.0	100.0	0.0035	20	0.013	30.0	1075	$>2.47 \times 10^6$	1850	550	235	28	
5	0	42.0	100.0	0.0035	20	0.015	40.0	1040	$>2.47 \times 10^6$	1875	560	240	28	
6	0	42.0	< 0.1	0.0035	20	0.013	0.0	900	$>2.47 \times 10^6$	1150	400	160	21	
7	0	42.0	< 0.1	NO SAFETY		0.013	0.0	1250	$>2.47 \times 10^6$	1600	530	190	30	
8	~15	42.0	100.0	0.00059	20 MW/sec	0.013	30.0	1250	$>2.47 \times 10^6$	3200	800	300	52	693
9	~15	42.0	100.0	0.00059	20	0.01	30.0	775	$>2.47 \times 10^6$	2600	590	245	37	
10	~15	42.0	100.0	0.00059	20	0.008	30.0	475	$>2.47 \times 10^6$	2000	445	210	27	427
11	~15	42.0	100.0	0.0035	20	0.013	30.0	1075	$>2.47 \times 10^6$	2600	670	260	38	623
12	~15	42.0	100.0	0.0035	20	0.011	30.0	675	$>2.47 \times 10^6$	2000	510	220	28	499
13	~15	42.0	100.0	0.0035	20	0.01	30.0	525	$>2.47 \times 10^6$	1750	440	205	22	
14	~15	42.0	< 0.1	0.00059	20	0.013	0.0	1030	$>2.47 \times 10^6$	2250	640	250	38	
15	~15	42.0	< 0.1	0.0035	20	0.013	0.0	420	$>2.47 \times 10^6$	900	340	170	15	
16	0	42.0	100.0	0.00058	20	0.05/sec		125	2.4×10^6	450	250	160	~11	
17	0	42.0	5×10^{-8}	NO SAFETY		0.005/sec		170	1.9×10^6				<10	
18	0	4.2	< 0.1	0.0035	20 MW/sec	0.013	0.0	900	$>4.7 \times 10^5$	1250	540	200	22	
19	0	4.2	< 0.1	NO SAFETY		0.013	0.0	1250	$>4.7 \times 10^5$	1900	790	280	35	629
20	0	4.2	< 0.1	NO SAFETY		0.015	0.0	1500	$>4.7 \times 10^5$	2050	910	315	40	
21	0	4.2	< 0.1	0.0035	20 MW/sec	0.015	0.0	1400	$>4.7 \times 10^5$	1700	770	265	32	
22	0	4.2	5×10^{-8}	0.0035	20	0.005/sec		32	2.0×10^5					

(a) PLATE TEMPERATURES ABOVE 1950°F INDICATE MELTING OF THE HOT SPOT; ACTUAL TEMPERATURE OF MOLTEN METAL IS LESS BY (1950 - 1210) = 740°F. INDICATED TEMPERATURES BETWEEN 1210 AND 1950°F SHOULD BE 1210°F.

(b) INTEGRATED POWER INCLUDES ALL ENERGY GENERATED DURING TRANSIENT. IN A TYPICAL INITIAL 100 MW CASE (CASE NO. 1), THE DURATION OF THE TRANSIENT IS ~75 msec. THE ENERGY GENERATED IN THIS TIME AT 100 MW IS 75×10^{-3} sec $\times$ 100 MW = 7.5 MW-sec.

(c) THIS IS THE MAXIMUM NOMINAL OXIDE-WATER INTERFACE TEMPERATURE AT THE CORE HORIZONTAL MIDPLANE AT THE TIME THE POWER PEAKS (DOES NOT INCLUDE HOT SPOT EFFECTS). CORRESPONDING SATURATION TEMPERATURE (AT 600 psi) IS 486°F.

FIG. VI-B-1
TYPICAL TRANSIENT CURVES FOR A RAMP REACTIVITY ACCIDENT

TABLE VI-B-2

SUMMARY OF ASSUMED REACTIVITY ACCIDENT CONDITIONS

EVENT	MAXIMUM REACTIVITY INSERTION RATE ($\Delta k/k/\text{sec}$)	MAXIMUM AVAILABLE REACTIVITY INSERTION ($\Delta k/k$)
1. Inadvertent withdrawal of all five control plates at maximum drive speed	0.0044	0.0991
2. Failure of shim-regulating cylinder drive (gravity fall, retarded by hydraulic shock absorber)	0.0065	0.0495
3. Voiding of bulk fluid in ITC (Void producing highest positive reactivity effect entering at ITC coolant velocity - 4 fps)	0.084	0.032
4. Cold-water accident (step-change in inlet water temperature from 120°F to 35°F)	~ 0.218	0.0072
5. Complete voiding of single ITC target tube	Step function	0.001
6. Flooding of single beam tube	Step function	~ 0.0

Selection of a worst case condition from the information provided in Table VI-B-2 requires simultaneous consideration of both the reactivity insertion rate and the maximum total insertion for each case. In the first two events, although large total reactivity insertions are hypothetically possible, the corresponding maximum rates are moderate. For example, to realize the full reactivity insertion cited for Event 1, a time span of approximately 1-3/4 min is required. The corresponding time for Event 2 is approximately 15.4 sec. Thus, in these assumed cases, although the potential total reactivity insertion is large, the rates are sufficiently low essentially to ensure safe termination of the postulated accidents by safety system action initiated automatically, or perhaps even by manual means.

Conversely, the last three events listed in Table VI-B-2 have the potential for high reactivity addition rates, but the values of maximum reactivity gain are small. All of these accidents can be accommodated by the reactor without incurring major core damage even if automatic safety system action should fail.

The remaining accident listed in Table VI-B-2, Event 3, is characterized by combined high values of reactivity addition rate and total worth. On this basis the maximum credible reactivity accident envisioned for the AARR is that resulting from massive voiding of the bulk fluid in the ITC. Analysis of this accident, presented in Section VI-B-3 below, shows that even in this extreme case, the reactor core is protected from damage through safety-plate scram.

Based on the analyses performed, including the comparison with SPERT data (see Appendix C below), it is concluded that no major core damage would result from any of the accidental reactivity additions considered. Some melting of localized hot spot areas is predicted for a few extreme cases. Because of the small fraction of core volume involved, and the anticipated low frequency of occurrence, it is concluded that even these extreme cases do not produce serious problems.

2. Control Plate Accidents

The HFIR final safety report documents treat control plate accidents as follows:

"Two types of control plate removal accidents were considered. One is the runaway case in which the five shim elements (inner shim-regulating cylinder and four shim-safety plates) are driven out of the reactor with the maximum speed of the shim drives, or the shim-regulating cylinder is driven out with the faster regulating drive. It is also possible to add the velocity of the latter to that of the former, for a maximum shim-regulating cylinder movement of about one inch, to obtain the worst condition.

"The shim drive speed is 5.75 in./min (0.096 in./sec), and the regulating drive speed is 15 in./min (0.25 in./sec). With the control elements initially at the clean core, symmetrical, critical position, the differential worth of the

shim-regulating cylinder and of the set of four shim-safety plates is about $0.01 \Delta k/k$ per in. For this initial control element position and for the maximum drive speed condition with all five control elements being withdrawn, the reactivity addition rate would be $0.0044 \Delta k/k$ per sec for the first inch of travel and $0.0019 \Delta k/k$ per sec for about the next two inches, after which it decreases.

"Near the end of the fuel cycle, the maximum reactivity addition rate would be much smaller, and the initial differential worth of the safety plates would be about $0.0008 \Delta k/k$ per inch.

"Starting from low power (<100 kw) and full flow the self-shutdown mechanisms alone limited the peak heat flux to less than the burnout heat flux, when reactivity was added at the rate of $0.005 \Delta k/k$ per sec [17].* The peak power in this case was only 170 Mw, and the total energy release was less than 10 Mw-sec.

"It was concluded from this analysis that reactivity additions resulting from runaway control elements would result in no serious core damage and particularly no melting of the hot spots.

"The other case in which the control elements can add reactivity accidentally is associated with the structural integrity of the shim-regulating cylinder drive mechanism. The use of a control element that must be inserted against gravitational and hydraulic forces introduces a possible problem in connection with failure of the supporting or drive mechanisms; such a failure would result in withdrawal of the control element and thus in a positive reactivity addition. In the event of a mechanical failure of the drive mechanism, a hydraulic cylinder on the lower end of the drive rod will act as a shock absorber (normally used to balance the downward forces of gravity and pressure on the shim-regulating cylinder drive mechanism) and will limit the shim-regulating cylinder maximum withdrawal speed to 39 in./min (0.65 in./sec) even if the force-balancing feature is depressurized. The corresponding maximum reactivity addition rate would be about $0.0065 \Delta k/k$ per sec.

"The results from the analog calculations made in connection with the control element reactivity accidents indicate that for beginning-of-cycle operation at full flow and full power and with the safety plates in the least sensitive position (normally should not be in this position) $0.050 \Delta k/k$ per sec can be inserted continuously without the hot spot heat flux exceeding the burnout heat flux [16]. The hot spot peak plate temperature calculated for this case was only 450°F ,

* Bracketed numbers refer to corresponding cases in Table VI-B-1.

and the total energy release during the brief transient was about 11 Mw-sec. (This includes all energy generated during the transient.)"

3. Voids in the Bulk Fluid in the ITC

The existence of a water-moderated ITC in the AARR introduces the possibility of reactivity additions resulting from displacement of some of the ITC water. Just how much reactivity can be added by this means depends upon what stationary objects happen to be in the ITC at the time the water density is changed. With nothing but water initially in the ITC, the maximum change in reactivity will not exceed $0.032 \Delta k/k$; the void fraction* associated with this change is 0.70. With an array of samples in place, the maximum change in reactivity will be much less.

The design of the ITC will be such as to minimize the possibility that the bulk fluid zone could be voided. All analyses of the current design concept indicate that significant voiding of this zone could not occur. Nevertheless, an analysis is presented of the consequences to the reactor in the event of such a voiding resulting in a reactivity gain.

None of the ORNL transient analyses previously performed for HFIR are directly applicable to this problem due to the different utilization of this zone of the reactor planned for the AARR. These differences are described in Appendix C of this report. Therefore preliminary calculations appropriate to AARR conditions using a modified version of the HFIR analog model were made at ORNL.

In these studies the following reactor model was employed to determine the sequence of events and consequences of an optimum void swept into the ITC under full coolant flow conditions:

(1) Reactivity is added at a constant rate corresponding to $0.084 \Delta k/k$ per sec to a maximum value of $0.032 \Delta k/k$. The reactivity addition time is the time required (3/8 sec) for bulk coolant to travel from the top to the bottom of the active ITC at 4 ft/sec. A second set of calculations was also performed for illustrative purposes at a rate of $0.042 \Delta k/k$ per sec up to a maximum of $0.016 \Delta k/k$.**

(2) The reactor initially is operating at 100 MW; both the high flux and rate trips are operative.

(3) Neutronic, thermal and safety system conditions appropriate to beginning- and end-of-core-cycle times were used in the calculations.†

* Defined as void volume/total flux trap (ITC) volume.

** In typical cases with a loaded ITC, the actual value of maximum reactivity gain will be somewhere between 0.016 and $0.032 \Delta k/k$.

† See Appendix C for numerical values.

(4) Hot spot insulation occurs for an incipient boiling heat flux set at a value of 3×10^6 Btu/hr-ft².

Salient results of these calculations are presented in Table VI-B-3. Examination of this table shows that worst conditions are achieved for end-of-cycle calculations. However, in no case does the hot spot heat flux reach burnout conditions, and metal temperatures remain well below the melting point. Maximum values of net positive reactivity remain well below prompt-criticality, and the total energy release is moderate in all cases. It is concluded, on the basis of these results, that the reactor is adequately protected against fuel melting in the remote event of an ITC voiding accident.

In addition to the foregoing, each of the ITC voiding accident calculations was repeated assuming no protection by any of the reactor safety systems. These calculations were otherwise identical to those described above. The results obtained are briefly summarized as follows:

For the case of reactivity addition at the rate of $0.042 \Delta k/k$ per second (max. value $0.016 \Delta k/k$), hot spot insulation occurs approximately 100 msec after the beginning of the transient. The hot spot melts within an additional 100 msec for the beginning-of-core-life calculation and within an additional 50 msec for the end-of-cycle calculation. The power response is nearly linear in each case during the time of reactivity addition. For the beginning-of-core-cycle case, the power level reaches a peak value of approximately 400 MW followed by a quasi-equilibrium level of 380 MW. Corresponding values for end-of-cycle calculations are 720 MW peak power level and 650 MW quasi-equilibrium. It is difficult to predict the degree and time history of core meltdown under these conditions. In both cases of core cycle time studied, the calculations indicate average metal temperatures above the incipient boiling point which may cause steam blanket insulation and consequent meltdown to occur over a significant portion of the core. Also, the calculation for end-of-cycle core conditions indicates that bulk boiling would occur in core coolant channels producing average power, and hence a major flow reduction in the core would be expected. Since these conditions develop within approximately 0.4 sec after initiation of the transient, insufficient time is available to effect shutdown by the secondary safety system (poison injection) and it is therefore concluded that shutdown would be accomplished through core disassembly.

Obviously, for the other ITC voiding accident studied, which assumes a higher reactivity addition rate and maximum value, the same final conclusion is reached, with meltdown occurring more rapidly. It is emphasized, however, that in neither of these hypothetical cases are the consequences severe enough to cause failure of the AARR primary system. (The containment potential of the reactor vessel is analyzed in Appendix E below.)

The consequences of hypothetical reactivity accidents much in excess of the foregoing are described in Section VIII-B.

TABLE VI-B-3

RESULTS OF ANALOG CALCULATIONS FOR ITC VOIDING ACCIDENTS

RUN NO.	TIME IN CORE CYCLE	INPUT REACTIVITY ($\frac{\Delta k}{k}/\text{sec}$)	PEAK POWER (MW)	INTEGRATED POWER (MW-sec)	MAX. HOT SPOT METAL TEMP. (°F)	MAX. AVG. METAL TEMP. (°F)	MAX. AVG. COOLANT TEMP. (°F)	MAX. HOT SPOT HEAT FLUX ($\frac{\text{Btu}}{\text{hr-ft}^2}$)	MAX. POSITIVE NET REACTIVITY ($\Delta k/k$)	TIME OF PEAK POWER (msec)
1	BEGINNING	0.042	120	2	420	240	160	2.2×10^6	0.002	50
2	BEGINNING	0.084	125	2	430	240	160	2.3×10^6	0.002	30
3	END	0.042	130	15	685	260	165	2.4×10^6	0.002	60
4	END	0.084	170	18	780	285	170	2.8×10^6	0.004	60

4. Cold-Water Accidents

A change in the fuel region average coolant temperature from 150°F to 65°F, which corresponds to a change in inlet temperature from 120°F to 35°F at a constant reactor power level of 100 MW, would add approximately 0.0072 $\Delta k/k$. This value of reactivity gain is based upon a fuel zone temperature coefficient, representative of the average over the specified temperature range, with a value of $8.5 \times 10^{-5} \Delta k/k$ per °F.

For a more realistic case, in which the cold water enters all parallel regions of the reactor, the same change in inlet temperature would add less reactivity due to the positive temperature coefficients in the non-fuel zones.

Analog calculations were made to determine the consequences of sudden cold water injection to the operating reactor using the HFIR analog code at ORNL. The following reactor model was employed in these studies:

(1) The reactor is operating initially at 100 MW; cases were studied both with and without operation of the high flux and rate trips. No other protective measures were assumed.

(2) Neutronic, thermal and safety system conditions appropriate to beginning- and end-of-core-cycle times were used in the calculations.*

(3) Hot spot insulation occurs for an incipient boiling heat flux set at a value of 3×10^6 Btu/hr-ft².

(4) A step function change of -85°F (120°F to 35°F) in inlet coolant temperature occurs and persists indefinitely.

(5) No credit is taken for negative reactivity compensation developed through lowering the coolant temperature in the ITC.

Principal results of the calculations are given in Table VI-B-4. Examination of these results shows that, in spite of the very conservative assumed conditions which are used to define the accident, only a moderate power excursion occurs and the reactor is adequately protected from damage by safety system action.

Although the data of Table VI-B-4 indicate hot spot melting for cases studied without safety system action, these results are highly conservative since the analog model ignores the added margin against incipient boiling due to the increased sub-cooling of core water. If this effect were properly accounted for in the calculations, it is almost certain that hot spot melting would not be indicated.

* See Appendix C for numerical values.

TABLE VI-B-4

AARR RESPONSE TO COLD WATER ACCIDENTS

RUN NO.	SAFETY SYSTEM ACTION	TIME IN CORE CYCLE	PEAK POWER (MW)	QUASI- EQUILIBRIUM POWER (MW)	INTEGRATED POWER (MW-sec)	MAX. CHANGE AVG. COOLANT TEMP. (°F)	MAX. HOT SPOT METAL TEMP. (°F)	MAX. AVG. METAL TEMP. (°F)	MAX. HOT SPOT HEAT FLUX Btu hr-ft ²	MAX. POSITIVE NET REACTIVITY (Δk/k)	TIME OF PEAK POWER (msec)
1	YES	BEGINNING	170	0	20	115	460	230	2.9×10^6	0.004	40
2	NO	BEGINNING	240	200	-	50	> MELTING	270	$> 3 \times 10^6$	0.004	50
3	YES	END	140	0	15	115	700	250	2.6×10^6	0.0025	40
4	NO	END	150	150	-	65	> MELTING	250	$> 3 \times 10^6$	0.0025	60

5. Reactivity Accidents Associated with Experiment Facilities

a. Accidents in Vertical Facilities for Sample Irradiation

The ITC basically is a large water hole in which high-energy neutrons leaking from the reactor core are thermalized to provide a high thermal neutron flux. Thus, the ITC is a large sink for neutrons. Displacement of some of the ITC water by voids lowers its reactivity control, and therefore causes an increase in reactivity.

The largest target tube planned for the ITC has an inside diameter of 0.630 in., and the corresponding interior plan area of this tube represents approximately 1.5% of the total cross-sectional area of the ITC. The reactivity effects of voids in the ITC have been measured in the HFIR Criticality Facility. If the largest target tube were filled with water (no metal) initially, and then were fully voided, the resulting reactivity gain would be less than $0.001 \Delta k/k$. If samples were in place in this tube, and if these samples were ejected as a result of a sudden formation of voids, again the reactivity effect would not be worse than a gain of $0.001 \Delta k/k$.

For the case of voiding of one target tube, therefore, if the power level were P_0 ($P_0 \leq 100$ MW) at the beginning of the voiding, and if the voiding occurred rapidly, the reactor power level would rise rapidly to less than $1.2 P_0$. Subsequently, the power level would increase slowly, on a pseudo-period longer than 30 sec, and there would be time for remedial action, such as a power reduction by the reactor automatic control system. At worst, a high-flux scram would be initiated and safety-system action would shut down the reactor before its power level could go much above 120 MW.*

An important characteristic of the irradiation thimbles and ITC from the reactor safety standpoint is the restriction on permissible reactivity changes in these areas. Experiments will be completely defined and examined on an individual basis. Reactivity changes accompanying the operation of the rabbits will be determined prior to sample changes (by experiment in a reactivity measurement facility,** for example) in order that proper precautions can be taken or adjustments can be made to the reactor. Under these conditions, no severe reactivity accident involving the use of the rabbit facilities can be postulated at this time.

b. Collapse and Flooding of Beam Tubes

The reactivity effect of collapse of a beam tube or a through

* Ed. note: The safety system envisioned at the time of the Project termination would have provided a trip at a flux/flow ratio of 1.3, the equivalent of 130 MW at full flow. No pure high-flux trap was to be provided.

** This would be a facility elsewhere at ANL; there are no plans currently for incorporation of a reactivity measurement facility into the AARR complex proper.

tube under external load has been considered. If one tube were to collapse and be flooded with water, the reactivity effect of this flooding would be negligibly small - perhaps positive, but probably negative. This estimate is based both upon computation and upon measurements in the HFIR criticality experiment program. The flooding of more than one tube would result in a loss of reactivity, because the additional water would act as a neutron poison in the weakly absorbing beryllium reflector.

6. Metal-Water Reaction Analysis Associated with Reactivity Accidents

Fuel melting accidents in the AARR are of interest, not only because of fission product release but also because the metal content of the reactor core may react chemically with water. The reactions of interest are exothermic and hence constitute energy sources for possible damage to reactor containment systems. Appendix B contains basic metal-water reaction data relevant to the materials which compose the AARR core, associated structures and the beryllium reflector. Drawing upon these data, analyses are presented here for assumed accident conditions involving large, reactivity induced power excursions. Most of the material has been extracted from HFIR final safety report documents, specifically Reference 8 of Section I.

Summarized results of reactivity-induced power excursion analyses for the AARR are contained in this Section and in Appendix C of this report. This material presents reactor response data for hypothetical and specific reactivity accident situations considered to be credible for the AARR. The results of major significance for these calculations are given in Table VI-B-1.

"Also, several of the transients previously analyzed were re-analyzed, using a more sophisticated analog model. The new model includes an expression for steam void feedback, based on a correlation proposed by Griffith, Clark, and Rohsenow,¹ which is applicable for subcoolings as low as $\sim 20^\circ\text{F}$ for the HFIR. To obtain some degree of space dependence the core was divided into five axial regions in addition to the hot spot and hot streak "cores." Proper weighting factors for coefficients were applied in each of the five regions.

"A summary of the latest analog calculations made is given in Table VI-B-5. In these calculations more severe accidents than the 1.3% $\Delta k/k$, 30 msec ramp were considered, and in some cases safety rod action was omitted. If one assumes that the model is reasonably correct and that some local melting can be tolerated without destroying the mechanical integrity of the rest of the element, as was the case in the SPERT experiments,^{2,3} with 5 msec and 4.6 msec periods, then it can be concluded that the HFIR can withstand the 1.3% $\Delta k/k$, 30 msec ramp accident with only a small amount of melting and no metal-water reaction. This assumes that the safety rods work. Referring to Case 8, Table VI-B-5, it is seen that the calculated metal hot spot temperature is only 2150°F , and this does not consider a significant increase in radial conduction away from the

TABLE VI-B-5
HFIR ANALOG CALCULATIONS

CASE	FUEL COOLANT VELOCITY	INITIAL STEADY STATE POWER	SAFETY SYSTEM DIFFERENTIAL WORTH	REACTIVITY	RAMP	PEAK POWER	NEGATIVE FEEDBACK FROM MODERATOR FUEL PLATE & VOIDS (PEAK POWER)	VOID FEED- BACK (PEAK POWER)	INTEGRATED POWER	PLATE TEMP HOT SPOT (MAX)	PLATE TEMP REGION 3 (MIDCORE) (MAX)	BULK TEMP REGION 5 (OUTLET) (PEAK POWER)	TIME TO PEAK POWER	MEAT TEMP TARGET AVERAGE (MAX)	SURFACE TEMP TARGET AVERAGE (MAX)	BULK TEMP TARGET AVERAGE (MAX)
	(%)	(MW)		(% $\Delta k/k$)		(MW)	(% $\Delta k/k$)	(% $\Delta k/k$)	(MW-sec)	(°F)	(°F)	(°F)	(msec)	(°F)	(°F)	(°F)
1	10	10	8.3¢/in, 4g	1.3	30 msec	750	.35	0	--	2200	1075	210	68			
2	10	10	8.3¢/in, 4g	1.3	30 msec	620	.35	.15	--	1350	700	190	55			
3	10	10	8.3¢/in, 4g	1.3	STEP	1100	.45	0	--	2900	1475	225	50			
4	10	10	8.3¢/in, 4g	1.3	STEP	740	.3	.2	--	1400	760	195	35			
5	10	10	8.3¢/in, 4g	2.0	STEP	1940	1.1	.9	--	1750	975	190	28			
6	10	10	NONE	1.3	30 msec	780	.6	.5	--	1500	775	200	--			
7	100	100	8.3¢/in, 4g	1.3	30 msec	1400	.50	0	56	3600	1100	300	50			
8	100	100	8.3¢/in, 4g	1.3	30 msec	1320	.55	SMALL	44	2900	1000	290	48			
9	100	100	NONE	1.3	30 msec	1450	.6	SMALL	46	3000	1050	275	48			
10	100	100	NONE	1.3	30 msec in 30 msec out	800	.3	SMALL	26	1600	620	225	40			
11	100	100	8.3¢/in, 6g	1.0	STEP	660	.22	SMALL	32	2200	650	250	34			
12	100	100	8.3¢/in, 6g	1.05	30 msec	660	.23	SMALL	32	2000	650	240	50			
13	100	100	8.3¢/in, 6g	1.25	60 msec	700	.30	SMALL	34	2000	670	250	65			
14	100	100	8.3¢/in, 6g	1.3	30 msec	1250	.50	SMALL	44	2800	950	280	50	540	260	130
15	100	100	8.3¢/in, 6g	1.5	30 msec	1820	.70	SMALL	48	3200	1150	300	45	580	270	135
16	100	100	8.3¢/in, 6g	2.0	60 msec	2100	--	--	--	--	--	575	--	620	280	135
17	100	100	50¢/in, 4g	1.3	30 msec	1040	.40	SMALL	38	2500	825	250	46	--	--	--

- CONDITIONS:
1. END OF CYCLE
 2. MODE 1 OPERATION
 3. FIVE AXIAL REGIONS
 4. h AT H.S. = 0 FOR $q > q_{b.o.}$
 5. TEMPERATURES ABOVE 1200°F (MELTING POINT FOR ALUMINUM) MUST BE CORRECTED FOR HEAT OF FUSION AS FOLLOWS: $T(\text{ACTUAL}) = T(\text{CAL.}) - 750^\circ\text{F}$
IF $T(\text{CAL.}) > 1950^\circ\text{F}$; $T(\text{ACTUAL}) = 1200^\circ\text{F}$, IF $1200 \leq T(\text{CAL.}) \leq 1950^\circ\text{F}$
 6. 100% FUEL COOLANT VELOCITY = 50 fps

hot spot after it is film blanketed. Since the ignition temperature for the metal-water reaction under transient heating conditions is considered to be about 2550°F, no significant reaction would take place, unless autocatalytic effects resulted in the generation of larger amounts of nuclear energy. This latter consideration is discussed later.

"The main reactivity feedback mechanism for the full-flow case is fuel plate expansion, which results in water expulsion. Thus it is not expected that small increases in safety rod differential worth and rod acceleration would significantly decrease plate temperatures (see Cases 8, 14, and 17).^{*} However, to prevent a possibly large amount of core melting after the initial peak, it is necessary for the safety rods to cancel out most of the reactivity addition soon after (≈10 msec) the peak in power, so that shutdown will not depend on core disassembly. Our present analog model does not include any type of core disassembly and thus cannot in itself predict what will happen without safety rod action. Furthermore, the steam void feedback correlation is not applicable for bulk boiling of the coolant. Even so, calculations were made and some of the results are given in Table VI-B-5. As indicated by a comparison of Cases 8 and 9, the peak power was only slightly higher than achieved with safety rod action. The equilibrium power level achieved following the peak was 500 Mw and was achieved, following a dip to 100 Mw, about 70 msec after the peak in power. At full flow, 600 psi and 500 Mw, there would be very little, if any, bulk boiling even with good heat transfer to the water, assuming no flow instability. Of course at this power level all heat transfer surfaces would be film blanketed, causing melting of the fuel plates, at which time, if not before, the coolant pressure drop would sweep the molten metal out of the core, the temperature increase of the aluminum, assuming that it is still in contact with the fuel, would be about 450°F, which is not enough to initiate instantaneous ignition of the Al."

Therefore, on the basis of the foregoing discussion it is concluded that for the reactivity accidents considered above, there would be insignificant rapid additional energy release due to metal-water reactions. It should be noted that the ITC voiding accident for the AARR, as described in Section VI-B-3 above, is considered to be much less severe than the transients studied here, thereby providing additional conservatism in the conclusion stated above.

7. Safeguards Against Reactivity Accidents

Having examined the transient characteristics of the AARR for postulated reactivity additions, as described in Appendix C below and under specific accident situations described in previous paragraphs of this

^{*}Ed note: Case 17 was inadvertently omitted in both the HFIR and ANL tables. It is now included in Table VI-B-5.

Section, it is important to summarize those features of the AARR design which are intended to prevent accidental reactivity additions or to mitigate the consequences thereof. These safeguards are such as either (1) to eliminate credible, rapid reactivity additions of a magnitude having the potential for reactor damage or (2) to limit possible reactivity addition rates to those which can readily be suppressed by the reactor safety system acting with reasonable performance requirements. Design features of the AARR relevant to each of the foregoing are elaborated upon in some detail in the following paragraphs.

a. Safeguards on Reactivity Addition Magnitudes

(1) Control Element Malfunctions

The mechanical design of the shim-safety elements, drives and associated structures is such as to prevent the loss of reactivity control poison from the core in the unlikely event of plate or drive breakage. A mechanical stop is provided which limits the downward travel of a broken element such that a reactivity loss is incurred from any allowable initial critical positioning of the control elements.

(2) Cold Water Accidents

The estimated maximum reactivity addition due to a "worst case" cold water accident at 100 MW power level is $0.0072 \Delta k/k$. The reactor design will reduce this value due to flow mixing, proper valving of heat exchangers, and measurement and control of reactor inlet coolant temperature. Parallel coolant flow through all zones of the reactor tends to reduce the available reactivity gain.

(3) Voids in ITC Irradiation Facilities

The design of ITC target (rabbit) tubes for sample irradiation is such that complete voiding of any single tube will cause a reactivity addition much less than that required for fuel melting. Each tube will provide for independent cooling of samples such that loss of cooling capability within one tube will not cause a similar loss of capability in other tubes. Inlet and outlet flow passages for the bulk coolant within the ITC will be designed such that the occurrence of a massive flow blockage will be rendered incredible.

(4) Fuel Movement

The design of the core and supporting structures is such as to minimize inadvertent fuel motion in any direction which would tend to increase reactivity. Also, the location of reactivity control elements completely exterior to the fuel zone tends to reduce the possibility for safety plate jamming due to fuel displacement.

(5) Core and Control Element Separation

The reactor design is such as to minimize the possibility of separation of the reactivity control elements from the core in the remote

event of a massive pressure vessel rupture. To provide this safety feature, the control elements are bottom driven and the core support attachment to the pressure vessel is located below the vessel mid-height.

b. Safeguards on Reactivity Addition Rates

(1) Control Rod Malfunctions

The maximum reactivity addition rate due to simultaneous removal of all five control plates by runaway action of the shim and regulating control drives is limited to a safe value through appropriate limits set upon maximum drive speeds. In addition to limiting the automatic control drive rate, the maximum reactivity available for automatic control action is limited by a mechanical stop.

(2) Massive ITC Voiding

The reactivity gain possible due to the sweeping of voids, assumed to be present in the reactor bulk coolant, into the ITC will be controlled in rate by an appropriate choice of coolant flow rate for the bulk ITC water.

This value has been tentatively set at 4 ft/sec, for which it is estimated that the associated maximum reactivity addition rate available is low enough to be safely controllable by the reactor safety system. In addition, this flow rate is sufficiently high to ensure that void accumulation does not occur within the ITC.

(3) Irradiation Samples

The reactivity worths of irradiation samples will be determined prior to insertion into the reactor, to assure that the previously set maximum permissible worth will not be exceeded. The design of sample transfer systems will effectively limit the maximum rate of sample motion to a value commensurate with reactor safety.

C. Fuel Handling Accidents

1. Reactivity Considerations

It has been shown experimentally, at Oak Ridge National Laboratory, that when the beryllium reflector is replaced by water, the HFIR reactor core to be used in AARR is subcritical even in the complete absence of control rods. (See Reference 4.) In particular, each fuel element, fresh or spent, is grossly subcritical when fully immersed in water, without a beryllium reflector. This feature of subcriticality is utilized in the handling procedures of fuel elements in the AARR.

A description of the fuel handling system is given in Section IV-C. Criteria 25.1 through 25.11 (Section II above) will be governing in design to assure adequate subcriticality, shielding, and cooling of fuel in fuel handling and storage facilities.

Not more than one fuel element will be in transit at any given time, whether it is a spent element or a new element. All other fuel elements will be in controlled storage areas, or in place in the reactor pressure vessel. All fuel storage areas will be designed to ensure gross subcriticality even when fully loaded with fuel elements. (See Criterion 25.4.) No more than one pair of fuel elements, constituting an entire core, can be stored in any one storage cell. The storage cells will include fixed, solid neutron absorbers which will bound both the inner and outer radial dimensions of the core.* Cores in adjacent storage cells will be separated by at least 6 in. of water, which alone would reduce reactivity interaction to essentially zero.

If a reactor accident occurred which resulted in a disarray of the fuel elements, special measures might be required to ensure subcriticality of the damage fuel in the defective-core storage facility. The design of that facility will accommodate such emergency measures.

Fuel in storage will be covered to prevent damage from falling objects. (See Section IV-C-1.) The design of each HFIR fuel element includes solid circular extensions, above and below the active portion of the fuel zone. Thus, even if a fuel element were inadvertently placed on top of a core in storage, there would be a minimum distance of 1 ft from fuel to fuel, and hence negligible reactivity interaction.

Prior to unloading of spent fuel elements, one element at a time, from the reactor pressure vessel, all control plates will be inserted fully to assure shutdown while fuel is being moved. Prior to loading of new fuel elements, one element at a time, into the pressure vessel, all control plates will be inserted fully, with the possible exception of one or two plates which would be partially inserted and cocked. The desirability of cocking one or two control plates, rather than inserting all plates fully, will be evaluated in the safety studies for the Final Safety Analysis Report. Unloading and loading of fuel elements in the vessel will proceed

* Ed. note: Neutron absorber around the outside of the core was not discussed in Section IV-C above.

slowly and will be monitored with nuclear instrumentation to assure at all times that continued movement of the fuel element could not result in reactor criticality.

2. Cooling of Reactor Cores in Transit and in Storage

Following shutdown of the reactor for refueling, the primary coolant circulation will be continued until such time as the direction of flow through the core can be safely reversed and the core cooled by natural circulation without overheating. The duration of continued forced circulation cooling necessary before this can be accomplished will be based on HFIR experience. The two fuel elements constituting the core will then be removed from the reactor, one by one, and transferred, always under water, to the adjacent Spent Core Storage Pool. Natural convection transfer of decay heat from the core to the Reactor Pool and storage pool water will continue throughout the fuel element transfer operation. Adequate safety devices will be incorporated into the design of the core grappling and transfer mechanisms, so that the accidental dropping of a core and impairment of natural circulation cooling through the core coming to rest with the core coolant channels horizontal is not considered credible.

The arrangement of the cooling systems for the reactor and storage pool complex is such that the decay heat from a fuel element in transit can be adequately removed from any region of the complex for an indefinite period of time. Therefore, a malfunction of the transfer mechanism, rendering the fuel element temporarily immovable at some point in the transfer operation, would not result in core overheating. Since the element is accessible and is adequately shielded by water and the concrete pool walls throughout its entire path from reactor to storage pool, adequate protection and time would be available to permit special corrective action to be taken in such a case.

The quantities of water in the various sections of the complex are sufficient to ensure that only a very slow rise in water temperature would result from cooling loop failure, and there would be adequate time for detection of the failure and for taking the necessary steps to correct the difficulty. For example, the Spent Core Storage Pool (24-core storage capacity) and the transfer canal above it contain approximately 8,300 cu ft of water. The maximum heat generation rate in the pool occurs just after the 24th core has been transferred from the reactor (after a normal cycle) to the pool, and the latter already contains 23 previously used cores in various stages of decay. The rate of temperature rise in the pool water upon failure of the cooling system at this time would be $\sim 3.5^{\circ}\text{F}$ per hour. This initial maximum rate decreases rapidly as the 24th core decays; after 24 hours the rate of rise has dropped to $\sim 2^{\circ}\text{F}$ per hour, and the time required to raise the pool water temperature from its normal operating maximum of 100°F to the atmospheric boiling point would be about three days. All of these calculations assume no heat loss to the surroundings. Thus, there would be adequate time to bring in auxiliary portable cooling means or to institute a temporary bleed-and-feed procedure for removing excess pool heat. In any event, the reactor could be shut down and the cooling system for the Reactor Pool used to cool the spent fuel pool, until repairs had been effected.

The results of calculations of heat fluxes in the HFIR core at various times after shutdown, as well as consideration of the effects of accidentally placing a spent fuel element in a horizontal position, have been reported in the HFIR final safety documents⁴ as follows:

"When the vessel is open the reactor cannot be operated at any substantial power level, although it is possible to have some forced convection cooling under these conditions. Therefore there is little possibility of a meltdown due to loss of flow. The possibility of a power excursion exists; but, as pointed out in Appendix C,^{*} there is little likelihood that any credible reactivity addition could cause significant damage.

"One other safety problem encountered in the operation of the HFIR is that of protection against afterheat because of the high power density associated with the fission product decay following shutdown. It has been found⁵ that one second after shutdown following 15 days' operation at 100 Mw, the maximum fuel plate heat flux will be $\sim 42,000$ Btu/ft² hr. This falls to $\sim 6,200$ Btu/ft² hr one hour after shutdown... Immediately after shutdown, the fuel element,^{**} which is still generating in excess of 7 Mw of heat, is protected by continuing the forced convection cooling either by the use of the primary coolant pump motors or by the battery-driven pony motors,⁶ any one of which is sufficient. The fuel will be retained in the reactor under forced convection cooling for a period of about one hour after which it may be removed from the reactor vessel. At this time, the total power will be ~ 1.1 Mw. It has been experimentally determined⁷ that, when in an upright position and situated to provide a good free convection loop, the natural convection burnout heat flux will be in excess of 47,000 Btu/ft² hr. Thus, the fuel elements can easily be cooled after removal from the reactor following this delay.

"The only problem which arises in this connection relates to the temperature distribution to be expected in the fuel element should it be placed in a horizontal position, thus destroying the efficiency of the natural convection cooling. Experiments and analysis by Gambill⁸ indicated that under these conditions a heat flux of 1250 Btu/ft²-hr might result in a hot spot surface temperature as high as 1000°F. Consequently, the handling tools and the operating procedures are specifically designed to effectively prevent spent fuel from being oriented at any appreciable angle from the vertical."

The results of calculations presented in Reference (6) indicate that an AARR core would probably melt if kept on its side indefinitely.

* This refers to Appendix C of Reference (4).

** This refers to the entire core (two fuel elements).

Melting would occur slowly since the heating rate is low. For internal heat generation corresponding to a maximum heat flux of $1250 \text{ Btu/ft}^2\text{-hr}$ assuming complete insulation, the time required to add the heat of fusion is about 188 sec.

3. Loss of Pool and Canal Water

The entire pool complex will be lined throughout with stainless steel to minimize leakage. The various regions of the complex will be separable by water-tight gates, so that should a leak develop, the affected portion can be cleared of active materials, drained and repaired. Adequate means will be provided in the design to detect leaks in the pool cooling systems and to isolate any leaking components before pool water levels could drop significantly.

Inlet and outlet lines connecting the pools with their respective cooling systems will enter the pools near the top, and specific steps will be taken in the design to prevent syphoning through these lines. Drain lines will also enter near the tops of the pools, and will be stubbed off near the point of entry so as to require the attachment of a length of pipe to the stub, extending downward into the pool, before intentional draining.

In the case of the Reactor Pool, certain bottom and side wall pool penetrations are unavoidable. Particular care (such as double barriers) will be taken in the design of penetrations for the main reactor coolant pipes through the bottom, and for the beam tubes, rabbit sample transfer tubes and associated small lines through the side walls of the pool. Careful consideration will be given both to minimizing the possibility of leaks or ruptures resulting from normal operating conditions such as vibration and cyclic piping expansion, and to restricting the rate of water loss should such leaks or ruptures occur.

It is therefore postulated that the possibility of a major, rapid loss of water from the Reactor Pool will be extremely remote unless a major perturbation to the primary system, such as a major pipe rupture, occurs and causes a sudden, large amplitude displacement of the main coolant pipes at the pool penetration seals. Even this possibility will be rendered remote by designing the primary system piping supports to prevent excessive motion in the event of pipe rupture.

It should further be noted that the Spent Core Storage Pool is located at an elevation below the bottom of the gate between this pool and the Reactor Pool. Therefore an accidental draining of the Reactor Pool would not result in draining of the Spent Core Storage Pool and uncovering of the stored cores, even if the gate separating the pools were not in place or had been damaged in some manner. In this case the inventory of water in the Spent Core Storage Pool would be reduced to about 3500 cu ft, and the maximum initial rate of rise of water temperature (see Section VI-C-2) would be $\sim 8^\circ\text{F}$ per hour. The time necessary to reach the boiling point would be about 22 hours, assuming no heat loss to the surroundings, and an additional 20 days would be required for sufficient water to boil off to uncover the spent cores. ORNL-HFIR decay heat calculations⁴ and experimental data⁹ indicate that the decay heat from a spent core can be removed by

air-cooling 30 hours after shutdown, without fuel melting. It is therefore concluded that the accident described here cannot result in fuel melting and release of fission products from cores stored in the Spent Core Storage Pool.

In view of the AARR Spent Core Storage Pool design features of stainless steel lining, top entry of pool coolant and drain lines, and lack of communication with other pools (e.g., the Reactor Pool, which has bottom penetrations) below the level of the stored cores, it is almost inconceivable that an uncontrolled leak could develop in this pool to result in uncovering spent cores stored therein. However, the HFIR decay heat calculations⁴ and experimental data⁹ show that if this were to occur at any time significantly later than 30 hours after shutdown of the most recently operated core in the pool, no fuel melting and no fission product release would occur. (If the accident occurred just slightly more than 30 hours after shutdown, some melting might occur because of blockage of both water and air circulation during the time when the falling water level lies within the core.)

D. Equipment Malfunction and Primary Coolant System Rupture

Accidents discussed in this section are those involving system malfunction, failure of vital plant equipment, or rupture of the primary coolant system where reactivity insertions are not the prime concern.

1. Loss of Primary Coolant Flow

A complete or partial loss of primary coolant flow can occur as a result of pump failure (either for mechanical or electrical causes), inadvertent valve closure, or blockage either in the system or in the core. Loss of flow is detected rapidly by redundant flow instrumentation in the reactor outlet line. Flow signals are compared individually and continually with correspondingly redundant reactor flux signals. These comparators provide separate output trip signals at two different high values of the flux/flow ratio.

Normally the automatic flux control system functions to limit the flux to a value consistent with the existing flow. However, if the loss of flow is great enough, or occurs rapidly enough, the flux control system cannot cope with it. When the flux/flow ratio rises to the first trip setting the shim-safety rods are driven in rapidly and automatically until the flux/flow ratio falls below the trip setting. If for any reason the ratio increases to the second trip setting, the reactor is automatically scrammed. Extreme low flow (below that provided by a single primary coolant pump operating on its pony motor) will cause automatic scram directly.

a. Primary Coolant Pump Failure

Two one-half capacity primary pumps provide normal coolant flow to the reactor. The use of two such pumps rather than a single, full-capacity pump reduces the probability of a complete loss of coolant flow by pump failure. In addition to the normal a-c drive motors, each pump is equipped with a d-c pony motor coupled directly to the pump shaft and supplied with Class-I uninterrupted power. The pony motors are sized such that either pump driven by its pony motor is capable of providing at least the total flow needed to remove decay heat from the core following reactor shutdown. Both pony motors are energized at all times when the reactor is operating.

(1) Simultaneous Loss of Both Main Primary Coolant Pump Motors

An analog computer study, based upon the Title-I design, has been carried out to determine the flow coastdown characteristics of the AARR primary coolant loop in the event of simultaneous loss of utility power to both main coolant pumps. Computations were made for the normal operational case where the pony motors on both pumps are in operation at the time of the utility power failure and continue to operate thereafter, as well as for abnormal cases where one or both pony motors either are not functioning at the time of the utility power failure, or fail simultaneously with the utility power. The results of these calculations are presented in Table VI-D-1.

The following conclusions have been drawn from the flow coastdown study:

TABLE VI-D-1

PRIMARY SYSTEM FLOW COASTDOWN
(LOSS OF BOTH MAIN COOLANT PUMP DRIVE MOTORS)

Percent of Rated Flow	Time for Coastdown to Indicated Flow (sec)		
	Two Pony Motors Operating	One Pony Motor Operating	No Pony Motors Operating
100	0	0	0
50	1.8	1.8	1.8
30	4.0	4.0	3.8
20	7.0	6.7	6.2
15	12.	10.	9.0
13.2	20. (a)	-	-
10	-	25. (a)	14.
5	-	-	22.
0	-	-	32.

(a) Flow maintained at indicated flow percentage by pony motor(s)
after this time.

(a) The important parameters determining rate of flow coastdown upon loss of power are rated motor torque and speed, and total rotary inertia. Over 90% of the shaft inertia is that of the motor. Water inertia is relatively unimportant.

(b) Flow coastdown in AARR should not be any faster than 1.8 sec to reduce flow to 50% of rated flow.

(c) Rate of coastdown is not affected by operation of pony motors during the initial stages of the transient (i.e., up to ~ 3 sec). Later, however, the difference becomes significant, since flow levels off at $\sim 13\%$ with two pony motors operating, at $\sim 10\%$ with one pony motor operating, and goes to zero without pony motors.

Experimental measurements of coastdown were made in the CP-5 reactor and in a general purpose hydraulic test loop at Argonne. In all cases, good agreement was obtained between the measurements and analog computations using the same coastdown model as used to determine the anticipated AARR coastdown rates given in Table VI-D-1.

Analog studies of the variation in incipient boiling heat flux in the hot channel (ϕ_B), as compared with the actual maximum heat flux (ϕ_{max}), during an AARR loss of flow accident are in progress, but have not yet been completed. However, the quantity (ϕ_B/ϕ_{max}) has been obtained vs time in HFIR, both in analog computer studies and experimentally with the reactor operating initially at 100% power (100 MWt) and flow. It has been demonstrated, both analytically and experimentally, that reactor scram was not required, and that setback and reverse control action were sufficient to avoid unacceptable values of (ϕ_B/ϕ_{max}) following a loss of flow accident. Essentially the same control system and parameters are planned for AARR as are in use in HFIR. The HFIR analog studies have shown that core temperature excursions are mild, even at end of core life when the control rod differential worth is decreased and the formation of Al_2O_3 scale on fuel plates has increased fuel temperatures significantly.

(2) Loss of Single Main Primary Coolant Pump Motor

With loss of a single main primary coolant pump, it has been estimated that the primary flow (assuming zero flow in the down pump loop) would be reduced to about 80% of the normal flow with both pumps operating. This would enable the reactor to be operated at approximately a corresponding power level, provided no further decrease in flow occurred. The automatic flux control system (setback and reverse) would effect the power reduction automatically, and would then proceed to control at the new level. No damage to the reactor would occur.

(3) Sequential Loss of Main Primary Coolant Pump Motors

In a sequential loss of primary coolant pumps, wherein the loss of one pump is followed by loss of the second pump after an interval of time, it is estimated that even if the second pump motor were to fail during the transient caused by the failure of the first, the consequences to the core would not be as severe as for the simultaneous loss of both

pumps, even if the loss of the second pump were to occur at the most pessimistic moment during the transient. Further analog studies of such accidents, including the effects of the automatic flux controller, the trip settings and time interval between tripouts, will be conducted during the Title-II design.

(4) Loss of Primary Coolant Pump Pony Motors

As described earlier, the pony motors are always energized during reactor operation and hence are always available to drive the main pumps at a reduced flow. Each pony motor has its own storage battery bank and its own battery charger;* each battery charger has the capacity to drive the pony motor at rated speed and simultaneously to charge its battery tank.

The state of each pony motor is monitored to the maximum extent possible during reactor operation with the main pump motors operating. In the unlikely event that a pony motor should be found inoperative, the reactor power will be reduced while flow is maintained by the main pump motors (with the second pony motor also operating). If the faulty pony motor cannot be restored quickly, the reactor may be shut down manually, by administrative decision. In the highly unlikely event that both pony motors should be lost, reactor power will automatically be set back to 10%, and if pony motor operation cannot be restored quickly, the reactor may be shut down manually.

Failure of a pony motor simultaneously with or shortly after loss of both main coolant pumps (or while the plant is operating at low power on the pony motors only, with the main motors shut down) will result in an immediate automatic reactor scram. Since each pony motor is sized to provide adequate flow for removal of all decay heat following shutdown, in such an event the remaining pony motor would cool the core sufficiently to prevent core damage.

Because of the extent of redundancy provided in the pony motors and their independent power supplies (see Section IV-I above), it is considered incredible that both pony motors could fail simultaneously with or shortly after loss of both main pump motors. This situation is analogous to total failure of all utility and emergency power (also considered incredible), the consequences of which are discussed in Section VIII-E below.

b. Inadvertent Valve Closure

The inlet and outlet lines for each primary pump are equipped, respectively, with a butterfly valve and a stop-check valve. The butterfly valves are manually controlled locally, and are provided solely to permit repairs and maintenance on the pumps, which are situated at a low point in

* Ed. note: This wording may be somewhat misleading. Each pony motor is powered from a different d-c power supply, but other critical electrical loads may be powered from the same supplies.

the primary coolant loop, without draining the entire system. The valves are padlocked open during normal operation.

The stop-check valves on the pump outlet lines are provided (1) so that the main coolant pumps may be started against a closed valve to limit starting current, (2) to permit isolation of the pumps for repair without draining the entire primary system, and (3) to prevent reverse flow in the pumps during switching operations and on the rare occasion when only one pump is operating. For the first of these functions, the valves are equipped with motor operators operated manually from the control room. The valves and their operators are designed to fail in position, which under all normal modes of operation is open. The electrical circuits for the motor operators are independent, and are routed to minimize the possibility of a single malfunction or accident resulting in driving both valves closed simultaneously. Accidental closure of one of the valves would produce a situation analogous to failure of a single main coolant pump. As indicated above, this condition is handled automatically by the reactor control system without damage to the reactor.

There are no other valves in the main primary coolant loop except those bordering the system. Therefore there is no single valve in the primary loop whose closure would result in a complete interruption of primary flow.

c. Core Flow Blockage

The AARR primary coolant system will employ a full-flow filter, located near the inlet to the reactor pressure vessel, which will minimize the probability of foreign materials reaching core coolant channels. Nevertheless, a core flow blockage must be considered credible and the consequences thereof examined.

The flow blockage accident for the AARR core is analyzed for the maximum excess energy which could conceivably be released. All instrumented shutdown and/or power setback variables, such as decreased coolant flow and high fission product activity signals, are assumed to be ineffective and reactor shutdown is accomplished only through inherent reactivity compensation. These assumptions are made, not to cast doubt upon the reliability of operation of the reactor protective systems, but rather to provide conservatism in the calculated results.

The following conditions are considered to describe the sequence of events following flow blockage within a local area of the AARR core:

(1) Forced convection cooling is suddenly terminated within the affected core zone.

(2) The reactor power level is held at 100 MW by automatic movement of the regulating control cylinder until it reaches its limit of control. This control system is assumed to have available $\pm 0.007 \Delta k/k$ rapid reactivity control at the time of initiation of the accident.

(3) Immediately following termination of flow within the core

zone initially affected, metal and associated water temperatures rise rapidly. The metal temperature achieves the minimum value of 2552°F (1400°C) necessary for aluminum-water ignition under rapid heating conditions (see Appendix B below). The associated aluminum-water reaction is complete, releasing the full chemical energy potential of the metal involved.

(4) Partial disassembly of the ignited metal occurs causing a local increase in core water-metal ratio. However, recognizing the unusual strength of the U_3O_8 -Al cermet at high temperatures as exhibited in TREAT tests (see Appendix B below), it is assumed that the bulk of the fuel plate meat is retained within the core.

(5) The above process propagates from the point of inception due to flux peaking and fuel plate distortion. This initially produces a reactivity gain with an estimated maximum value of 0.0003 $\Delta k/k$ for an optimum local removal of metal accompanied by replacement with water; further local loss of metal causes a reactivity loss.¹¹

(6) Subsequent reactivity loss and eventual reactor shutdown results only from heating of the water within the core zone subjected to flow blockage. It is assumed that this water reaches the saturation temperature appropriate for average pressure in the core, but no credit is taken for reactivity loss due to steam formation which would surely occur. Further, the additional negative reactivity contribution due to fuel plate expansion is neglected in the calculations.

(7) In addition to the assumed positive reactivity contribution from the automatic control system and from creation of an optimum local flux trap, the inherent shutdown mechanism must compensate for a portion of the reactivity temperature deficit of the core which is gained upon cooling. The total reactivity loss resulting from the reactor changing from 70°F isothermal conditions to temperatures corresponding to 100 MW power level is 0.004 $\Delta k/k$. Assuming the fuel zone contributes this full reactivity effect for a change in reactor operation from 120°F isothermal conditions to full power level, the fraction of this which must be compensated to effect reactor shutdown depends upon the fraction of the core unaffected by the flow blockage and hence capable of being cooled under normal flow conditions.

Using the foregoing assumptions, the maximum fraction of the core experiencing flow blockage is determined by satisfying the condition of zero reactivity at reactor shutdown. The minimum magnitude of the fuel zone water temperature coefficient of reactivity occurs at the end of core life, at which time its value is approximately $-5 \times 10^{-5} \Delta k/k/^\circ F$ at 155°F.

The normal average water temperature in the core at 100 MW power level is approximately 150°F and the average pressure level is approximately 550 psi. Saturation temperature at this pressure is 477°F and hence in rising from average to saturation temperature the coolant undergoes a temperature change of 327°F. For a uniform temperature change of this magnitude throughout the total core coolant volume, a reactivity change of approximately $-0.0164 \Delta k/k$ would result. Letting α represent the fraction of the core volume with blocked flow, wherein the coolant content is saturated water, the associated reactivity loss is -0.0164α . The reactivity gain

from cooling the unaffected core zone is $0.004 (1-\alpha)$. The overall reactivity balance required at shutdown, as a function of α , is written as follows:

$$0 = 0.003 + 0.007 (1-\alpha)(0.004) - \alpha (0.0164)$$

from which $\alpha \approx 0.55$.

Thus it is found that flow blockage over approximately one-half of the total core volume is required to yield zero reactivity at reactor shutdown.

Returning now to the assumption of complete chemical reaction of the aluminum within the core zone experiencing blocked coolant flow, approximately 4×10^4 grams of aluminum will react. The total chemical energy release is then approximately 720 MW-sec. The reaction rate and corresponding energy release rate is assumed to follow the large particle rate law cited in Appendix B below, viz.,

$$\text{Rate} = 0.96 \frac{\text{gm-Al}}{\text{cm}^2\text{-min}}.$$

Since extensive fuel plate disassembly is not assumed in this analysis, the rate calculation is based upon the original heat transfer area of the fuel within the affected zone ($2.22 \times 10^5 \text{ cm}^2$). The rate of aluminum consumption is then $3.55 \times 10^3 \text{ gm Al/sec}$. Thus the chemical reaction of 4×10^4 grams of Al requires approximately 11.3 seconds for completion, and will not produce damaging pressure loads upon the reactor vessel or other primary system components.

The hydrogen released as a result of the aluminum-water reaction amounts to approximately 90 ft^3 at reactor pressure and temperature conditions. Although the hydrogen constitutes an additional potential energy source, the maintenance of primary system integrity and absence of oxygen eliminate consideration of a significant energy contribution from this source. It is considered that the evolved hydrogen will be subsequently removed through the primary system degasification system.

The preceding analysis has been presented to demonstrate the maximum conceivable consequences of a flow blockage accident in the AARR core. The assumed conditions and sequence of events used in the study are considered to be extremely conservative particularly with respect to the assumed method of reactor shutdown and the amount of chemical energy released. It is believed to be most unlikely that none of the protective signals would cause automatic shutdown of the reactor in time to limit the extent of core meltdown to a value much less than that calculated above. Further, ignoring the self-shutdown effects of metal expansion, steam formation and gross fuel loss from the core constitutes an extremely pessimistic approach to evaluation of flow blockage effects. Nevertheless, in spite of the conservatism employed in the analysis, the results show that although extensive core damage occurs, the associated energy release is well within the containment potential of the AARR reactor vessel and no undue safety hazard is incurred (see Appendix E below).

d. Loss of Reflector Cooling

Abnormal conditions such as partial flow blockage and flow coast down in the beryllium reflector have been reviewed on a preliminary basis.

The design of the reflector for AARR is identical to HFIR with the exception of the outside diameter and the number of beam tube and irradiation facility penetrations. Oak Ridge National Laboratory has demonstrated, by tests in HFIR and by observing the response of HFIR to a power failure, that flow coastdown will not cause any damage to the reactor.

Partial flow blockage due to a foreign object or material is always a possibility in a reactor. The coolant flow for the beryllium reflector is a stream of primary coolant in parallel with the core coolant. The water first enters a common reflector and control plate coolant plenum through eight flow orifices. Part of this stream is then distributed to beryllium regions through four inlet strainers. The design and arrangement of the plenum, the eight inlet orifices and the four reflector inlet strainers are such as to render incredible the complete, simultaneous blockage of more than one inlet orifice or strainer.*

The normal flow rate to the beryllium is approximately 1085 gpm, and the maximum interior temperature in the beryllium at 100 MW and normal flow conditions is 290°F. If one of the eight inlet flow orifices were to become blocked, a conservative assumption would be that the flow to the reflector would be reduced to ~1000 gpm. This reduction in flow would cause the maximum reflector temperature to increase to approximately 315°F. The reduction in structural strength and corrosion resistance corresponding to such an increase in temperature is very small and can be neglected.

e. Flow Reversal Considerations

Flow reversal in the HFIR core, as a result of an accident or normal shutdown procedure, involves a situation wherein the coolant within the flow channels is momentarily stopped. Depending on the power level at which the reversal is initiated, a temperature rise of the surface above the nucleate boiling temperature might or might not occur.

In situations where overshoot of the surface temperature occurs but burnout does not occur, the persistence of the temperature is of the order of ~10 sec based on test data. Chugging (alternate flow into and out of the channel) occurs, and it appears from the data that the heat transfer phenomena limitation on allowable power level is related to the chugging process. The channel acts during the chugging process as a "pump," inasmuch as the heating of the fluid causes steam formation that expels the fluid. Fluid then re-enters the channel and the expulsion is repeated after further heat is absorbed.

In experiments using a 0.040 in. rectangular channel with 18 in.

* In fact, the complete blockage of even one inlet orifice borders on incredibility.

heated length and a circumferential maximum/average heat flux ratio of 2.2, departure from nucleate boiling began at a power level corresponding to an average heat flux of 1.80×10^5 Btu/hr-ft², with an initial inlet coolant temperature of 135°F and a system pressure of 750 psig. At low pressure (13 psig) and the same inlet temperature as above, natural circulation instability occurred at an average heat flux of 3.5×10^4 Btu/hr-ft², and increased in severity as the available power limit corresponding to 7.0×10^4 Btu/hr-ft² was approached.

The effect of channel space is important to the design margins because the allowable power level decreases with reduced channel space. Burnout design margins for the core, which is composed of involute plates, will therefore differ from those for the reflector, which has well defined channel spaces formed by holes in large solid pieces.

For the core, the final operating burnout margins will be selected conservatively with regard to the analysis of flow reversal during fuel handling and non-core-damage accidents involving reverse flow in the vessel. The establishment of procedures for cooling of the reflector during fuel handling will likewise involve proper consideration of test data, and temperature limits of the reflector.

2. Loss of Primary Coolant

A loss of primary coolant may result from either a leak in, or a rupture of, the primary coolant system, through which primary coolant is discharged to the Reactor Pool, to the containment, or into the secondary coolant system. The magnitude of the hazard resulting from the primary coolant loss will depend upon the size and location of the leak or rupture, and may range all the way from nuisance leakage requiring little or no curtailment of plant operation to a major loss leading to the Maximum Credible Accident.

a. Leakage

Leakage of primary water less than, or equal to the maximum make-up rate of 50 gpm to the make-up storage tank, which supplies the primary pressurization pumps, can be sustained indefinitely. Leakage rates exceeding this will result in a decrease in the make-up storage tank water inventory (normally 3000 gal), and can be endured only for a limited time. A leakage rate up to approximately 200 gpm (or 400 gpm if both pressurization pumps are operating) can be sustained for approximately 20 min (or ~9 min if both pumps are running) without loss of primary system pressure. At higher rates than these, or after the indicated time intervals, loss of system pressure will occur.

Leakage rates in excess of 50 gpm are detected by monitoring make-up storage tank water level. A drop in level to a preset point will actuate an alarm in the control room. Radiation monitors in the Reactor Containment Building, the stack system, and the secondary coolant loop will serve as alarm back-up. If the source of water loss cannot be quickly located and corrected, the level will continue to drop and will eventually render the pressurization pumps inoperative. At this point system pressure

will fall, and action as described in Section VI-D-3 below will occur. The first low pressure trip described therein will correct the situation if the source of leakage is in the degasification and purification system, and not in the main primary loop. However, if the leakage is indeed in the primary loop, pressure will fall to the second low pressure trip point, at which reactor scram occurs.

It should be noted that the radiation hazard associated with any leak will be a factor dictating subsequent action. For example, even if the leakage is too small to affect operation of the primary system, the resulting radiation hazard might dictate a manual shutdown of the plant.

b. Primary System Rupture

There are two principal factors to be evaluated in analyzing the effects of ruptures in the AARR primary system. The size and degree of completeness of the rupture will determine whether or not flashing of primary coolant occurs within the core (with possible core melting), whereas the location of the rupture will determine the extent of water loss, the ultimate disposition of the water, and the route by which fission products, if released from the core by melting thereof, will pass into the Reactor Containment Building atmosphere. As pointed out in Section VII-B-2 below, the latter has an important effect on the subsequent overpressure history in the Containment Building.

As described in Section VI-D-3 below, two primary system low pressure trips are provided, the first of which sounds an alarm and causes certain corrective actions to occur, and the second of which causes automatic reactor scram. The corrective actions mentioned provide capability to add water to the primary loop at a rate of ~ 400 gpm for a period of ~ 9 min, at which point the make-up storage tank supplying the pressurization pumps will run dry and the addition of water will cease. For relatively small ruptures, therefore, the rate of pressure loss will be retarded by the addition of make-up water. Indeed, for ruptures equivalent to a 3/4-in. diameter opening in the system or smaller, no depressurization will occur as long as the make-up water supply is available. However, for ruptures in large pipes it is clear that make-up addition will have an insignificant effect on the events which follow.

For a major rupture, the primary system will depressurize very rapidly, and automatic reactor scram will be initiated during the depressurization by the second low pressure trip. As system pressure approaches atmospheric (assuming the rupture to be outside the Reactor Pool), some flashing of coolant in the hot channels will occur, and there will be some generation of steam films on the fuel plate surfaces at hot spots. (The water temperature at the hot channel exits is estimated to be 246°F , and the fuel plate oxide-water interface temperature at the hot spots is estimated to be 377°F , at 100 MW.) Fuel plate temperatures will rise, because the decrease in heat removal rate resulting from steam blanketing is greater than the decrease in heat generation rate with decreasing power level, as the control plates shut the core down. Some melting of the fuel plates at the hot spots may occur during the first few seconds after the pressure loss, depending upon the heat transfer mechanisms which are effective during this period. However, such melting, if it occurs at all, cannot occur extensively

enough in the time available to obstruct control plate passages and prevent reactor scram.

This assertion is substantiated in an analysis made by Oak Ridge National Laboratory in their initial response¹² to AEC questions on the HFIR Final Safety Analysis Report. This analysis is based on an initial step-change in pressure from normal operating pressure (600 psig) to atmospheric pressure. Reactor scram is assumed, with rod insertion beginning 50 msec after the pressure loss. All fuel surfaces are assumed to be blanketed with a 1.5 mil thick layer of steam with the remainder of the coolant passage volume remaining filled with water. Heat transfer occurs by conduction through the steam film, and is assumed to be proportional to the temperature difference between the fuel plates and the water; the latter is taken as that which prevailed prior to the pressure loss, and is assumed to remain constant throughout the few seconds covered by the analysis. This implies continuation of some coolant flow, at least during this short period. (This in fact would be the case, since there would be some flow coastdown, although perhaps quite brief, even in the case of flow interruption by a main coolant pipe rupture.)

For convenience, the HFIR analysis is reproduced below:

"A quantitative analysis of a sudden or step decrease in pressure is difficult because of the virtual impossibility of predicting the exact sequence of events in the core following such an event. The number and magnitude of local hot spots and the physical behavior of the fuel in these regions could have a significant effect on the heat removal characteristics of the rest of the core. For example, if the fuel melted and dripped out of the fuel plates, it might rearrange itself into a configuration better suited for removing heat. On the other hand, if several plates melted or buckled and came in close contact with one another, the heat transfer mechanism could be inhibited to some extent.

"Nevertheless, it is possible to take a semi-quantitative look at the overall situation from an integral point of view and to draw some conclusions based upon this examination.

"The model utilized for this is as follows. It will be assumed that with the reactor operating at 100 Mw (9.48×10^4 Btu/sec), a step decrease in pressure occurs in which the pressure is instantly decreased from forty atmospheres to one atmosphere. It is further assumed that, following this pressure reduction, the average rate of heat removal will be proportional to the difference between the fuel temperature and the temperature of the surroundings. The latter is assumed to remain unchanged

during the pressure loss. In developing this model, the following nomenclature is used:

$G(t)$	=	total heat generation rate (Btu/sec)
$T(t)$	=	fuel temperature - temperature of surroundings ($^{\circ}\text{F}$)
T_0	=	initial value of $T(t)$ ($^{\circ}\text{F}$)
T_w	=	temperature of surroundings ($^{\circ}\text{F}$)
$T_m(t)$	=	fuel temperature ($^{\circ}\text{F}$)
C	=	specific heat of fuel (Btu/lb $^{\circ}\text{F}$)
M	=	total fuel weight (lb)
S	=	heat transfer area (ft^2)
t	=	time (sec)
K	=	heat removal constant (Btu/ $^{\circ}\text{F}$ sec)
h	=	heat transfer coefficient (Btu/ ft^2 hr $^{\circ}\text{F}$)
κ	=	conductivity of steam (Btu/ft hr $^{\circ}\text{F}$)

"The change in the value of $T(t)$ with time may then be expressed as

$$MCT(t) = G(t) - KT(t), T(0) = T_0$$

whence

$$T(t) = e^{-\mu t} \left[T_0 + \int_0^t \frac{e^{\mu \tau} G(\tau)}{MC} d\tau \right]$$

so that

$$T_m(t) = T_w + e^{-\mu t} \left[T_0 + \int_0^t \frac{e^{\mu \tau} G(\tau)}{MC} d\tau \right]$$

where here

$$\mu \equiv K/MC.$$

"The total heat generation function $G(t)$ is obtained from analog studies of the system response following a loss of pressure. For convenience it is represented as a "stairstep" function having the form:

$G(t)$	=	9.48×10^4 Btu/sec	$0 < t < 0.05$ sec
$G(t)$	=	6.21×10^4 Btu/sec	$0.05 < t < 0.15$ sec
$G(t)$	=	2.51×10^4 Btu/sec	$0.15 < t < 0.25$ sec
$G(t)$	=	1.42×10^4 Btu/sec	$0.25 < t < 0.35$ sec
$G(t)$	=	9.48×10^3 Btu/sec	$t > 0.35$ sec

(Actually $G(t)$ decreases proportional to $t^{-0.2}$ after a few seconds but this is neglected here.)

"Also, $M = 150$ lb, $S = 430$ ft^2 , $C = 0.25$ Btu/lb $^{\circ}\text{F}$. Now K is an overall heat removal constant such that $K = hS/3600$. The appropriate value of h is obtained by assuming that a film of steam 1.5 mils thick blankets the fuel. The conductivity, κ , of steam

increases almost linearly with temperature from 0.014 Btu/ft hr °F at 200°F to 0.051 Btu/ft hr °F at 1600°F. We have $h = 12\kappa/0.0015$ so that $K = 955.5\kappa$ Btu/°F sec, $\mu = 25.5\kappa$ sec⁻¹. In the temperature equation both K and μ are taken to be constants; however, they are adjusted to appropriate average values for each of the temperature regimes.

"Values of the average core temperature were found by using the average water temperature $T_w = 150^\circ\text{F}$, and the initial fuel temperature $T_{m(o)} = 202^\circ\text{F}$. These are plotted in Fig. VI-D-1. The equilibrium temperature of 648°F is well below the melting point.

"The hot spot fuel temperature was also estimated using a water temperature of 461°F and an initial metal temperature of 480°F. The power density may be as high as 2.5 times the average and the values of $G(t)$ were adjusted accordingly. Under these assumptions, the hot spots would, as shown in Fig. VI-D-1, reach a temperature of 1210°F which is very close to the melting point. An additional 167 Btu/lb would be necessary to supply the required heat of fusion. Thus a complete reaction of the U_3O_8 fuel with aluminum, which could supply adiabatically a maximum of 109 Btu/lb of aluminum, would not be sufficient to cause melting. Hence, there seems to be little possibility of a metal-water reaction which requires an ignition temperature of 2140°F.

"It must be recognized that the validity of the heat removal assumptions used above are open to question and that various physical occurrences in the core may alter them either for the better or for the worse. It is believed, however, that the temperatures deduced from this model are fairly realistic. This is reinforced by the fact that implicit in the hot spot model is the assumption that only about 25% of the nuclear heat is removed during the transient immediately following the pressure loss, and that only 15% is removed in the average case. It seems reasonable to believe that at the temperatures involved, there is sufficient thermal driving force to accomplish this."

An interval of 50 msec is assumed between the instant of pressure loss and the generation of a scram signal. Recent tests of the time interval from scram signal to completion of control plate insertion, performed in actual HFIR operation, show that times of 275 to 315 msec are achieved, 280 msec being the normal value in most cases. It is therefore reasonable to assume that plate insertion would be completed 350 msec after the pressure loss.

FIG. VI-D-1
VARIATION OF REACTOR POWER AND
CORE TEMPERATURE WITH TIME
FOLLOWING A STEP LOSS IN PRESSURE

SOURCE: ORNL DWG. 65-12251 (10/12/65)

The HFIR analysis above indicates that under the assumed conditions, fuel plate hot spot temperatures might just reach the fuel melting point some 325 msec after the pressure loss, and certainly no significant melting would occur before completion of control plate insertion.

Because there is some question as to the heat transfer mechanisms which would be functioning under such an accident condition, two other analyses were performed to verify the conclusion drawn. In the first of these, the same assumptions as above were made with regard to pressure loss and scram, but no credit was taken for any heat removal from the fuel plates. It was further assumed that the hot spots in the core constitute 5% of the core mass, and that heat is generated in the hot spot regions, during both power operation and shutdown, at 2.5 times the average rate for the entire core at any instant. It was estimated that it would require $\sim 12\frac{1}{2}$ MW-sec of heat generation in the entire core to raise the hot spots to melting temperature (~ 170 msec), and 11 MW-sec additional would be required (in the entire core) to melt the hot spots (~ 230 msec additional, with falling reactor power during scram). Since control plate insertion would begin 50 msec after pressure loss (representing only 5 MW-sec of heat generation at 100 MW), it can be concluded that the plates would already have achieved a substantial fraction of their insertion (probably about one-third) before the hot spots reached melting temperature, and that plate insertion would be complete before the hot spots were entirely melted.

It is judged that substantially more of the core than simply the $\sim 5\%$ represented by the hot spots would have to melt before the heavy outer cylinder of the outer fuel element could be distorted sufficiently to distort in turn the inner shim-regulating cylinder and cause it to block motion of the outer shim-safety plates which cause scram. On the same basis as above, a total of ~ 42 MW-sec of heat absorption would be required to raise all of the fuel in the core to the melting point, and an additional ~ 14 MW-sec would be required to melt one-half the fuel. These events would not occur until ~ 1 sec and $\sim 1\frac{1}{2}$ sec* after pressure loss, respectively, considerably later than completion of control plate insertion.

In the second check analysis, no credit was taken for the reduction in power level from 100 MW as the control plates are inserted. Even in this case, the hot spots do not reach melting temperature until ~ 125 msec after pressure loss, and melting of the hot spots would not be complete until ~ 237 msec had passed. Raising the entire core to melting temperature would require ~ 420 msec, and melting of one-half the core would require ~ 140 msec additional. Here again it can be concluded that control plate insertion would be accomplished prior to any obstruction of plate motion due to fuel melting.

Events following the initial loss of pressure and reactor scram in various primary system rupture cases are analyzed below, according to the location of the rupture as designated in Figure VI-D-2. Rupture locations considered are (1) inlet line inside the Reactor Pool, (2) outlet

* These values are based on a 3-dollar shutdown margin for decay of fission power.

FIG. VI-D-2
 DIAGRAM OF PRIMARY SYSTEM FOR RUPTURE ANALYSIS

line inside the Reactor Pool, (3) inlet line outside the Reactor Pool (and therefore in the Reactor Containment Building service floor), and (4) outlet line outside the Reactor Pool. In all cases, "inlet line" refers to that portion of the main primary coolant loop piping between the main pump discharge and the reactor vessel inlet nozzle, and "outlet line" refers to the portion between the reactor vessel outlet nozzle and the main pump suction.

Case 1. Ruptured Inlet Line Inside the Reactor Pool

It is assumed that a complete pipe severance occurs and that continued pump operation causes primary system water to be discharged to the pool and pool water to be drawn into the reactor vessel. The coolant flow rate is reduced because of the change in pump characteristics under these conditions. The pump behavior curve shown in Fig. VI-D-3 is an estimate of flow rate versus available net positive suction head. The solid portion of the curve depicts the range of flow rates for which no appreciable cavitation occurs. At lower flow rates, indicated by the dashed portion of the curve, cavitation may become significant, and therefore the predicted pump behavior is less certain.

Under the conditions of the assumed rupture, the coolant flow rate through the primary system was estimated using the data of Fig. VI-D-3, assuming that system pressure drops are proportional to the square of the flow rate, and noting that a value of core inlet pressure of approximately 28 psia is maintained by atmospheric pressure plus the head of 30 ft of water measured from the pool free surface to the core inlet. Using various assumed values of flow rate, an iterative process was employed until the assumed value agreed with the data of Fig. VI-D-3. The results of these calculations indicate that the flow rate drops from the design value of 11.6×10^6 lb/hr-ft² to approximately 5.2×10^6 lb/hr-ft². The resulting core exit pressure is then approximately 10 psia.

The state of the reactor core under these conditions can be evaluated by reference to Fig. VI-D-4, which is based on HFIR flux plots and the burnout correlation of ANL-6675;¹³ the figure depicts burnout heat flux versus flow at various core exit pressures. The initial inlet temperature (T_0) of the water to the core would be equal to the pool temperature ($\sim 75^\circ\text{F}$). Thereafter the pool temperature would rise to a value less than 120°F , depending on the secondary coolant system and outside weather conditions. At a core exit pressure of 10 psia and a flow rate of 5.2×10^6 lbs/ft² hr, the burnout heat flux determined from Fig. VI-D-4 corresponds to the maximum value of heat flux which would occur at ~ 68 MW power level. This is approximately 10 times the maximum power level existing in the shut down core, and it is therefore concluded that no core melt would occur due to the assumed rupture.

Case 2. Ruptured Outlet Line Inside the Reactor Pool

Similar to the foregoing case studied, it is assumed that a complete pipe severance occurs and that pump operation causes continued primary system flow at a reduced rate. Following the calculational procedure outlined for the previous case and accounting for the differences due to the assumed location of the rupture, it is found that a flow rate of approximately 9.4×10^6 lb/hr-ft² would be maintained. The core inlet pressure

FIG. VI-D-3
NET POSITIVE SUCTION HEAD VS CORE MASS FLOW RATE

FIG. VI-D-4
PERMISSIBLE CORE HEAT FLUX VS MASS VELOCITY
(INLET TEMPERATURE 75°F)

would be approximately 90 psia and the exit pressure slightly below 30 psia.

Referring to the burnout data presented in Fig. VI-D-5, applicable to this case, it is concluded that the AARR core could operate marginally under these conditions at a power level of 100 MW. Thus there exists an even greater margin against burnout of the shut down core than that predicted for the previous case. It is again concluded that no core melting would occur.

For the two cases of primary system rupture described in preceding paragraphs, the assumed break locations are such that there would be no gross net loss of primary coolant, since the Reactor Pool would serve to contain the system. Although depressurization would occur, coolant flow rates would be maintained at a level sufficiently high to ensure adequate cooling of the shut down core indefinitely, and it is concluded that no fission product release would occur. However, the release of N^{16} from the primary system to the Reactor Pool would probably result in tripping the Containment Building isolation system to seal the building. A small rise in pool water temperature might occur as the pool and primary system approach an equilibrium temperature, but the resulting containment overpressure would be insignificant.

In the latter two cases of primary piping rupture under consideration herein, the assumed locations of the ruptures, i.e., outside the Reactor Pool, lead to subsequent accident conditions and consequences thereof quite different from the former. In these cases it is apparent that a substantial loss of primary system coolant would occur and that the coolant lost from the system would be discharged to the Reactor Containment Building (RCB) where it would contribute to a pressure rise in the building. There would be an eventual loss of forced coolant flow to the reactor core, causing partial core meltdown and release of fission products from the fuel. Furthermore, core metal temperatures might reach levels conducive to the initiation of metal-water reactions, thereby providing additional energy sources which could contribute to the RCB pressure level.

A detailed calculation of these accident conditions, including the potential for core melt and extent of chemical reactions between reactor core materials, associated structures and available water or steam, requires an appropriate determination of time-dependent blowdown and subsequent conditions within the reactor primary coolant system following an assumed rupture. In general, it is necessary to determine the behavior of many related quantities during and following blowdown, in order to accomplish a meaningful evaluation of the overall consequences. To illustrate, Table I of reference (14) lists the important factors to be considered in a typical blowdown calculation; this table is reproduced herewith as Table VI-D-2.

It is apparent through examination of Table VI-D-2 that an analysis of just the blowdown phase of a loss-of-coolant accident constitutes an extremely complex problem. Calculations depend upon a multitude of assumed accident conditions, and many physical and design parameters of the reactor plant. Because the design of the AARR is now in a preliminary stage, it is not considered feasible to attempt such detailed calculations at this time. These problems will be studied further as the plant design

FIG. VI-D-5
PERMISSIBLE CORE HEAT FLUX VS MASS VELOCITY
(INLET TEMPERATURE 120°F)

TABLE VI-D-2IMPORTANT FACTORS DURING BLOWDOWN (14)

1. System Design and Initial Conditions

Primary System and Core Design

Primary System Volume

Initial Coolant Temperature

Initial Coolant Pressure

2. Rupture of Coolant System

Break Size and Shape

Break Location

Time for Break to Occur

3. Primary System Response

Mechanical Effects

Thermal Effects

Amount of Water Remaining in the Primary System

4. Containment Response

Pressure and Temperature of Containment Atmosphere

5. Core Response

Mechanical Effects - Core Damage and/or Core Relocation

Nuclear Effects - Core Shutdown and Nuclear Power Decay

Core Thermal Effects - Amount of Heat Transferred During Blowdown
and Initial Temperatures for Core Heatup

becomes definitive and results of these analyses will be presented in the Final Safety Analysis Report for the AARR.

One further point is noted here: in contrast to Pressurized Water Reactors designed for power production, the AARR primary coolant loop contains only a small amount of water at a temperature above the atmospheric pressure boiling point. During reactor operation at 100 MW only the coolant near the exit of core hot channels exceeds this temperature. Thus, upon depressurization there will be negligible flashing of water into steam, and the major factors contributing to loss of primary system water are those due to (1) decompression of the initial inventory and piping, and (2) normal gravity flow and any continued pumping following occurrence of the assumed rupture. The former effect amounts to a loss of only approximately 50 gal of water for complete depressurization of the primary system; the latter depends primarily upon the design of the primary coolant loop, the location of the rupture and the operating characteristics of the primary coolant pumps for off-design conditions. Although these considerations are described in greater detail in the following paragraphs for each of the two assumed rupture locations yet to be elaborated upon, they are stated here since insofar as possible core melt and metal-water reactions are concerned, they lead to the important expectation that primary system blowdown will not in itself uncover the reactor core and associated structures. This, of course, provides beneficial cooling for decay heat removal, and thereby reduces the probability of achieving metal temperatures necessary for the large and rapid chemical energy release associated with metal ignition.

The discussions following return to elaboration of rupture cases (3) and (4) as defined above and depicted in Fig. VI-D-2. Consideration is given to the accident conditions and their consequences up to the point of evaluation of subsequent core heat-up, meltdown and evaluation of metal-water reactions. Discussions then follow in combined form because of the similarities of the two cases in these respects.

Case 3. Ruptured Inlet Line Outside the Reactor Pool

It is assumed that a complete severance of the 24-in. reactor inlet line occurs just below the pool in the RCB service floor. Decompression of the system causes ejection of approximately 50 gal of water at a temperature of 120°F. Gravity induced flow of water through the ruptured inlet line would continue until the reactor vessel were drained to the level of the inlet nozzle and the inlet pipe were emptied between the point of rupture and the inlet nozzle. Simultaneously, reactor vessel water would flow through the vessel outlet line due to syphoning and the influence of any continued pumping action which might occur. This flow would continue until the water level in the vessel dropped to the elevation of the top of the outlet line gooseneck. At this point, action of a syphon-breaker in the gooseneck would terminate flow from the vessel. It is then assumed that the remaining water in the outlet pipe, contained between the top of the gooseneck and the suction side of the primary pumps, would continue to flow until the pumps ran dry, thereby delivering that additional volume of water to the RCB.

It is difficult to estimate the relative flow rates through these parallel paths. Flow resistance through the inlet line would be small, but that through the outlet line might be enhanced by the pumps, although their efficiency under the assumed conditions would certainly be small. Regardless of this consideration, however, the mean temperature of the total water spilled into the RCB should be no greater than 120°F, since water on the outlet side of the core (at 156°F) would have to pass through the main heat exchangers before reaching the pipe rupture location. With the top of the outlet pipe gooseneck located above the level of the reflector upper plenum shroud, the total water volume delivered to the RCB would be approximately 13,200 gal.

Following this water loss, approximately 4,300 gal of water would be retained within the reactor vessel, with a level extending to just above the reflector upper plenum shroud. All of this coolant would then be stagnant, and core temperatures would rise at an accelerated rate due to residual decay heat.

As stated earlier, subsequent core heat-up and the potential for metal-water reactions are discussed following the elaboration of the remaining case of assumed pipe rupture, immediately below.

Case 4. Ruptured Outlet Line Outside Reactor Pool

Similar to the previous case, it is assumed that the complete pipe severance occurs just below the pool in the RCB service floor. Again, decompression of the system causes ejection of approximately 50 gal of water, but at the normal vessel outlet temperature of 156°F. It is assumed that pumping action continues, forcing flow through the system until the pumps run dry. The total water volume contained between the point of rupture and the suction side of the pumps is approximately 9,000 gal, so that an equivalent volume would be spilled into the RCB. With the pumps dry, the check valves in each pump discharge line would close, thereby maintaining a water-filled inlet line and preventing subsequent syphoning action. In addition to the water delivered to the RCB by the action of the pumps, it is assumed that the inventory contained between the point of rupture and the top of the outlet line gooseneck would leak from the rupture. This would constitute an additional 600 gal of water.

Of the total water volume delivered to the RCB, the initial 2,300 gal would be at normal vessel outlet temperature of 156°F. The remaining amount, approximately 7,350 gallons, would be at a lower temperature, since the shut down core delivers less than 10 percent of full power.

In this case the reactor vessel would be filled with water after termination of pumping action. The vessel water would be stagnant, and as in the previous case, core metal temperatures would rise. The following paragraphs are directed toward the development of an estimate of the temperature history of the shut down core and the consequences which might result from core meltdown and possible metal-water reactions.

It is important to note that for the two cases of primary system rupture just described, even though the system would depressurize essentially

instantaneously, coolant flow would continue through the core zone for a period of time following the accident. This flow is due to a combination of depressurization effects, stored kinetic energy in the water, gravity heads and continued action of primary system pumps. Although it is difficult to estimate the duration and time history of core flow rates for these cases, it seems reasonable to assume that in view of the large water volumes lost from the primary system due to these effects, reduced flow would continue for a significant period of time, probably of the order of several seconds. Since the reactor is made sub-critical by control plate scram within 350 msec following the rupture, it is concluded that during the time required to achieve stagnant coolant conditions in the reactor vessel, the decay heat of the core would have been reduced to a low value.

It is considered likely that the bulk of the reactor core, with the possible exception of the core hot spots, could withstand this stage of the accident without melting.

ORNL data for decay heat generation shows that at one second following shutdown the associated power level is 7 MW and falls quite rapidly in the ensuing few seconds.¹⁵ Applying an additional conservatism factor of 1.5 to the ORNL value of 42,000 Btu/hr-ft², a postulated nominal maximum heat flux in the core at 7 MW would be ~63,000 Btu/hr-ft². Based upon an overall nominal maximum-to-average ratio of 1.5 applied to one-half the total decay heat generation attributed to beta radiation, the average heat flux would be ~53,000 Btu/hr-ft². At the hot spots in the core (based upon the design value of 2.5 for the ratio of hot spot-to-average heat flux) the heat flux would then be ~93,000 Btu/hr-ft².

Heat transfer tests performed at ANL, employing 0.040 in. coolant channels and low pressure, show that with only natural circulation of the coolant, available heat fluxes in excess of 70,000 Btu/hr-ft² can be tolerated without melting.¹⁶ Based upon these data and the assumption that the fuel zone geometry is not grossly distorted, it is seen that the maximum shutdown heat fluxes are below the minimum critical value determined in the tests throughout the bulk of the core. Since the reactor design provides coolant flow paths in parallel with the core and the decay power level falls rapidly during a short time following shutdown, it is considered likely that with the exception of hot spots the AARR core would be adequately cooled by natural convection following termination of forced flow.

The hot spots of the AARR core are considered to represent no more than 5 percent of the active fuel plate volume, and consequently melting of these regions would not produce a major fission product release relative to the full core potential. Also, a complete metal-water reaction of hot spot aluminum content would yield only ~65 MW-sec of energy and evolve only ~150 cu ft of hydrogen at STP. Ignition of this hydrogen would produce an additional energy release of ~48 MW-sec. These potential energy sources are small in comparison to others contributing to the RCB overpressure resulting from the assumed accident. Furthermore, the volume of hydrogen evolved is insufficient to cause enough additional water displacement from the reactor vessel to cause the core or reflector to become uncovered.

In the case considered, wherein the assumed pipe rupture occurs in the inlet line to the vessel, a free air volume would exist within the vessel above the reflector upper plenum shroud. The evolved hydrogen could collect therein and form an explosive mixture with the remaining air. Thus, the possibility of deflagration within the vessel exists for this case, the result of which could cause expulsion of much of the water in the vessel. This would enhance the possibility of further core melting and subsequent metal-water reactions which could drastically increase the consequences of the accident. Therefore it appears desirable to provide a means for venting hydrogen from the vessel to obviate this possibility. This problem will receive further study during the detailed design phase of the AARR plant.

The foregoing discussions regarding the potential for core meltdown and associated metal-water reactions have been developed to provide an estimate of what is considered to be the most likely sequence of events and consequences thereof following a massive rupture of the AARR primary system occurring outside the Reactor Pool. Admittedly certain assumptions have been made which at this stage of the reactor design and at the current state of technology (primarily with respect to understanding of the heat transfer mechanisms) cannot be considered to be substantiated by quantitative analyses of a decisive nature. Because of these limitations on analytical methods it is difficult to define a conservative process for evaluation of the accident without resorting to a completely pessimistic and unrealistic postulation involving the assumption of thermal insulation of the fuel plates, which would eventually lead to metal temperatures sufficiently high to cause ignition phase metal-water reactions.

Proceeding along these lines, although the sequence of events is considered to be clearly incredible, it is estimated from integration of the decay heat function that the core hot spots would reach the instantaneous ignition temperature of 1750°C at approximately 5-1/2 sec following shutdown, and that the average core metal would reach this temperature at a corresponding time of approximately 19 sec. This temperature is far in excess of the aluminum melting point (660°C), and it seems obvious that much of the aluminum content of the core, in particular the cladding material, would be lost from the fuel plates prior to the achievement of the ignition temperature and would likely freeze in the water or on structural materials below the core without significant chemical reaction. However, proceeding with a worst-case approach, no credit is taken for this effect. Upon ignition it is assumed that the aluminum burns according to the large-particle rate law given in Appendix B below, as follows:

$$\text{Rate of aluminum consumption} = 1.6 \times 10^{-2} \frac{\text{gm}}{\text{cm}^2\text{-sec}} .$$

Each square cm of heat transfer area of the AARR core has an associated aluminum mass of 0.18 gm, and thus the chemical reaction goes to completion within slightly more than 11 sec following ignition. This calculation assumes no increase in metal area available for reaction with the water.

The total chemical energy release due to this reaction is approximately 1300 MW-sec, and the total hydrogen evolved is approximately

3000 cu ft at STP. Complete ignition of the evolved hydrogen would release an additional 965 MW-sec. In this case, just the volume of the hydrogen released would be sufficient to drive water from the vessel and uncover the core and reflector regions, and again a provision in the design for venting the evolved gas from the vessel appears to be highly desirable.

It is again emphasized, however, that the credibility of metal-water reactions just described is extremely doubtful. It is indeed difficult to visualize that temperatures of 1750°C could possibly be achieved in a flooded core. Further, it would be expected that extensive core disassembly should occur at temperatures far below 1750°C, and that the much lower reaction rates associated with metal temperatures below 1600°C are far more likely. For example (see Appendix B), at a metal temperature of 1500°C and assuming that the area available for reaction is twice the original heat transfer area of the core, the reaction rate is a factor of 1400 lower than that assumed above for the case of ignition. A complete reaction of the core metal under these conditions would require a time of approximately 260 min. It is further considered quite doubtful that complete reaction would occur, since burning particles could be easily quenched by contacting relatively cool structural materials as well as the pressure vessel walls.

Therefore, in spite of the initial pessimistic assumption of core thermal insulation, it is concluded that something significantly less than a complete metal-water reaction would occur and that the time scale for the overall chemical energy release would be measured in minutes rather than seconds.

Because of the fact that the core and reflector are expected to remain submerged in stagnant water, as pointed out earlier, and because only a small fraction of the core (at hot spots) is expected to melt, the incidence of a beryllium-water reaction in connection with such accidents as described in this Section is not considered credible. Ignition of the beryllium would require reaching temperatures in the beryllium of the order of 1280°C; this could be accomplished only if the beryllium were uncovered and either contacted by pieces of molten and burning aluminum, or exposed to the thermal radiation from a large mass of molten and burning aluminum within line of sight of the beryllium. Neither of these is credible with the rupture accidents described here. Similarly, because of the limited extent of core melting anticipated with these reactions, the postulated accident of a massive quantity of burning fuel material melting its way through the bottom of the reactor vessel and subsequently through other barriers below (the so-called "China Syndrome") is not considered credible. (Beryllium-water reactions and the "China Syndrome" accident are further discussed in Section VIII-C.)

In summary, it is concluded that the following would be the results of major loss of coolant accidents in the AARR primary system:

Case 1: Ruptured Inlet Line Inside Reactor Pool

(1) Initial release of ~50 gal of primary water at 120°F to Reactor Pool.

- (2) Almost instantaneous primary system depressurization.
- (3) Automatic reactor scram.
- (4) Continued circulation of primary coolant, with the Reactor Pool serving as part of the coolant loop; mass velocity in core reduced from normal 11.6×10^6 to $\sim 5.2 \times 10^6$ lb/hr-ft²; core inlet and outlet pressure ~ 28 and ~ 10 psia, respectively.
- (5) No core melting, and no fission product release.
- (6) Continued operation of secondary coolant system to remove core decay heat.
- (7) Rise in Reactor Pool water temperature to equilibrium value ($< 120^\circ\text{F}$) determined by outside weather conditions as they affect secondary coolant temperatures.

(Note: If main primary and secondary coolant pump motors were incapacitated by, say, a Class-III (utility) power failure simultaneously with or shortly after the rupture, pony motors on these pumps would be more than adequate to remove decay heat from the core.)

Case 2: Ruptured Outlet Line Inside Reactor Pool

- (1) Initial release of ~ 50 gal of primary water at 156°F to Reactor Pool.
- (2) Almost instantaneous primary system depressurization.
- (3) Automatic reactor scram.
- (4) Continued circulation of primary coolant, with Reactor Pool serving as part of the coolant loop; mass velocity in core reduced from normal 11.6×10^6 to $\sim 9.4 \times 10^6$ lb/hr-ft²; core inlet and outlet pressures ~ 90 and ~ 30 psia, respectively.
- (5) No core melting, and no fission product release.
- (6) Continued operation of secondary coolant system to remove core decay heat.
- (7) Rise in Reactor Pool water temperature to equilibrium value ($< 120^\circ\text{F}$) determined by outside weather conditions as they affect secondary coolant temperature.

(Note: If main primary and secondary coolant pump motors were incapacitated by, say, a Class-III (utility) power failure simultaneously with or shortly after the rupture, pony motors on these pumps would be more than adequate to remove decay heat from the core.)

Case 3: Ruptured Inlet Line Outside Reactor Pool

- (1) Initial release of ~ 50 gal of primary water at 120°F to RCB.
- (2) Almost instantaneous primary system depressurization.
- (3) Automatic reactor scram.
- (4) Drainage of primary water in top of reactor and inlet line above rupture point, through rupture by gravity and through outlet line by syphoning and continued action of primary pumps, until water level falls to top of outlet line gooseneck (above core and reflector). Thereafter, forced draining of water in outlet line downstream of gooseneck by action of gooseneck syphon-break, until primary pumps run dry. Total water drained $\sim 13,200$ gal at average temperature $\leq 120^\circ\text{F}$.

(5) Rise in core metal temperature (now in stagnant water) as result of decay heat generation.

(6) Probable melting of core hot-spot areas (~5% of core).

(7) Possible metal-water reaction of hot-spot areas (~65 MW-sec energy release) and evolution of ~150 cu ft of hydrogen at STP (combustion of which would release an additional ~48 MW-sec).

(8) Possible (considered almost incredible) further metal-water reaction and hydrogen release, significantly less than theoretically possible ~1300 MW-sec energy release plus ~965 MW-sec additional from hydrogen combustion, with energy release time measured in minutes rather than seconds.

Case 4: Ruptured Outlet Line Outside Reactor Pool

(1) Initial release of ~50 gal of primary water at 156°F to RCB.

(2) Almost instantaneous primary system depressurization.

(3) Automatic reactor scram.

(4) Forced drainage of water equivalent to contents of outlet line between rupture point and primary pumps, by continued action of these pumps, until they run dry. Gravity drainage of water in outlet line between rupture and top of gooseneck. Total water drained ~9650 gal at average temperature <156°F. (Initial 2300 gal released at 156°F; remainder at lower temperature because of core shutdown.)

(5) Rise in core metal temperature (now in stagnant water) as result of decay heat generation.

(6) Probable melting of core hot-spot areas (~5% of core).

(7) Possible metal-water reaction of hot-spot areas (~65 MW-sec energy release) and evolution of ~150 cu ft of hydrogen at STP (combustion of which would release an additional ~48 MW-sec).

(8) Possible (considered almost incredible) further metal-water reaction and hydrogen release, significantly less than theoretically possible ~1300 MW-sec energy release plus ~965 MW-sec additional from hydrogen combustion, with energy release time measured in minutes rather than seconds.

(Note: This is considered the worst credible accident in AARR - see Section VII.)

3. Loss of Primary Pressure (Other than by System Rupture)

Under normal operation, water will be bled at a constant rate from the primary coolant loop through a manually adjusted let-down valve,* and discharged to the low pressure primary degasification and purification system. After treatment, this water is delivered to a make-up storage tank. Primary system pressure is maintained at the operating level by a pressurization pump which draws water from this tank and returns it to the primary loop through a pressure control valve responsive to reactor inlet pressure.†

* Ed. Note: This valve is called the pressure control valve in Section IV above. It is manually set to control reactor inlet pressure.

† Ed. Note: This valve is called the flow control valve in Section IV above and is manually adjusted for the correct flow.

Loss of primary loop pressure (other than by system rupture) can be caused by failure of the operating pressurization pump, failure (open) of the let-down valve, or failure (closed) of the pressure control valve. In the event of falling system pressure, corrective action will be initiated in two stages. An initial significant loss of pressure (first trip) will result in the following automatic actions, in addition to control room annunciation:

a. Start-up of the standby pressurization pump in parallel with (and identical to) the operating unit, if it is not already in use.*

b. Closure of both the let-down valve to the degasification and purification system, and a block valve in series with it. (This is accompanied by automatic closure of the block valve in the small line which normally passes primary loop water to the fission product monitor.)

These actions will either return the primary pressure to its normal operating value or reduce the rate of pressure loss to a very small value corresponding to normal primary loop leakage (assuming no system rupture), depending on the cause of the original pressure loss. For operating pressurization pump failure, or let-down valve failure (open), system pressure will return to normal. For pressure loss initiated by failure (closed) of the pressure control valve, the pressure will hold at a value slightly below the first low pressure trip setting, and fall very slowly thereafter at a rate corresponding to normal loop leakage. In this case adequate time will be available to restore pressure by manually controlled intermittent operation of the emergency pressurization pump. This is a third pump which has a lower capacity than the normal operating and standby pumps, but is supplied with Class-I uninterrupted power and discharges directly to the primary loop (bypassing the pressure control valve).

In the event of simultaneous loss of both the operating and the standby pressurization pump, either by pump or power failure, regardless of pressure at the moment of failure, the let-down valve, its block valve, and the fission product monitor block valve will be closed, and the smaller emergency pressurization pump will be started, all automatically.

In any event, ample time will be available for assessment of the situation and administrative decision as to further reactor operation, mode of shutdown, etc. Damage to the reactor core will be prevented under any of the failure mechanisms described.

A further rapid loss of pressure after these automatic corrective actions have occurred can only be caused by system rupture. For this situation, a second trip level is provided which initiates reactor scram (see Section VI-D-2). The value of the first trip setting will be selected so

* Under special conditions requiring a high rate of degasification and/or purification of primary coolant, the setting of the let-down valve [Ed. note: Should be flow control valve] for a relatively high bleed rate may necessitate simultaneous operation of both pressurization pumps to maintain desired primary loop pressure.

as to assure that pressure loss by any of the three modes indicated will be arrested prior to reaching this second trip point.

4. Failure of Utility Electrical Power

Although the utility electrical service is considered to be a dependable source of power, local sources of electrical energy are installed at the AARR Facility. These sources are designated Class-I and Class-II power.

Class-I power is a battery power system which is always connected to the vital equipment such as emergency cooling pumps,* reactor controls and instrumentation which cannot tolerate any interruption of power. Two separate battery banks, rated for 250 V service, provide emergency power for the primary and secondary coolant pump pony motors.† Either battery bank is capable of furnishing all power required for the emergency cooling equipment for one-half hour. However, since only one pony motor is required to operate in each coolant system (primary and secondary) to provide the necessary flow, either battery bank is in effect capable of cooling the reactor (at power below 10% of design power level) for one hour.

Uninterrupted power is supplied to the reactor instrumentation and control equipment by means of two 125-volt battery banks.‡ Each of these battery banks is capable of providing power for one-half hour to all vital instrumentation and control equipment.

Equipment required to maintain the facility at a minimal operating condition during and after shutdown of the reactor is connected to the Class-II electrical power source. Two full-capacity diesel-generators provide this electrical power. Equipment connected to this supply would suffer a short period of down-time extending from the moment utility power is lost until the diesel-generators are delivering power to the equipment. A conservative estimate for the re-energizing time of this equipment is one minute. (Existing diesel-generators at ANL are started in less than fifteen seconds.) Sufficient diesel fuel storage capacity is provided to operate both of the fully loaded diesel engines for twenty-four hours.

The loss of utility power is sensed by a loss of voltage and a reduction in the flow of primary coolant, and the reactor power is automatically set back. The pony motors on the primary and secondary pumps continue to run, as they are connected to the Class-I power supply, the emergency pressurization pump is started, and the let-down valve** and block valve through which water discharges from the primary loop to the degasifier are closed.

* Ed. Note: This should be "main primary pump d-c pony motors."

† Ed. Note: The secondary coolant pump pony motors were removed from the system in the September 1, 1967 design.

‡ Ed. Note: These are 64/32 V battery banks as described in Section IV-I above.

** Ed. Note: This should be the pressure control valve.

The emergency (Class-II) generators are started from a loss-of-voltage signal and come up to voltage and synchronous speed. Circuit breakers in the Class-II switchgear close automatically to deliver power to the Class-I battery chargers, and other emergency equipment. Equipment such as radioactive exhaust fans and reactor clean-up system* are restarted automatically.

If, after a utility power failure, one diesel-generator fails to start, the output of the operating generator is connected through a tie circuit breaker to pick up the scheduled load of the faulty generator.

If neither diesel-generator is able to supply emergency power, the battery system is capable of supplying all of the emergency cooling equipment for one-half hour. Since the emergency cooling equipment is supplied in duplicate, the total battery system would be able to carry the minimum required emergency cooling equipment for one hour.

5. Instrumentation and Control Malfunction

a. General

Sufficient instrumentation will be installed to indicate accurately the state of the reactor, the coolant systems, the auxiliary equipment, and the plant area at all times. Visual and audible alarms will alert the operators to abnormal conditions, and where appropriate will initiate protective action. In general, the reactor safety circuits are relied upon to protect the plant against the effects of a malfunction in any other part of the instrumentation and control system. Within the reactor safety system itself, instrument channel redundancy, isolation of redundant channels, diversity of dissimilar protective channels and periodic testing are employed to make the probability of failure of the safety system extremely low.

b. Reactor Instrumentation

A malfunction of the instrumentation associated with automatic flux control could conceivably result either in failure to reduce reactivity when appropriate, or in the addition of reactivity when not appropriate. Either event is very unlikely, because redundant control channels are employed in the interest of maximum continuity of reactor operation. However, should a failure of this control system lead to reactor powers that are too high for the existing rate of coolant flow, automatic motor-driven insertion of the shim-safety rods would be initiated by the control system.

Similarly, it is possible that a control circuit fault (short-circuit) could cause one or more shim rods to drive incorrectly in the direction of increasing reactivity. In most instances this malfunction would be terminated by control circuit interlocks, such as withdrawal inhibits, etc. However, in this example, as well as in the preceding one, the ultimate protection is a scram that is initiated automatically by the safety system.

* Ed. Note: "Reactor clean-up system" is called "clean-up exhaust system" in Fig. IV-52.

Malfunctions of the reactor instrumentation that transmit to the operator an apparently reasonable, but incorrect, signal will be detected through redundant channels measuring the same parameter.

c. Process Instrumentation

Process instrumentation failures can result in erroneous action of control and block valves, and other incorrect system responses, which can in turn cause undesired changes in primary coolant flow, pressure, and temperature. Loss of flow and loss of pressure have been discussed above. (See Sections VI-D-1, -2, and -3.)

Malfunction or failure of certain process instruments can also result in control or block valve adjustments tending to increase primary and secondary flow rates. Since these flow rates are normally close to the upper limit of the pump capability of each system, the increase in flow rate and the attendant reduction in primary water temperature at the reactor inlet will be relatively small. This small reduction in primary temperature will result in a relatively low net gain in reactivity. The rate at which the primary coolant temperature reduction will occur will not be high enough to cause serious reactivity transients. At worst, the increased power may cause a high level scram with no danger of core damage.

d. Consideration of Fires

A fire that occurs among the interconnecting cable ways may in general result in unwanted short-circuits, and/or open circuits.

Nuclear Instrumentation

With one exception, either grounding or open-circuiting of the inner conductor of a coaxial cable of a safety system channel will result in the transmission of a "channel-tripped" output signal to the scram circuit logic. The exception involves the grounding or open-circuiting of the inner conductor of a coaxial cable carrying ionization chamber compensating voltage. This failure will cause an apparent increase in chamber output signal because the gamma compensation is lost. This effect is not of itself hazardous.*

Process Instrumentation

The short-circuiting, grounding, or open-circuiting of process instrument signals will, in general, result in either zero signals, or off-scale, very high signals. However, a thermocouple extension lead short-circuit in the presence of the heat of a fire could form a new hot junction that conceivably might be indicated as a reasonable temperature on the associated instrument. It is judged that this delicate effect could be maintained at most for only a few seconds before the junction temperatures would change radically and/or other circuits would begin to behave in definitely abnormal ways. In any case fire-caused faults in the process instrumentation can be expected to result very quickly in abnormal signals

* Ed. Note: As indicated in Section IV-H above, there is only one channel using a compensated ionization chamber (the wide-range linear channel) and this channel has no automatic control or safety function.

that will be detected by alarm or scram circuits depending on the instrument channels affected.

Control Rod Drives

A situation may be considered in which a short-circuit is initiated by a fire in the electrical leads between the rod drive starters and the reactor control console. Further, this event could mimic closing of the control contacts calling for rod drive in the direction of increasing reactivity. Such an event is discussed in paragraph b above.

6. Secondary Coolant System Malfunctions

Losses of secondary coolant, secondary coolant flow or secondary system heat removal capacity are of direct concern primarily in consideration of the consequences of natural disasters such as earthquakes and tornadoes, and are discussed in Section VI-F-2 below.

E. Radioactivity Release Unrelated to Primary System Rupture

1. Core Meltdown without Primary System Rupture

In the event of core meltdown caused by either reactivity excursions or flow blockage, but where the primary system is not breached, fission products will be contained in the completely shielded primary system. Sufficient shielding is provided to reduce the dose rate in normal access areas of the plant to a value such that personnel normally working in these areas can be evacuated without undue exposure, for the case of a core meltdown where the fission products are distributed throughout the system. The operation of the off-gas system under these conditions is briefly discussed in Section IV-M above.

The off-gas radiation level at which off-gas system isolation will occur will be such that radiation dose levels from the release of activity from the stack will not exceed permissible levels at any point under conservative atmospheric conditions. Further, with the dispersed fuel within the shielded primary system, the direct radiation dose from the containment is tolerable. Following shutdown of the reactor, the primary coolant can be discharged to the radioactive waste system for treatment, holdup, and eventual disposal. The procedure to be used, of course, is dependent upon the extent of the core meltdown and condition of the primary system.

2. Stack Release of Radioactivity during Normal Operation

The radiation level of Reactor Containment Building exhaust air will be monitored prior to discharge of the air to the stack, as described in Sections IV-H and IV-L above. Two trip points will be incorporated into the exhaust air radiation monitoring instrumentation. These trips will result, successively as the radiation level increases, in (1) the sounding of an alarm in the control room, and (2) the automatic actuation of the Reactor Building isolation system. Three monitoring instruments are used, and the circuits are arranged in two-out-of-three redundant configuration for the second (higher level) trip.

The set-point for the alarm trip will be selected such that continuous stack release at any level below the trip point will result in a cumulative radiation dose, received over a one-year period at ground level and at any distance from the stack, not exceeding the maximum permissible annual dose for individuals in uncontrolled areas, as defined in Reference (17). In determining the corresponding activity level at the stack exit, a conservative model of average wind-direction frequencies and average meteorological conditions will be chosen to reflect the influence of these conditions on the average dose rate to ground-level receptors.

The set-point for automatic actuation of the isolation system will be such that continuous stack release at activity levels lower than the set-point could be continued for a certain non-trivial specified time without significant radiation hazard to people outside the Reactor Containment Building. During that time interval, the Shift Supervisor will have an opportunity to determine and rectify the cause of this high activity level. The details of procedures, etc., will be clarified during Title-II design. In selecting the high

set-point, the associated short-term radiation doses will be calculated for conservative conditions of wind speed and meteorological conditions.

Purely for illustrative purposes, assume that as a result of release of activity at the level of the low set-point the largest annual dose is equal to one-half that specified as maximum allowable in Reference (17). Assume further that the high set-point is selected so that, under the more pessimistic meteorological conditions postulated for short-term release of activity, the maximum dose rate to receptors outside the Reactor Containment Building would be 1000 times the maximum dose rate corresponding to the low set-point.

If the level of activity of the exhaust air were at the high set-point for as long as one hour, the total activity released would correspond to that released at the maximum allowable rate for continuous release for a total period of 500 hours, and would yield ~6% of the maximum allowable average annual radiation dose. By comparison, it is calculated that stack release of MCA-contaminated air for 3 sec at the normal ventilation rate might yield an incremental plume-inhalation dose to the thyroid as high as 2 rem at 15,000 m (see Section VII-C-2 below). The present maximum allowable average annual nonoccupational dose is ~3 rem per year, for iodine isotopes I^{132} through I^{135} , as inferred from Reference (17). Thus, with respect to these iodine isotopes, the high set-point of air activity would be less than 1×10^{-4} of the postulated MCA level of air contamination.

It therefore appears feasible to establish two set-points which: (1) would afford the Shift Supervisor an opportunity for remedial action, for example, in the event of a radiation spill; and (2) would cause a prompt containment isolation at activity levels far below MCA levels, assuming the rate of increase of exhaust-air activity to be low.

3. Argon⁴¹ Release

An estimate has been made of the amount of A^{41} continuously purged from the 250-foot stack during normal operation of the reactor. In order to determine the amount of A^{41} leaving the stack, the sources for this radio-active isotope were examined, and it was found that the major sources of A^{41} activation are the neutron beams from the experimental beam tubes. All other sources can be neglected.

The analysis made to determine A^{41} concentration is based on 100 MW operation. Results are compared with both the maximum permissible biological concentration and the maximum permissible background level for counters at the ANL site. The maximum permissible biological concentration of A^{41} is 2×10^{-6} $\mu\text{c/cc}$ for a 40-hour week base or 4×10^{-7} $\mu\text{c/cc}$ for a 168-hour week base.¹⁸ The maximum limit assumed for background level at counters at or beyond 1 km radius from the reactor is 10^{-9} $\mu\text{c/cc}$.

For reasons of neutron economy, the neutron beams inside the biological shield will not pass through air. Thus, A^{41} will only be produced by the beams outside this shield. When a typical group of neutron beam experiments is in progress, a maximum of about 5×10^4 cm^3 of air will be exposed to the beams. If it is assumed that the average intensity of the beams is 1×10^9 $\text{n/cm}^2\text{-sec}$, then the total rate of A^{41} production will be about 1.9×10^{-2} $\mu\text{c/sec}$.

Suitable local ventilation will be provided at each experiment to maintain the A^{41} concentration below the maximum permissible biological concentration as given above. In addition, a total ventilation rate of about 20,000 cfm will be maintained for the Reactor Containment Building to prevent A^{41} back-diffusion.

The maximum concentration downwind from the 250-ft stack is estimated to be about 2.6×10^{-14} $\mu\text{c/cc}$. This maximum concentration is less than the maximum permissible background level for counters by a factor of about 40,000.

The above results are based on decay of A^{41} after emission from the stack and rise of the plume¹⁹ above actual stack height. The worst meteorological conditions are assumed, i.e., inversion with a wind velocity of 2 m/sec.²⁰

In no event will the maximum permissible biological concentration be exceeded.

F. Fire and Environmental Accidents

1. Fire

The Reactor Containment Building and internal structures are of concrete and steel, and therefore not flammable. Small local fires could occur in experiment equipment or in the small quantities of combustibles which may be present. Such fires will not endanger the safe functioning of the reactor safety mechanisms. Within the containment volume, smoke detection devices will be utilized to inform both the control room and fire department of any local fires. Water hose cabinets, and portable water and CO₂ fire extinguishers will be available to combat localized fires. A specific analysis of fire hazards with respect to instrumentation and control wiring is presented in Section VI-D-5 above.

Fire and explosion hazards connected with experiments will be evaluated in detail as part of the safety review which will be carried out on all proposed experiments. Special protective features will be provided where necessary, as part of the experiment equipment so that none of the plant safety criteria will be compromised.

2. Natural Disasters

a. Earthquakes

The seismology of the area is discussed in Section III-E above. It is indicated that the Argonne site is not located in a major earthquake region and the occurrence of a large tremor is not anticipated. However, as indicated in Section II above, Criterion 1.4, structures and equipment important to safety in the facility will be designed to withstand accelerations greater than those which could ever be expected in the area in which the Argonne site lies. Such structures and equipment are designated as "Class-I" items; in the current design these items are as follows:

- (1) Reactor Containment Building, and all its penetrations.
- (2) Reactor Containment Building isolation valves, valve operators, associated radiation monitoring channels, etc.
- (3) Reactor vessel, internals and support.
- (4) Primary coolant system, including all related vital instrumentation.
- (5) Control rod drives and supports, and equipment necessary to provide control room indication of rod insertion.
- (6) All experiment facilities whose failure could add reactivity to the system.
- (7) Class-I uninterrupted power supply system including batteries, Class-II emergency diesel-generators including their fuel storage and supply systems, and all electrical switchgear and equipment necessary to transmit power generated by these systems to the equipment supplied therefrom.

(8) Control room and adjacent relay rack and cable rooms.

(9) Earthquake detector and associated scram circuitry.

In view of the Argonne site location and the items designated Class-I to be designed for earthquake resistance, it is not considered credible that an earthquake at or near the AARR site could cause a reactor accident.

The following analysis describes the results of a maximum earthquake occurring at the AARR site, and demonstrates that the plant can be maintained in a safe condition following such an event. Since the structures and equipment designed Class-I for earthquake-resistant design are the same as or are contained within structures to be designed for tornado resistance (see Section II above, Criteria 1.5 and 1.6), the analysis also applies equally to the case of a tornado striking the facility. In either case, it is assumed that structures and equipment designed to withstand earthquake and/or tornado conditions remain intact during the disaster, while all other structures and equipment are either destroyed or rendered inoperable.

Specifically it is assumed that, as a result of damage to structures and equipment not considered Class-I items, one or more of the following may occur simultaneously as the result of the disaster:

- (1) Failure of utility electric power supply.
- (2) Secondary coolant system flow failure.
- (3) Loss of secondary coolant system heat removal capacity.
- (4) Loss of secondary coolant.

If the disaster is an earthquake, the earthquake detector will initiate reactor scram at the moment of the tremor. If the disaster is a tornado strike, in most cases the operators will have received advance warning thereof (see Section VI-F-2-c below), and will have shut the reactor down by administrative procedure prior to the actual strike. In either of these cases, if the only plant failure is utility power failure, the primary and secondary pumps will continue to function on their pony motors* (supplied by Class-I and Class-II power) to dissipate the reactor decay heat to the cooling tower. Again in either of these cases, but where the disaster results in one of the secondary coolant system failures listed above, primary coolant circulation will be continued either by the main pump motors or the pony motors (depending on whether or not utility power failure also occurred), but primary coolant temperature will rise because of loss of secondary system capacity to dissipate the decay heat still being generated by the core. This effect is discussed below.

*Ed. Note: The secondary coolant pump pony motors were removed from the system in the September 1, 1967, design.

In the event of a tornado strike in which only utility power failure occurs, the plant will function as described in Section VI-D-4 above, with an automatic reactor power setback to 10% power. Thereafter the operators may effect a complete shutdown manually if desired, by administrative decision.

The worst case is one in which a tornado strikes without warning, a secondary system failure but no utility power failure occurs, and no action is taken by the operators to scram the reactor manually. In this case the reactor would continue to operate at 100 MW until the temperature of the primary coolant discharging from the main heat exchangers and passing the reactor coolant inlet temperature sensors rose to the inlet temperature scram trip setting (assumed in this analysis to be 135°F), at which point scram would occur. If coolant temperatures in the primary system were normal prior to the accident (120 and 156°F reactor inlet and outlet, respectively) and the secondary system failure were sudden and complete, then the delay in scram beyond the time of secondary system failure would be ~ 15 sec, and the average primary system coolant temperature would rise to 147°F at the time of scram (as compared to 137°F in normal operation). However, if primary coolant temperatures prior to the accident were above normal, e.g., just under the assumed scram trip level of 135°F at the reactor inlet and 171°F at the reactor outlet, then the average coolant temperature at the time of scram could be as high as 161°F. This value is used in the succeeding analysis.

As mentioned above, after reactor scram, the primary coolant temperature will rise (or continue to rise) because the secondary coolant system is incapable of rejecting the decay heat still being generated in the reactor. Table VI-F-1 shows the average primary coolant temperatures at various times after scram, assuming no heat losses from the system to the ambient in the Reactor Containment Building or to the Reactor Pool and structures, and assuming the main primary pump motors are shut off, leaving the pony motors running. The table also demonstrates the decline in the maximum heat flux from the fuel to the primary coolant as the afterheat decays following shutdown. Shutdown heat transfer tests conducted as part of the AARR research and development program have shown that heat fluxes as high as ten times those indicated, even under natural circulation and flow reversal conditions (in the case under consideration at least 10% flow is generated by the primary pump pony motors) would not result in fuel burnout.

Table VI-F-1 also lists the cumulative energy released to the primary coolant by the decaying core, and the estimated potential cumulative energy dissipation to the Reactor Pool and structure at the indicated average water temperatures. (The values shown for average water temperature do not include such dissipation.) Note that the energy release and the potential dissipation values are equal at 8 hours after scram. From this it can be concluded that the average temperature of the primary coolant could be expected to level off at a value somewhat below 300°F, with energy generated by the core balanced by system heat losses. Primary system pressure to assure adequate subcooling could be maintained by operation of the emergency pressurization pump, which is supplied with Class-I and Class-II power, or by the normal pressurization pumps if utility power is available.

This analysis demonstrates that even under highly conservative assumptions, the reactor system is capable of absorbing the decay heat generated by the core for at least 8 hours following shutdown and failure of

TABLE VI-F-1

PRIMARY COOLANT SYSTEM CONDITIONS
AFTER SECONDARY COOLANT SYSTEM FAILURE

Time After Scram (hr)	Average Primary Coolant Temperature (°F) (a)	Maximum Core Heat Flux (Btu/ft ² -hr) (b)	Cumulative Decay Energy Released by Core (Btu)	Cumulative Potential Energy Loss to Pool and Structure (Btu) (c)
1	203	8400	6.3×10^6	3.0×10^6
4	256	5200	1.5×10^7	1.3×10^7
8	306	4000	2.2×10^7	2.2×10^7
24	457	2500	4.2×10^7	1.5×10^8

(a) Assuming no heat losses from primary system; based on 161°F at time of scram.

(b) Based on very conservative assumption of \$1.00 shutdown margin.

(c) Estimated on the basis of indicated average primary coolant temperatures.

the secondary coolant system. Beyond that point, if repairs could not be effected and at least reduced capacity operation of the secondary system resumed, the primary system temperature could be reduced by emergency pressurization pump injection of cold water (and bleed-off of hot water from the primary system), the system depressurized gradually, and the reactor vessel opened. Natural convection transfer of further decay heat directly to the pool would then suffice, and the auxiliary means, if necessary, could be arranged to keep the pool at a reasonable temperature until normal services were restored.

b. Floods

The natural drainage of the AARR complex, with the excess water going to the Des Plaines River, eliminates any danger of natural floods. Also the complete air-tight welded-metal liner within the containment volume precludes in-leakage.

c. Tornadoes

Although there is no record of a tornado ever having struck any part of the Argonne National Laboratory Du Page site, sufficient tornado activity has been reported in northeastern Illinois to warrant evaluation of possible tornado effects as a part of the safety analysis.

Tornadoes are characterized by very high wind velocities, a twisting wind action, and considerably reduced barometric pressure in the center of the vortexing air mass. However, accurate measurements of wind velocities and reduced pressures to be expected are unavailable.

The Chicago Forecast Center, U. S. Weather Bureau, has overall responsibility for weather forecasts* and warnings in Northern Illinois. At the Laboratory, forecasts and warnings are received from the Chicago Forecast Center via the National Warning Service (NEWAS); the Emergency Broadcast System may also be used for warnings. A forecast is issued when weather conditions prevail that are conducive to tornadoes. Warnings are issued when tornadoes have been sighted or their existence reasonably substantiated by radar.

Under established emergency procedures, upon receipt of a tornado warning, a tornado watch is established at strategic locations throughout the Laboratory site.** An actual tornado sighting would be reported through regular procedures to the AARR Facility Manager or responsible Shift Supervisor on duty, and the reactor would be shut down by administrative decision until the danger of tornadoes had passed.

Although it is highly unlikely that a tornado would strike the AARR facility with no warning whatsoever, this possibility has been taken into account in the design. As indicated in Section II, Criteria 1.5 and 1.6,

* Ed. note: The word "forecasts" has recently been changed to "watches."

** Ed. note: Laboratory emergency procedures following receipt of a tornado watch or warning have recently been revised.

essentially the same plant features which have been designated for Class-I design to withstand earthquake forces are also being designed for tornado-resistance or to be enclosed within a tornado-resistant structure. Therefore, it is not considered credible that a tornado strike on the AARR facility could cause a reactor accident.

An analysis of the results of an earthquake or tornado strike has been presented in Section VI-F-2-a above.

G. Missile Generation

1. Missile Potential from Primary System Components

The case of missiles originating within the reactor primary cooling system, or as a result of damage to primary system components, has been the subject of a limited investigation with some preliminary results obtained based upon present design concepts. These studies will proceed concurrently with the development of a final design for the AARR in order that maximum protection against missile hazards is provided in the plant. The discussions that follow are divided between two major areas of concern: (a) production of missiles from reactor internals, and (b) missile production by primary system rupture.

a. Missiles Originating in Reactor Internals

The production of missiles within the core zone of the reactor could conceivably occur as the result of an inadvertent reactor power excursion. However, it was shown in Section VI-B above, that the maximum energy release during AARR transients considered credible is approximately 20 MW-sec. It is not considered likely that the rapid release of this amount of energy will cause mechanical damage to the core, and thereby produce missiles.

For hypothetical power excursions wherein complete failure of scram capability is assumed to occur or an excursion is assumed to be induced by an excessively large reactivity insertion, reactor shutdown may occur by core disassembly. Such disassembly may be accompanied by a violent steam explosion; hence, core zone missiles may be generated. Even under these unlikely conditions, it is doubtful that such missiles could penetrate the walls of the reactor pressure vessel.

The minimum horizontal distance between core zone components (core and Be reflector) and the vessel wall is approximately 2 ft. Also, a thermal shield is interposed between the core zone and the vessel (see Figure IV-3). In the vertical direction the corresponding minimum distances are much greater being, for example, approximately eight feet to the neck wall at the bottom of the vessel. Using the method of analysis presented in ANL-5948,²¹ it is calculated that the range of a missile through reactor vessel water would be very short. The expression used to describe the velocity of a missile as a function of distance traveled is as follows:

$$V(x) = V_o \exp \left[- \frac{C_D \rho_w A}{2M} x \right]$$

where; V_o = initial missile velocity, ft/sec

C_D = Drag Coefficient

ρ_w = water density, slugs/ft³

A = missile area normal to direction of travel, ft²

M = mass of missile, slugs

x = distance of travel, ft.

As an example calculation using the above expression, it is assumed that a beryllium missile is produced by spalling from the reflector, and travels horizontally toward the vessel walls. The following constants are assumed in the calculation:

$$C_D = 1.0$$

$$\rho_w = 2.0 \text{ slugs/ft}^3$$

$$A/M = 2.0 \text{ ft}^2/\text{slug}$$

Using these values in the foregoing expression and neglecting the retarding effect of the thermal shield, the missile would reach the vessel wall ($x = 2 \text{ ft}$) with a velocity of less than 2% of its initial value. This calculation can only be considered illustrative, inasmuch as a value appropriate for the quantity A/M is difficult to predict. Nevertheless, the damping provided by reactor vessel water is clearly indicated by this example and hence vessel penetration by internally generated missiles is considered most unlikely.

b. Missiles Originating in Primary System Rupture

The production of missiles by failure of the reactor primary coolant system will be a subject of continuing analysis. As stated in Section IV-D above, the reactor vessel will be designed for 1000 psig and 250°F (or as modified by radiation heating) in accordance with Section III of the ASME Boiler and Pressure Vessel Code. The capability of the vessel to operate at normal plant conditions will be examined periodically through hydrostatic testing as part of the reactor refueling procedure.

The strength of the vessel described herein is illustrated by the results of preliminary calculations of its ability to withstand shock loads and slow pressure increases. Details of the analyses performed and the results obtained are presented in Appendix E below. Briefly stated, the results of these analyses indicate a containment potential of the AARR primary system much in excess of any inadvertent pressure load that can be reasonably postulated. The pressure vessel is capable of withstanding, without rupture, the consequences of the hypothetical reactivity accidents as described in Section VIII-B below, which produce sudden pressure loads. The results of quasi-static calculations indicate that the vessel is capable of containing a maximum internal pressure of 6000 psig,* which is more than a factor of seven above the pressure relief set point for the reactor primary system. Thus it appears inconceivable that missile production through pressure vessel failure in a normal mode should occur due either to reactor power excursions or to malfunction of process controls.

It remains then to examine the possibilities for missile generation due to abnormal failure modes of the reactor pressure vessel. Reference to Figure IV-39 shows that the location of the reactor pressure

* Ed. Note: This number is based on the Inconel-600 vessel and is not necessarily correct for the stainless steel vessel finally selected.

vessel with respect to the biological shield and its immersion in the Reactor Pool is such that only missiles traveling essentially vertically constitute a potential hazard to the integrity of the Reactor Containment Building (RCB). In particular, the pressure vessel head is a potential missile of this type. Therefore, assuming a failure has occurred which results in a sudden release of the vessel head, an analysis has been made to determine possible motions of the head and the potential damping effect of the Reactor Pool. This analysis considers head motion in response to various applied average pressures.

The motion of the head is influenced by the water level in the Reactor Pool at the time of the assumed accident and the drag force exerted upon the head by the water. If the pool is filled with water to its normal level, then the head must travel through approximately 20 ft of water and a force retarding its motion will be experienced. Normal "drag coefficients" are usually measured with the subject body immersed in a fluid, wherein the dimensions of the body are very small compared to its fluid environment. This clearly is not the case for pressure vessel head motion in the pool, especially as the head approaches the pool free-surface. However, to illustrate how the drag forces might influence the motion of the head, assuming the drag force to act over the entire travel distance through the pool, the following equation of motion has been solved:

$$\frac{d^2X}{dt^2} = \frac{144g}{\rho_H t_H} \bar{P}(x') - \frac{C_D \rho_w}{2\rho_H t_H} \dot{X}^2 - g$$

where: X = travel distance of head, $0 \leq X \leq 67$ ft*
(distance to RCB roof)

C_D = drag coefficient, effective for travel
through water, $0 \leq X \leq 20$ ft, = 1.0, 0.5, 0.0

ρ_w = density of pool water $\cong 62$ lbs/ft³

t = time in seconds after beginning of motion

$\bar{P}(x')$ = average net pressure in psi acting on head
over distance x'

ρ_H = density of head $\cong 500$ lbs/ft³

t_H = thickness of head $\cong 25$ in. = 1.25'*

g = acceleration due to gravity $\cong 32$ ft/sec.²

The foregoing equation was programmed for solution by an analog computer, and solutions for various values of x' were found, such that the velocity of the head at the RCB roof ($X = 67$ ft) just equals zero. In

* Ed. Note: This dimension was revised in the September 1, 1967 design.

this manner, the lengths of power strokes and average pressures applied over the chosen power strokes, just tolerable without containment damage, were found. The motion of the head through pool water was varied by changing the value of the drag coefficient, C_D . Table VI-G-1 summarizes the results of these analyses.

Table VI-G-1

Minimum Combinations of Power Stroke and Pressure which
Yield Pressure Vessel Head Travel to Roof of
Reactor Containment Building for Various Values of Drag Coefficient

Power Stroke (ft)	Average Applied Pressure (psi)		
	$C_D = 1.0$	$C_D = 0.5$	$C_D = 0.0$
0.5	2050	890	390
1.0	1290	535	220
2.0	700	290	125
5.0	265	105	55
8.0	145	70	35
12.0	70	40	20

Examination of Table VI-G-1 reveals the strong dependence of the results upon the assumed damping effects of the water in the reactor pool on the motion of the head. As an example of the application of the data of this table, consider the motion of the head, after breakage of its bolts, due to the relief of the reactor primary system pressure. The net initial pressure is 575 psi* which falls to zero after a motion of less than 0.5 ft. Comparing these results with the data of Table VI-G-1 it is seen that this combination of average pressure and power stroke is well below that required for the head to reach the RCB dome for all assumed values of C_D . At this average pressure level, a power stroke of nearly one foot is necessary to cause damage to the RCB.

The results of these brief and rudimentary studies reveal the need for further analysis of the problem of pressure vessel rupture. The sensitivity of the results to the degree of damping of head motion that can be provided by the water in the Reactor Pool illustrates the need for better understanding of this effect. These studies will continue through analysis and/or scale tests as the AARR design develops, so that the question of the need for and type of hold-down or retarding device can be resolved, and proper protection be afforded the Reactor Containment Building against this potential source of damage.

* Ed. Note: This is based on an assumed reactor inlet pressure of 600 psia. The average pressure for this calculation, then, is $575/2$, or 288 psi.

Primary system rupture through beam tube failure, and consequent missile generation, has been examined and is considered highly unlikely. The beam tubes are designed for pressure loads at least as great as those that can be accommodated by the reactor vessel proper. However, under certain hypothetical accident conditions, lateral forces might be imposed upon the tubes causing shear failure within the vessel or at the vessel wall. Nevertheless, a backup safety feature in the form of a safety diaphragm is provided in each tube, to prevent breaching the primary system under such conditions.*

Operational procedures will be employed to eliminate the possibility of missile damage to the Reactor Containment Building in the remote event of combined failure of a beam tube and its safety diaphragm. Filler plugs will be inserted in unused beam port facilities; these plugs will be mechanically secured in the beam ports, thereby obviating their missile potential. For beam tubes in use, the associated experiment equipment will be examined for missile generation potential. If a possibility of missile generation exists, local missile protection in the form of concrete walls, blast shields, etc., will be provided around the equipment.

2. Missile Potential from Experiment Facilities and Miscellaneous Equipment

The production of missiles within the Reactor Containment Building constitutes a hazard directly to personnel working therein and to the integrity of the reactor cooling system and the building itself. Missiles may be generated from a variety of sources other than primary system components, and the following discussions are directed toward an evaluation of safety hazards and methods of control related thereto.

As an engineered defense against such missile damage, all high-speed equipment, which is normally part of the complement within the Reactor Building, will be subjected to evaluation for missile generation potential, and will be properly self-shielded if necessary. This process will also be applied to all equipment brought into the building for experimental, maintenance or other needs, through appropriate administrative control. In addition, the design of the reactor and the Reactor Containment Building will be critically examined to determine locations which might possibly be vulnerable to missile damage, and proper shielding will be installed. Of course, the complete radiation shielding provided for the primary cooling system, which will be in place whenever the reactor is operating, will serve as effective protection against external missile damage to the system and consequent operating hazard. Nevertheless, precautionary measures as described above will be employed to minimize hazards from potential missiles of this type.

In the following paragraphs consideration is given to missiles which might originate from: (a) penetrations in the head of the pressure vessel

* Ed. Note: The backup for beam tube tip failure was changed from a diaphragm to the beam tube collimator (or shield plug) itself and its seal flange (see Section IV-E above).

such as experiment loops and reactor instrumentation and (b) regions external to the pressure vessel and outside of containment.

a. Missiles Originating in Reactor Vessel Head Penetrations

An examination reveals that the potential for missiles arising from the failure of vertical facility penetrations and instrument penetrations in the reactor vessel top head and possessing sufficient energy to cause perforation of the Reactor Containment Building is so remote as to be considered incredible. First, all such penetrations (experiment and instrument) have attached external piping which would certainly act as a missile restraint in the event of a rupture or failure of any penetration. The possibility of a double break resulting in a free object having sufficient energy to become a missile cannot be envisaged. Second, the design strength of the penetrations is in excess of that of the reactor vessel shell; therefore, in the event of an extremely high pressure transient, failure of the shell will result before failure of the nozzles or penetrations.

Top penetrations not in use will be sealed with flanges. In the case of the experiment facilities, the largest flange will be about 8 in. in diameter and about 2 in. thick. It is highly improbable that bolt failure can occur and that a flange can be propelled vertically upward through the Reactor Pool with sufficient energy to perforate the containment. For example, calculations show (see analysis of containment structure perforation during tornadoes, pages 11-15 of Supplement 1 to this report)* that for a cylindrical object equivalent in volume and mass to a flange measuring 4 in. in diameter and 8 in. long, a velocity of approximately 190 feet per second is required to perforate a 3/8-in. steel plate (for example, the steel containment liner). Actually the flange (or object) would have to obtain a velocity of about 245 feet per second at the pool surface to have a velocity of 190 feet per second at the apex of the containment liner.

Considering that there will be at least 20 ft of pool water above the reactor vessel head, it is inconceivable that a broken flange will not deviate from a vertical course. Further, because of drag forces its kinetic energy (flange or other object) will be quickly dissipated and its range will be short. The damping potential of the pool water is illustrated in Table VI-G-1 for assumed motions of the pressure vessel head for various drag coefficients. This same table is indicative of the damping effect on a flange should it become a missile.

Although an accident resulting in a missile originating from the vertical experiment facilities possessing sufficient velocity to perforate the containment cannot be postulated at this time, a further analysis of the problem will be made when the vessel head design becomes more definitive. In particular, velocities that can be attained by such missiles just after failure and the damping effect of the water must be examined further

* Ed. Note: This refers to Question 87, Section III, Part 2 of this report.

to determine possible missile velocities at the pool surface. Thereafter, the need for and the type of hold-down or shielding devices can be resolved such that the proper protection can be afforded the Reactor Containment Building against this potential source of missile damage.

b. Missiles Originating Outside the Reactor Vessel or Containment Building

The possibility of missiles originating from regions external to the reactor vessel as a result of, or associated with, the MCA will be considered in detail in the final safety analysis of AARR. Currently no missile induced accident of this type can be postulated.

Missile generation external to the Reactor Containment Building is considered possible, particularly under severe weather conditions. An analysis, presented in Supplement 1 to this report, shows that there is no conceivable way in which missiles generated by a tornado can perforate the Reactor Containment Building.*

* Ed. Note: This refers to Question 87, Section III, Part 2 of this report.

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VII. MAXIMUM CREDIBLE ACCIDENT (MCA) ANALYSIS

A. Introduction and Summary

In Section VI above, credible accidents in various categories (reactivity insertions, fuel handling accidents, equipment malfunctions and accidents caused by natural phenomena) have been analyzed. Evaluation and comparison of these accidents show that the worst accident analyzed is that initiated by a complete severance of the main primary coolant outlet line from the reactor vessel at a point outside the Reactor Pool. Such a rupture would lead to discharge of a significant fraction of the primary coolant water to the Reactor Containment Building service floor, and would probably result in some melting of the core (at the hot spots) and some re-release of fission products through the pipe rupture to the Reactor Containment Building. Core melting could lead to an aluminum-water reaction, probably limited to the molten aluminum at the hot spots, but possibly more extensive, and such reaction would be accompanied by evolution and possible combustion of the hydrogen generated therein. This is indeed the Maximum Credible Accident for the AARR.

It is extremely difficult, if not impossible, to predict accurately and reliably such parameters as the exact path and timing for release of primary coolant, fission products and heat energy to the Reactor Containment Building atmosphere, the exact amount and temperature of coolant released, the amount and type of fission products released, and the extent and time duration of core melting, aluminum-water reaction and hydrogen combustion. Therefore a computational model for the Maximum Credible Accident (MCA) was established, in which conservative assumptions were made as to the parameters mentioned, in order to provide a firm basis for evaluating the consequences of the accident. Taken in toto, the assumed parameters are far more conservative than what is expected to be the true condition, as demonstrated in Table VII-A-1.

In Section VII-B below, the MCA Computational Model is described in detail, and the bases are presented upon which the consequences of the accident are analyzed. In Section VII-C, the anticipated consequences are set forth, and in Section VII-D, the results of additional analyses are presented to substantiate further the conservatism of the results shown in Section VII-C.

A summary of the conclusions which may be drawn from the analysis of the MCA computational model for AARR is as follows:

(1) The exclusion area radius (1500 m), outer radius of the low population zone (5000 m) and population center distance (7000 m) defined in Section II, Criterion 19.1 on the basis of site characteristics described in Section III are reasonable and lie well within the definitions established in the Code of Federal Regulations, Title 10, Part 100.¹

(2) The radiation doses calculated from the MCA computational model are summarized in Table VII-A-2. These doses are far below the limiting doses suggested in the referenced Code. Two-hour and 24-hour doses calculated for various points within the exclusion area and low population zone

TABLE VII-A-1

COMPARISON OF MCA COMPUTATIONAL MODEL WITH ANTICIPATED ACTUAL MAXIMUM ACCIDENT

Parameter	Actual Accident	MCA Computational Model
<u>Primary coolant release</u>		
Quantity	9700 gal	25,000 gal
Temperature	< 156°F	164°F
Time to complete (a)	~ 1/2 min	~ 1 min
<u>Core melt</u>		
Extent	~ 5%	100%
Time to complete (a)	1/2 to 1 min	0 (instantaneous)
<u>Fission product release</u>		
Quantities (% of total core inventory)		
Rare gases	5%	100%
Halogens	2.5% (b)	50% (b)
Solids	0.05%	1%
Path	Core ↓ Stagnant water in vessel ↓ Building air	Core ↓ Building air (direct)
Time to complete (a) (incl. dispersal in bldg.)	>> 1 min	0 (instantaneous)
<u>Core metal-water reaction</u>		
Extent	< 5%	100%
Energy release	< 65 MW-sec	1308 MW-sec
Time to complete (a)	> 260 min	0 (instantaneous)
<u>Hydrogen evolution and combustion</u>		
Quantity (STP)	< 150 cu ft	3000 cu ft
Energy release	< 48 MW-sec	965 MW-sec
Time to complete (a)	> 260 min	0 (instantaneous)
<u>Decay heat release</u>		
Path	Core ↓ Stagnant water in vessel ↓ Pool water Building air ↓ Building air	Core ↓ Building air (direct)
<u>Mechanical systems operating in RCB after MCA</u>		
Building air conditioning (recirc. refrigeration)	Yes	No
Recirc. air cleanup system	Yes	No

(a) Assuming time zero at instant of rupture.

(b) Of the amount shown, 50% is assumed to deposit out on surfaces within the RCB, and 50% remain available for leakage from the building.

TABLE VII-A-2

SUMMARY OF MCA RADIATION DOSES

<u>Within Exclusion Area</u>	<u>Whole-Body Dose (rem)</u>	<u>Inhalation Dose to Thyroid (rem)</u>
2-hr dose ^(a) at 240 m from AARR	25. ^(b)	---
2-hr dose ^(a) at 350 m from AARR	---	300. ^(b)
2-hr dose ^(a) at center of 40% of Laboratory population (800 m)	3.	150.
2-hr dose ^(a) at boundary (1500 m - entirely within ANL site)	<u>1.0</u>	<u>63.</u>
<u>Within Low Population Zone</u>		
24-hr dose at inner boundary (1500 m)	---	115.
24-hr dose at 2000 m	---	75.
24-hr dose at outer boundary (5000 m)	---	20.
∞ -time dose at 500m (actually within exclusion area)	25. ^(b)	---
∞ -time dose at 2500 m	---	300. ^(b)
∞ -time dose at outer boundary (5000 m)	<u><1.</u>	<u>115.</u>

(a) All 2-hr doses given in this table are the maximum doses for any 2-hr period.

(b) Limiting doses suggested in Code of Federal Regulations, Title 10, Part 100.⁽¹⁾

indicate that there would be ample time available to warn nearby residents and on-site personnel, and to evacuate them if necessary, before limiting doses would be approached.

(3) The MCA computational model takes no credit for mechanical devices operating after the MCA to remove either heat or fission products from the Reactor Containment Building atmosphere. However, the current facility design includes such devices, which would most probably be operable after an MCA and would very significantly reduce the MCA radiation doses below those summarized in Table VII-A-2.

(4) Computation of thyroid inhalation doses by another method, which statistically distributes plume-inhalation doses in accord with measured meteorological data of wind speed, wind direction and the temperature effects that determine atmospheric stability, shows that the doses calculated from the MCA computational model are conservative by a factor of at least three and possibly by as much as a factor of six.

(5) Inhalation doses to the thyroid resulting from the release of contaminated Reactor Containment Building atmosphere during the period of radiation sensing and closing of the containment isolation valves, following a MCA, would amount to a maximum of ~11 rem anywhere in the exclusion area, and a maximum of ~5 rem anywhere in the low population zone. These additional doses are insignificant in comparison with those received from inhalation of the Containment Building leakage plume.

(6) Doses calculated from the MCA computational model are pessimistic in that they are based on a guaranteed maximum containment leakage rate which is a factor of ten higher than the design maximum leakage rate. Reasonable success in attaining the design goal will significantly reduce the MCA doses projected.

(7) Personnel located in AARR buildings other than the Reactor Containment Building, and in other nearby Laboratory facilities are protected by special features which will enable their warning and afford ample time for their evacuation in the event of an accident, before excessive radiation doses are sustained.

(8) It is concluded that the AARR facility may be constructed at its designated location within the boundaries of the DuPage County, Illinois, site of Argonne National Laboratory, and operated at a power of 100 MWt without undue hazard to the general public or to the general Laboratory population.

B. Computational Model for Maximum Credible Accident (MCA) Analysis

In this Section are given the assumed conditions for the Maximum Credible Accident (MCA) for the AARR, and the computational model and parameters used for the analysis of the radiation hazards to Argonne National Laboratory personnel and to the general public.

The only accidents that need be considered for the category of "Maximum Credible Accident" are those involving both a rupture of the primary coolant system and the release of a significant mass of the fission products which are normally contained within the clad fuel plates. Such a release of fission products would occur from the aluminum-clad U_3O_8 -aluminum cermet fuel only if there were melting of the fuel plates.

1. Description and Evaluation of the MCA

The MCA is assumed to be an accident occurring during operation at 100 MWt near the end of the core life (2000 MWd), resulting in the combination of a rapid massive rupture of the primary coolant system and a total melt of the reactor fuel. The MCA analyzed herein is defined by the following assumed series of events:

- (1) There is a massive rupture of the primary coolant system.
- (2) All of the primary coolant water is ejected into the Reactor Containment Building (RCB) at a maximum anticipated reactor vessel outlet temperature of 164°F,* thereby contributing to development of an overpressure in the RCB. Automatic reactor scram occurs as the result of loss of primary coolant pressure. (It should be noted, however, that the assumption of a 164°F effluent temperature for all primary coolant water is equivalent to assuming that the reactor is not shut down by control rod action at the time of the system rupture, but continues to operate at 100 MWt for about one minute, until all of the water has passed through the core at the normal pumping rate and has been released to the building through the rupture. The core would then shut down by loss of moderator.)
- (3) The entire reactor core melts, and there is instantaneous dispersal of fission products into the RCB; of the fission products in the fuel at this time, 100% of the noble gases, 50% of the halogens and 1% of the solids are so dispersed.² Of the halogen fission products released into the RCB, half are assumed to deposit on walls and other surfaces; the other half are considered to be available for leakage from the building.
- (4) 100% aluminum-water and thermite reaction occurs, and the evolved hydrogen ignites and burns; the heat from the primary reactions and from the hydrogen combustion is transferred to the air in the RCB, and contributes to the overpressure developed in the building.

* Since performance of these calculations, the reactor vessel outlet temperature has been reduced to 156°F. Calculated overpressures based upon the 164°F value are thereby rendered additionally conservative.

(5) The entire decay heat generated by the reactor core is released to the RCB atmosphere throughout the duration of the building overpressure condition.

(6) The leakage rate of contaminated air from the RCB is assumed conservatively to vary as the square root of the building overpressure during the entire period of the overpressure condition; the reference point is a guaranteed maximum leakage rate of 1.0% of the building air mass per day at the design overpressure of 7.5 psi.* The leak is assumed to be in the form of a single ground-level leak.

(7) No credit is taken for removal of either decay heat or fission products from the RCB air by mechanical equipment; however, absorption of heat by the building structure, and loss of heat through the building walls, are considered in the calculations.

(8) The RCB ventilation ducts and other penetrations are automatically sealed by isolation valving upon sensing of high radiation levels in the ventilation exhaust system (see Section VII-C-2).

These assumptions clearly result in overly pessimistic results for the initial RCB overpressure. Continuing studies will attempt to incorporate more realistic assumptions as the detailed design of the AARR facility progresses.

In some preliminary safety analysis reports it has been assumed that half or less of the reactor fuel melts during a MCA. At this stage of the preliminary design and safety evaluation of the AARR, neither the extent of possible fuel melting and its subsequent disposition in the RCB nor the extent of possible metal-water reactions can be established quantitatively with any confidence. Therefore, assumptions (3), (4) and (5) were conservatively made.

In computing the radiological effects of the MCA, it was assumed that the fission products released from the molten core were instantaneously and uniformly dispersed throughout the RCB from the moment of core melting, and that the entire leakage from the building was concentrated in a single ground-level leak. In reality, fission product dispersal would take a finite time, and in fact might occur slowly enough that the peak concentration of the products in the building air would not be reached until after the peak overpressure (and leakage) had been passed. Furthermore, it appears likely that the leakage rate would represent the summation of leakage at a number of points around the perimeter of the 120-ft (ID) building and at various elevations from 0 to 90 ft above grade. Leakage from many points would reduce the density of the radiation plume which is postulated to form, in comparison with leakage from a single point. Similarly, in the event of leakage from higher elevations it is reasonable to assume that there would be additional dispersion and dilution due to air currents. Consequently, assumption (6) is considered very conservative.

*Ed. note: The RCB design existing on September 1, 1967 was based on a design overpressure of 5.4 psi with a guaranteed maximum leakage rate of 0.85 w/o per day. See also Question 246, Section III, Part 2 of this report.

2. Overpressure in the Reactor Containment Building

a. Conditions of Analysis

Based upon the MCA model described in Section VII-B-1 above, the overpressure history of the Reactor Containment Building was computed as a function of time, using various assumptions regarding the condition of the Reactor Pool and the path followed by the decay heat in its release from the shut-down core to the Reactor Containment Building atmosphere. The method used for overpressure calculations is presented in Appendix F below.

In the various cases studied, it was assumed that the Reactor Pool either (a) remained intact, or (b) was breached and its contents released to the Reactor Containment Building to mix with the water released from the primary coolant system. The decay heat from the shut-down reactor core was assumed to be released either (a) to the water in the Reactor Pool, and subsequently from the pool water to the Reactor Containment Building atmosphere, or (b) directly to the building atmosphere. In an actual accident, it is likely that either the Reactor Pool would remain intact and the decay heat would be released through the pool, or that the pool would be breached and the decay heat released directly to the Reactor Containment Building atmosphere. However, it was found that the worst case occurs when the Reactor Pool remains intact but the decay heat follows a direct path to the building atmosphere. Therefore, these latter conditions were assumed as the basis for the MCA overpressure calculation, the results of which are summarized graphically in Figures VII-B-1 and VII-B-2.

A further assumption was made for the MCA overpressure calculation that no artificial cooling mechanisms operate to remove heat from inside the Reactor Containment Building during the entire course of the overpressure condition. In the current facility design, the Reactor Containment Building is provided with two independent recirculating cooling systems having a combined heat removal capacity of 1.08×10^6 Btu/hr; these systems normally operate as part of the building air conditioning system (see Section IV-L above). Each system has its own separate chiller unit for transferring heat to the cooling tower, and each system is connected to a separate emergency (Class II-diesel generated) power bus.* It is therefore quite likely that at least one of these two systems would continue to operate during the MCA overpressure period to shorten the duration

*Ed. note: At the time this statement was made, the Title-II design effort was just beginning. Later studies of feasibility and cost of supplying the recirculating unit chillers with emergency power caused a change in philosophy. The September 1, 1967 design provided for emergency (Class-II) power to the service floor recirculating cooler fans only and for the future addition of emergency cooling coils using tower water as the coolant. Since the PSAR approval was sought on the basis of no refrigeration available following the MCA, this change is not believed to be significant.

FIG. VII-B-1
OVERPRESSURE AS A FUNCTION OF TIME (HOURS)
FOLLOWING A MAJOR RUPTURE

FIG. VII-B-2
OVERPRESSURE AS A FUNCTION OF TIME (DAYS)
FOLLOWING A MAJOR RUPTURE

of the period. Consequently, for information only, overpressure history calculations were also performed for the case of one refrigeration system functioning at a heat removal rate of 5.4×10^5 Btu/hr throughout the accident; the results of these computations are also shown in Figures VII-B-1 and VII-B-2.

A summary of some of the conditions assumed in the MCA overpressure analysis is given below. A complete list of the numerical parameters used in the computations is presented in Table VII-B-1.

1. The rupture occurs in the outlet piping of the reactor vessel or in a location such that all of the primary water is expelled at a vessel outlet bulk water temperature of 164°F.*
2. Automatic reactor scram occurs at the start of the accident, and the core releases its shutdown decay heat directly to the building air throughout the duration of the overpressure condition, in accord with Figure VII-B-3.
3. The aluminum-water and thermite reactions release energy directly to the air and instantaneously contribute to the containment overpressure.
4. The combustion of hydrogen evolved from the aluminum-water reaction releases energy directly to the air and instantaneously contributes to the containment overpressure.
5. The primary and secondary emergency cooling provisions are inoperative.
6. There are no means, mechanical or otherwise, operating to remove heat from the Reactor Containment Building other than normal (natural) heat losses. (A separate analysis was made for the one-half capacity refrigeration case mentioned above.)
7. The equipment and internal structures within the Reactor Containment Building not a part of the pool enclosure or of the exterior walls, basement floor, or dome are lumped together and defined as one mass with an average specific heat, density, conductivity, and surface coefficient of heat transfer, and an equivalent thickness and surface area. These internals store heat delivered from the containment atmosphere.
8. Heat is transferred from the containment atmosphere to all surfaces with which it is in contact, and these surfaces include the internals, dome, wall above grade, wall below grade, basement floor, concrete pool enclosure, spilled water on the basement floor, and pool water.

* See footnote, p. 277.

TABLE VII-B-1

PARAMETERS USED IN OVERPRESSURE CALCULATIONS

<u>PARAMETER</u>	<u>VALUE</u>
Initial weight of dry, containment air	82,871 lb
Initial weight of water vapor inside containment at 50% relative humidity	906 lb
Initial weight of Reactor Pool water	2.346×10^6 lb
Initial weight of spilled primary system water	2.15×10^5 lb (a)
Initial temperature of containment atmosphere	80°F
Initial temperature of Reactor Pool water	85°F
Initial temperature of spilled primary system water	164°F(b)
Initial temperature of all containment conducting structures, including internals, pool structure, dome, exterior walls, and basement floor	80°F
Constant, outside atmospheric temperature	85°F
Constant, outside atmospheric pressure	14.3 psia
Energy release from aluminum-water reaction	1,270 MW-sec
Energy release from thermite reaction	38 MW-sec
Energy release from hydrogen combustion	965 MW-sec
Reactor operation prior to accident	22 days at 100 MW
Containment volume	1.178×10^6 ft ³ *
Leakage rate	1 w/o/day at 7.5 psi **
Effective areas for convective heat transfer between:	
a. the containment atmosphere and the dome	15,600 ft ²
b. the containment atmosphere and the wall above grade	16,200 ft ²
c. the containment atmosphere and the wall below grade	18,500 ft ²
d. the containment atmosphere and the pool enclosure	5,600 ft ²
e. the containment atmosphere and the equivalent internals	37,500 ft ² (each of two sides)
f. the containment atmosphere and the pool water	500 ft ²
g. the containment atmosphere and the spilled primary water	11,300 ft ²
h. the pool water and the pool enclosure	5,600 ft ²
i. the spilled primary water and the basement floor	11,300 ft ²

TABLE VII-B-1 (continued)

<u>PARAMETER</u>	<u>VALUE</u>
Convective heat transfer coefficient between:	
a. the containment atmosphere and all surfaces with which it comes in contact	0.5 Btu/hr-ft ² -°F
b. the water inside the containment and all solid surfaces with which it comes in contact	1.0 Btu/hr-ft ² -°F
c. the outside atmosphere and the dome and wall above grade	1.0 Btu/hr-ft ² -°F
Conducting subsystem thicknesses and compositions:	
a. dome	3/8" steel and 1' concrete
b. wall above grade	3/8" steel and 2' concrete
c. wall below grade	3/8" steel and 2'-3" concrete +
d. basement floor	6' concrete
e. equivalent internals	0.37' concrete
f. pool enclosure	2' concrete
Heat capacities	
a. concrete	0.13 Btu/lb-°F
b. steel	0.11 Btu/lb-°F
Conductivities:	
a. concrete	1 Btu/hr-ft-°F
b. steel	26 Btu/hr-ft-°F
Densities:	
a. concrete	144 lb/ft ³
b. steel	485 lb/ft ³

(a) Since performance of these calculations, the water content of the primary system has been reduced to $\sim 1.61 \times 10^5$ lb. Calculated overpressures based upon the 2.15×10^5 lb value are thereby rendered additionally conservative.

(b) Since performance of these calculations, the reactor vessel outlet temperature has been reduced to 156°F. Calculated overpressures based upon the 164°F value are thereby rendered additionally conservative.

* Ed. note: Containment volume was reduced to 1.08×10^6 ft³. (see Section IV-J above).

** Ed. note: The RCB design existing on September 1, 1967 was based on a design overpressure of 5.4 psi with a guaranteed maximum leakage rate of 0.85 w/o per day. See also Question 246, Section III, Part 2 of this report.

+ Ed. note: The wall below grade was changed to 2'-6" thick (see Section IV-J above).

FIG. VII-B-3

INTEGRATED DECAY HEAT AS A FUNCTION OF TIME
(AARR CORE OPERATED AT 100 MW FOR 22 DAYS)

9. The dome and the wall above grade transfer heat from the containment atmosphere to the outside atmosphere via convection and conduction. The wall below grade and the basement floor are assumed to be perfectly insulated from the surrounding soil, but they store heat delivered from the containment atmosphere or from water inside the containment.

10. Leakage from the containment varies as the square root of the overpressure based upon a guaranteed leakage rate of 1.0 wt% per day at 7.5 psig.*

It may be possible, as calculation techniques are further refined, analytical models are further improved, and the AARR plant, system, and component designs are finalized by the architect-engineer, to justify and support the selection of less stringent assumptions for overpressure calculations. If such is the case, this will be reflected in the Final Safety Analysis Report.

b. Results and Discussion

An essentially instantaneous overpressure of 3.9 psi develops from energy released from the aluminum-water and thermite reactions, and hydrogen combustion. The spilled primary water releases sensible heat and water vapor to the containment atmosphere over a period of time and, during this time, decay heat is also being added to the containment atmosphere. The total of the heat additions exceeds the total of the heat losses in the first few minutes following the accident. This, together with the addition of water vapor to the containment atmosphere, causes the overpressure to increase to a maximum of 4.4 psi 5 min after the accident. If refrigeration is applied at a rate equal to one-half the normal containment cooling capacity, then the heat losses are thereby increased, and the overpressure rises to a maximum of only 4.3 psi in 4 min. After an interval of 5 min following the accident (or 4 min with refrigeration), the total of the heat losses from the containment atmosphere exceeds the total of the heat addition, and, aided by leakage, the overpressure begins to decrease. This decrease eventually reduces the overpressure to zero since the total of the heat additions to the containment atmosphere never again exceeds the total of the heat losses. The overpressure reaches zero after 28 days if no artificial cooling is provided. However, it reaches zero in only 8 days if one-half capacity refrigeration is applied to the containment atmosphere continuously from the start of the accident.

c. Justification of Parameter Values Used

It must be noted that the preceding analysis was based on design information associated with an aluminum core. However, some of the parameter values listed may differ slightly from other reported values due to late revisions of the design which, because of time limitations, could not be incorporated into the analysis.

The temperatures given in Table VII-B-1 are the steady-state temperatures which were assumed to exist before the accident occurred.

*Ed. note: See footnote, p. 278

Following the accident, some of these temperatures changed with time from the steady-state values which existed before the accident. The outside atmospheric temperature was assumed to remain constant at all times before, during, and after the accident. The value chosen for this temperature was a conservative, effective average for day and night atmospheric temperatures.

The air and water vapor were assumed to have the same (but not constant) temperature at all times. Thus, liberated heat was properly apportioned to the air and water vapor so they did maintain the same temperature.

All water in the system including the Reactor Pool water and the spilled primary water, was available as a heat sink. However, in the cases of large surface area associated with spillage, the water was a more effective heat sink than when it was retained in the pool, which has a relatively small surface area.

The values for the convective heat transfer coefficients listed in Table VII-B-1 were inferred from Reference 3, which states that, for most engineering applications, the free convection heat transfer coefficient in air has a typical value of from 1 - 5 Btu/hr-ft²-°F. Therefore, a conservative value, compared to this range, of 0.5 Btu/hr-ft²-°F was chosen for the value of the heat transfer coefficient between the containment atmosphere and the surfaces with which it is in contact. A coefficient for heat transfer communication between water and the solid surfaces with which it is in contact was taken to be 1.0 Btu/hr-ft²-°F. Since wind action increases the value of the heat transfer coefficient on the outer surfaces of the containment building that are in contact with the outside atmosphere, a larger value of 1.0 Btu/hr-ft²-°F was chosen for this heat transfer coefficient.

The heat transfer coefficients chosen for use were checked using the method shown in Reference 4, which states that the coefficients are functions of the temperature difference which exists. Representative temperature differences were used to show that the values chosen for the coefficients were conservative.

d. Hydrogen Combustion

The heat which was assumed to be released instantaneously to the contained atmosphere via hydrogen combustion contributed 1.7 psi to the overpressure. At 30 min following hydrogen combustion, the overpressure was only 0.2 psi more than if the hydrogen had not burned, since the heat released was rapidly dissipated to various heat sinks and to the surroundings. Thus, the effect on overpressure was transitory, and this effect would be essentially the same no matter when the hydrogen burned.

In the preceding analysis the evolved hydrogen was assumed to deliver all its combustion heat directly to the containment atmosphere at the instant of primary system rupture. However, if the hydrogen burns and the resulting combustion heat is instantaneously delivered to the containment atmosphere at the time of highest overpressure (3.7 psi at 16 min following the accident if the hydrogen has not yet burned), the overpressure would then instantaneously rise to 5.3 psi. It would rapidly drop to a

value of 4.3 psi at 26 min following the accident (or 10 min following the addition of all the combustion heat).

3. Fission Product Inventories in the Reactor Containment Building

The HFIR core has been operated for as long as 2200 MWd at a power level of 100 MW. In terms of its contribution to reactivity, the AARR beryllium reflector will be less effective than that of the HFIR, because of the presence of a larger number of streaming ducts (beam tubes). It is estimated that the total core life will be less than 20 days at 100 MW. Therefore, the fission product inventory has been based upon the end-of-life conditions after 2000 MWd of continuous operation at full power.

It is assumed that all the reactor fuel melts, and that there is an instantaneous uniform dispersal of fission products into the Reactor Containment Building (RCB). The model is: of the fission products in the fuel at this time, 100% of the noble gases, 50% of the halogens, and 1% of the solids are so dispersed. Of the halogen fission products released into the RCB, half are assumed to deposit on walls and on other surfaces; the other half are considered to be available for leakage from the building. Table VII-B-2 is a listing of the total inventories of the important fission product isotopes of iodine (I^{131} through I^{135}) in the air of the RCB immediately following the postulated contamination by an MCA fuel melt.

Table VII-B-2

Postulated Air Inventories of Fission Product Iodines in Reactor Containment Building after MCA Fuel Melt

<u>Iodine Isotope</u>	<u>Total Inventory in RCB Air (curies)</u>
I^{131}	5.15×10^5
I^{132}	9.52×10^5
I^{133}	14.06×10^5
I^{134}	16.44×10^5
I^{135}	12.76×10^5

Although a portion of the RCB is below grade, no credit has been taken for this in computing the direct gamma-ray dose from the building.

The Title-I design includes a 5000-cfm* air-exhaust line, with an efficient air filtration system. This component of the RCB ventilation system could be used to rapidly reduce air-activity levels in the RCB, by recirculation of air within the isolated building. The effectiveness of this system for this purpose will be evaluated during the Title-II design.

*Ed note: This was revised to a 4000-cfm system (See Section IV-L above).

In the present computations of radiation doses to receptors outside the RCB, cleanup of RCB contamination is not included.

4. Escape of Fission Products from the Reactor Containment Building

It is considered to be incredible that the containment of the RCB would be breached coincidentally with the occurrence of a MCA. There remain two avenues of escape of fission products from the RCB: (1) through the 250-ft stack before closure of the containment-isolation valves; and (2) through the walls of the RCB, by leakage. In Section VII-C-2, a conservative analysis is given of the incremental radiation doses which might result from release of fission products before the containment-isolation valves are fully closed and the RCB is isolated. The leakage rate of fission products through the isolated RCB depends upon the level of building overpressure, which, in turn, depends upon the detailed sources and sinks of heat in the RCB following the initiation of the MCA. The time-dependent leakage rate of contaminated air from the RCB is assumed conservatively to vary as the square root of building overpressure. The reference point is a leakage rate of 1.0% of the building air mass per day at the design overpressure of 7.5 psi.*

The basic, or "reference" model for the variation in building air pressure subsequent to the MCA fuel melt was discussed in Section VII-B-2 above. In that model, no credit was taken for reduction in overpressure by operation of auxiliary devices, e.g., an air-refrigeration system. In the Title-II design, various systems which would be effective in limiting the escape of fission products from the RCB will be evaluated. In Section VII-C-1, therefore, for illustrative purposes a comparison study is presented of the computed radiation doses for two cases: (1) the "reference" model, wherein only natural heat losses are assumed; and (2) a model wherein an air-refrigeration system is operative. Indeed, it is conservatively assumed that only one-half of the full refrigeration capacity is in use. This second model will be referred to as the "refrigeration case".

The calculations of radiation dose were simplified by conservative step-function approximations to the calculated time-dependent leak rates from the RCB. These fittings are shown in Figures VII-B-4 and VII-B-5.

5. Use of the Reactor-Siting Program AISITE II

A digital-computer program, AISITE II,⁵ has been employed for the basic computations of radiation doses to various organs of the body, as a function of total time of exposure and as a function of distance. The program is based upon: (1) the model of J. J. Di Nunno, *et al* (TID-14844),² in computing the isotope inhalation dose to an organ; (2) the model of WASH-740,⁶ in computing the direct dose from the escape plume; and (3) the formulation by F. Pasquill⁷ of categories of transport and diffusion characteristics of the plume.

The reference report (Reference 5) for the AISITE program lists meteorological parameters for only one category - the Pasquill Category F. The code has been modified at ANL to allow for the optional use of Category E meteorological data instead.⁷ The parameters used for these two meteorological categories of plume diffusion are listed in Table VII-B-3.

*Ed note: See footnote, p. 278

FIG. VII-B-4
STEP-FUNCTION APPROXIMATION TO TIME-DEPENDENT
LEAKAGE RATE FROM REACTOR CONTAINMENT BUILDING
FOLLOWING A MCA (REFERENCE CASE)

FIG. VII-B-5
STEP-FUNCTION APPROXIMATION TO TIME-DEPENDENT
LEAKAGE RATE FROM REACTOR CONTAINMENT BUILDING
FOLLOWING A MCA (REFRIGERATION CASE)

TABLE VII-B-3

PARAMETERS ASSIGNED TO METEOROLOGICAL CATEGORIESE AND F(INPUT TO AISITE II)

DISTANCE FROM POINT SOURCE (m)	CATAGORY E		CATAGORY F	
	$\theta(a)$ (degrees)	$h(a)$ (m)	$\theta(a)$ (degrees)	$h(a)$ (m)
100	15.0	7.7	10.00	4.9
322	14.5	20.0	9.20	12.0
644	14.2	37.0	8.65	20.0
804	13.8	43.0	8.50	25.0
965	13.5	49.0	8.37	29.0
1130	13.0	54.0	8.25	31.0
1290	12.6	58.0	8.15	33.0
4820	11.8	129.0	7.18	70.0
12850	9.2	194.0	6.50	110.0
19300	9.1	237.0	5.92	135.0

(a) The parameters θ and h are functions of the downwind distance of the plume from the point source. θ is the lateral cloud spread, in degrees; h is the vertical cloud spread, in meters.

The AISTE-II program includes a correction for decay of fission products in the plume during the time of travel of the plume front to the location of the receptor. However, it does not correct for the reduction in actual exposure during a given time interval due to the plume travel time; for example, with an assumed wind speed of 1 m/sec, for the first 25 min a receptor at 1500 m would receive essentially zero plume-inhalation dose, and the dose for the first two hours following the accident thereby would be reduced. Two sets of calculated doses to the thyroid are presented in Section VII-C-1 below for exposure periods of 2 hr; (1) the plume-inhalation dose received during the first 2 hr; and (2) the maximum dose received during any 2 hr time period. The listed doses to the body and to the bone are the maximum doses received during any 2 hr time period. At distances where the fractional corrections are sizeable, the total dose is small.

In calculating the net plume-inhalation doses to the thyroid, additional correction factors were applied for: (1) the dilution of the plume by the turbulence generated in the lee of the RCB (the "building-wake factor"); and (2) the meandering of the plume, thereby presenting average plume concentrations to the receptor during significant time intervals (15 min or more). Conservatively, neither of these two correction terms was applied to the calculated radiation doses to the whole body or to the bone because significant fractions of these doses are due to noninhalation doses from the plume. A detailed discussion of these two correction terms is given in Section VII-B-7 below.

The AISITE-II output separately lists the direct dose from the RCB and the total dose (sum of the plume dose and the direct dose from the RCB). No provision is included for time-dependence of leakage rate. Therefore, the plume dose rate as a function of time was determined from AISITE by appropriate calculation of incremental doses.

AISITE-II does not calculate the very significant attenuation of the direct gamma rays from the RCB by the thick steel and concrete building wall. It is computed that the direct gamma-ray dose rate from the Reactor Containment Building is reduced by factors of from 185 to 600 with a shield-equivalent of a 24-in. thickness of ordinary concrete. The actual value of the attenuation factor within this range depends upon the length of time after the fuel melt and upon the distance to the receptor. These detailed calculations were performed earlier, for a case of operation at 100 MW for a period of 90 days. Comparison calculations for cases of 20-day operation and 90-day operation, with AISITE-II, indicate that the 2 hr direct doses from the RCB are almost identical in magnitude. The total direct doses from the RCB which would result from exposure of receptors for 30 days differ by approximately 25% at distances close to the RCB, and the fractional error lessens with increasing distance. Since the attenuated direct dose is a small fraction of the total dose to the body organs, except perhaps within the first few hundred meters, the conservative values of the direct dose as determined for the 90-day case are listed. In these calculations it is assumed conservatively that the entire fission product inventory is dispersed in the above-grade portions of the RCB.

6. Choice of Atmospheric-Dispersion Parameters

In computing the 2 hr radiation doses, it was assumed that the dispersion parameters of Category F (moderately-stable conditions) apply, with an average wind speed of 1 m/sec.

In computing the total radiation dose exposures for periods of one day or longer, it was assumed that meteorological Category E (slightly-stable conditions) apply for the entire time, with an average wind speed of 2 m/sec. No account was taken of reductions in plume-concentrations from: rainstorms; gravity deposition of solids; or gross dispersion of the radiation plume by large variations in wind direction, higher wind speeds, and diurnal variation of atmospheric stability. (Typically, less stable conditions occur during daylight hours.) As shown in Section VII-D below, the reductions in total dose because of such variations in meteorological conditions are substantial. The actual doses which would be received over long periods of time would be much smaller than the computed doses, for the assumed constant conditions (Category E; wind of constant direction; and average wind speed of 2 m/sec), which are given in Table VII-C-1 and Figs. VII-C-3 to VII-C-5 of Section VII-C below.

It is difficult to compare average weather data measured at Argonne National Laboratory with the meteorological conditions characterized by Categories E and F. These conditions typically occur at night, in conjunction with surface wind velocities of 1 m/sec to 5 m/sec, and with certain parameters of cloud cover. Cloudiness is not recorded at Argonne, but data for a different indicator of atmospheric stability are available, namely, a joint frequency distribution of wind speed (at an elevation of 150 ft) and of the temperature differential between the ground and the elevation of 144 feet. (See Table VII-B-4.) Stable atmospheres are characterized by the existence of higher temperatures above ground level than at ground level. Category D conditions, which apply to less-stable atmospheres than those in Categories E or F, are characterized by Meade and Pasquill as "neutral".

Referring to Table VII-B-4, note that temperature-inversion conditions occur at Argonne approximately 45% of the time. An indication of the average wind speeds to be expected under these conditions is provided in Table VII-B-5. Insofar as effects of ground-level leakage are concerned, it seems reasonable to assume that the radiation plume travels at a speed represented by the data available for wind speed at an elevation of 19 ft. It is estimated that atmospheric conditions which are more unfavorable than Category F plus a wind speed of 1 m/sec would occur less than 10% of the time, and that even the degree of atmospheric stability assumed for the computation of the 30-day doses (Category E, with a wind speed of 2 m/sec) would be expected appreciably less than 20% of the time.

Therefore, the atmospheric-stability models employed in the calculations of the 2 hr doses (Category F; 1 m/sec), and of the long-period doses (Category E; 2 m/sec) appear to be reasonable.

TABLE VII-B-4 (a)

PERCENTAGE FREQUENCY DISTRIBUTION OF THE 150-ft WIND SPEEDS AND THE 144-5.5-ft TEMPERATURE DIFFERENCE

JULY 1951 - DECEMBER 1964

Temperature Difference, ΔT , °F	Wind Speed, mph											Total	Cumul Total
	Calm	1-3	4-7	8-12	13-18	19-24	25-31	32-38	39-46	> 46	Missing		
≤ -3.0	0.003	0.063	0.443	0.966	0.867	0.285	0.057	0.009	0.000	0.000	0.003	2.695	2.695
-2.9 to -2.0	0.035	0.374	1.726	3.083	2.309	0.738	0.155	0.024	0.003	0.000	0.026	8.474	11.169
-1.9 to -1.0	0.095	0.783	2.670	4.428	3.382	1.276	0.345	0.057	0.007	0.000	0.055	13.096	24.265
-0.9 to 0.0	0.189	0.731	3.123	8.588	8.668	3.907	1.136	0.228	0.034	0.002	0.182	26.788	51.053
0.1 to 1.0	0.050	0.208	1.117	4.274	4.607	1.409	0.226	0.024	0.003	0.000	0.051	11.967	63.020
1.1 to 2.0	0.044	0.151	0.945	3.797	2.721	0.392	0.030	0.002	0.000	0.000	0.030	8.113	71.133
2.1 to 3.0	0.041	0.080	0.577	2.733	1.308	0.056	0.002	0.000	0.000	0.000	0.009	4.805	75.938
3.1 to 4.0	0.039	0.096	0.543	2.536	1.087	0.029	0.000	0.000	0.000	0.000	0.008	4.338	80.277
4.1 to 6.0	0.062	0.179	0.829	2.762	1.122	0.023	0.003	0.001	0.000	0.000	0.022	5.002	85.279
6.1 to 8.0	0.059	0.149	0.579	1.332	0.448	0.006	0.000	0.000	0.000	0.000	0.013	2.585	87.864
> 8.0	0.104	0.233	0.768	0.966	0.236	0.007	0.000	0.000	0.000	0.000	0.004	2.318	90.182
Missing	0.081	0.349	1.285	3.496	2.971	1.104	0.299	0.054	0.007	0.000	0.172	9.818	100.000
Total	0.80	3.40	14.60	38.96	29.72	9.23	2.25	0.40	0.05	0.00	0.58	100.0	
Cumul Total	0.80	4.20	18.80	57.76	87.49	96.72	98.97	99.37	99.42	99.42	100.00		

(a) This tabulation is an as yet unpublished updating, to include data for additional years, of Table 238 of Reference 8. The total number of observations is 118,392.

Table VII-B-5

Average Wind Speeds at the 19-ft and 150-ft
Levels as a Function of Inversion Severity*

Temp. at 144-ft Level Minus Temp. at 5.5-ft Level (°C)	Frequency (% of Total Time)	Average Wind Speed (m/sec)	
		19-ft level	150-ft level
0 - 1.5	28	3	6
1.5 - 3.0	10	1-3/4	5
>3.0	8	1/1/4	4-1/2

*Ed. note: Data in this table are based on meteorological data in reference (8) and not on data in Table VII-B-4.

7. Building-Wake Factor and Plume Meander

When air flows around a large obstruction such as the Reactor Containment Building, a turbulent wake is produced on the lee side. Because of the rapid mixing in this region, materials leaking from the RCB are diffused quickly. According to J. J. Fuquay,⁹ the material leaking in one second is diluted into a volume given numerically by the product:

$$D_B = cAu \quad ,$$

where

A is the cross sectional area of the building normal to the direction of the wind, m²

u is the wind speed, m/sec

and

c is a diffusion coefficient, with values in the range 1/2 to 2. In the analyses for the AARR, the conservative value c = 1/2 has been used.

In calculating the concentration of materials at points downwind from the building, the building-wake dilution volume, $D_B = cAu$, is incorporated in the following manner:

$$X = \frac{Q}{\pi u \sigma_y \sigma_z + cAu} \exp \left[-\frac{y^2}{2\sigma_y^2} - \frac{z^2}{2\sigma_z^2} \right] .$$

In the equation above:

- X is the concentration, in grams or curies per cubic meter
 Q is the emission rate of the material, in grams or curies per second
 y, z are the lateral and vertical coordinates of the receptor, respectively

and

- σ_y, σ_z are the standard deviations of concentration according to the Pasquill-Gifford diagram.¹⁰

The "building-wake factor", R , is defined by:

$$R \equiv \frac{\pi \sigma_y \sigma_z}{\pi \sigma_y \sigma_z + cA} .$$

Thus R is the ratio of the material concentration, with building-wake dilution included, to the concentration which would exist if there were no building-wake effects. The variation of the building-wake factor with distance from the RCB is given in Fig. VII-B-6, for meteorological Categories E and F. It is applied directly as a distance-dependent correction factor to the plume-inhalation dose to the thyroid which is inferred from the AISITE-II output.

In the AISITE-II analyses, the receptor is assumed to be positioned at the centerline of the plume. As a result of the eddy diffusion process, the concentration of material in the plume resulting from building leakage is distributed according to a Gaussian function, i.e., the concentrations in any section perpendicular to the centerline may be represented by a bivariate normal distribution. For wind speeds of 1 m/sec or greater, the atmosphere is usually turbulent and the plume nearly always exhibits a degree of meander. The concentration experienced by a receptor at a fixed point therefore is represented by an average of the concentrations. Assuming that the edges of the plume are truncated at a value of one-tenth of the maximum concentration, the average concentration is 0.57 times the maximum. Therefore, a correction factor of 0.6 has been applied directly to the plume-inhalation dose to the thyroid inferred from AISITE-II.

FIG. VII-B-6
BUILDING-WAKE FACTORS FOR MEADE-PASQUILL METEOROLOGICAL CATEGORIES E AND F

C. MCA Radiation Doses Outside the Reactor Containment Building

1. Doses from Direct Building Radiation and Leakage Plume

Radiation doses from direct gamma-ray radiation from the RCB and from radiation effects of the plume from ground-level leakage are summarized in Table VII-C-1 and in Figs. VII-C-1 through VII-C-5. All distances are measured from the center of the Reactor Containment Building. This is equivalent to the approximation that all direct radiation is emitted from a point source at the center of the RCB, and that the leakage plume arises from leakage at this point. This approximation does not lead to significant error at distances large in comparison with 20 m, which is approximately the value of the radius of this building in horizontal plan view. Care must be taken in extrapolating the tabulated data to points closer to the RCB.

Figure VII-C-1 graphically displays: the distance-dependent 2 hr plume doses to the body and the bone; and the maximum 2 hr plume-inhalation dose to the thyroid, the maximum being taken over all 2 hr dose periods following the initiation of the MCA. This maximum dose to the thyroid is presented in Table VII-C-1 as "the 2-hour dose", rather than the usual value of dose in the first two hours. The difference, of course, arises from the time required for the plume to travel to the position of the receptor.

In Fig. VII-C-2, the 2 hr inhalation dose to the thyroid is graphed as a function of the initial time of the 2 hr dose period. Two distances have been chosen, 800 m and 1500 m, to illustrate the effect of plume-travel time. The effect of the plume lag is quite large, at large distances from the RCB. At 5000 meters, for example, with an assumed wind speed of 1 m/sec, 83 min of the first two hours are consumed by the plume travel time. Of course, the ∞ -time doses are little affected by plume travel times; nevertheless, this effect has been included in the calculation of the ∞ -time inhalation dose to the thyroid. Note that a constant RCB leak rate of 0.75%/day has been assumed for the calculations of 2 hr doses, whether or not an air-refrigeration system is in operation following the MCA fuel melt.

As discussed in Section VII-B, various auxiliary systems which would be effective in reducing the long-period MCA doses to receptors outside the RCB will be evaluated in the Title-II phase of the design work, e.g., an air-refrigeration system. The effects of such a system on these doses are summarized in Figures VII-C-3 through VII-C-5. In these analyses, the refrigeration system which is conceived is assumed to be operative at only half-capacity. Figure VII-C-3 shows the accumulated MCA inhalation dose to the thyroid at each of three distances from the RCB, 1500 m, 2000 m and 5000 m, as a function of the total number of days of continuous exposure to the passing plume. Curves are presented both for the reference case (no air refrigeration) and for the case of RCB air refrigeration at one-half of refrigeration capacity. For the reference case, at 5000 m, which is the designated distance to the outer boundary of the low-population zone (see Criterion 19.1c, Section II), the constant-conditions value of the ∞ -time dose to the thyroid is approximately 115 rem. This conservatively-computed dose is only slightly more than 1/3 of the value of the emergency dose (300 rem). The effect of half-capacity RCB air refrigeration would be to

TABLE VII-C-1

RADIATION DOSES FOLLOWING A MCA

Dose Period	Distance (m)	Whole-Body Doses (rem)			Bone Doses (rem)			Inhalation Dose to the Thyroid (rem)
		Direct Dose from RCB ^(a)	Total Plume Dose	Total	Direct Dose from RCB ^(a)	Total Plume Dose	Total	
2 hours ^(b)	100	16.	104.	120.	16.	290.	306.	334.
	200	3.	32.	35.	3.	85.	88.	315.
	500	0.1	6.8	6.9	0.1	16.5	16.6	220.
	1000	---	2.1	2.1	---	4.9	4.9	105.
	1500 ^(c)	---	1.0	1.0	---	2.5	2.5	63.
	2000	---	0.6	0.6	---	1.5	1.5	43.
	5000 ^(c)	---	0.15	0.15	---	0.35	0.35	12.
30 days ^(d) (Reference Case: No RCB Refrigeration)	200	21.	127.	148.	21.	1150.	1171.	6000.
	500	0.9	25.	26.	0.9	225.	226.	2500.
	1000	---	7.7	7.7	---	68.	68.	1050.
	1500 ^(c)	---	4.0	4.0	---	35.	35.	600.
	2000	---	2.5	2.5	---	22.	22.	410.
	5000 ^(c)	---	<1.	1.	---	5.8	5.8	115.
	10000	---	<1.	1.	---	2.3	2.3	45.
30 days ^(d) (Refrigeration Case)	200	21.	56.	77.	21.	330.	351.	3100.
	500	0.9	11.	12.	0.9	64.	65.	1200.
	1000	---	3.4	3.4	---	20.	20.	490.
	1500 ^(c)	---	1.8	1.8	---	10.2	10.2	280.
	2000	---	<1.	<1.	---	6.5	6.5	195.
	5000 ^(c)	---	<1.	<1.	---	1.7	1.7	55.
	10000	---	<1.	<1.	---	<1.	<1.	21.

(a) The tabulation of values for the direct dose from the RCB is based on very detailed calculations, including the large attenuation by the walls of the RCB, for the case of a reactor operation at 100 MW for 90 days. This is discussed in Section VII-B-5. The attenuated direct dose from the RCB is a small fraction of the total dose.

(b) All 2-hour doses are based on: a constant RCB leak rate of 0.75%/day; Category F meteorological conditions; constant gross wind direction; and wind speed of 1 m/sec. The inhalation dose to the thyroid has been corrected for building-wake dilution and plume meander (see Section VII-B-7). At each distance, this dose to the thyroid is the maximum for all 2-hr dose periods following the initiation of the MCA (see Figure VII-C-1). The plume doses to the body and to the bone have not been modified for: building-wake dilution; plume meander; or plume travel time.

(c) The exclusion area and low-population zone radii, as defined in Criterion 19.1, Section II above, are 1500 m and 5000 m, respectively.

(d) All 30-day doses are based on: appropriate time-dependent leak rate from the RCB (see Section VII-B-4); Category E meteorological conditions; constant gross wind direction; and wind speed of 2 m/sec. The inhalation dose to the thyroid has been corrected for building-wake dilution and plume meander; it is the ∞ -time dose ignoring "breathing" effects of the RCB after the overpressure is nullified.

FIG. VII-C-1

MCA PLUME DOSES TO BODY ORGANS FOR TWO-HOUR DOSE PERIODS
 (CONSTANT LEAK RATE: 0.75%/day;
 CAT. F; WIND SPEED: 1 meter/sec)

FIG. VII-C-2

MCA PLUME-INHALATION DOSE TO THE THYROID AS A FUNCTION
OF TIME OF INITIATION OF TWO-HOUR DOSE PERIOD
(CONSTANT LEAK RATE: 0.75%/day;
(CAT. F; WIND SPEED: 1 meter/sec)

FIG. VII-C-3
MCA PLUM-INHALATION DOSE TO THE THYROID AS A FUNCTION
OF TOTAL TIME OF EXPOSURE

FIG. VII-C-4

MCA PLUME-INHALATION DOSE TO THE THYROID
AS A FUNCTION OF DISTANCE FROM THE
REACTOR CONTAINMENT BUILDING

FIG. VII-C-5
MCA ∞ -TIME PLUME DOSES TO BODY ORGANS
(CONSTANT CONDITIONS; CAT. E;
WIND SPEED: 2 meters/sec)

reduce the dose to the thyroid by another 50%. (See also Table VII-C-1.)

Figures VII-C-4 and VII-C-5 include other graphs of various radiation doses of particular importance in the analysis of site acceptability. The doses given are for the reference case, and, for comparison, also for the refrigeration case; the curves show the rate of dropoff in dose level with increasing distance from the Reactor Containment Building.

2. Inhalation Doses due to Stack Release Prior to Isolation of the Reactor Containment Building

Following the fuel melt postulated in the MCA model, there will be some dispersal of fission products into the bulk air volume of the Reactor Containment Building. It is expected that the rate of increase in air contamination would be small enough that the radiation monitors in the building ventilation system would detect a potentially dangerous level of activity, and there would be a rapid automatic actuation of isolation valves to close off the RCB from the stack long before the activity of the air in the exhaust line reached the level postulated from the MCA. In this realistic case, the stack release would be a negligible contribution to the total radiation dose to people outside the RCB - much less than an inhalation dose of 1 rem to the thyroid, at any distance.

A brief description of the containment isolation system is given in Sections IV-H and IV-L above. The guiding philosophy for this system is discussed in Section VI-E-2 above, which includes an analysis of the very low radiation doses which would be expected if the maximum activity in the exhaust air only slightly exceeded the level of the high set-point of the RCB isolation system.

Firm, realistic criteria for the design actuation and closure times for the RCB isolation valves cannot be set until further detailed study of valve types has been carried out as part of the Title-II design effort. However, it is expected that these criteria will be governed by the principle that for receptors outside the RCB the total incremental plume-inhalation dose to the thyroid due to release of radiation during the process of isolation of the RCB will be limited to less than 25 rem at any distance up to the exclusion area radius (1500 m), and less than 10 rem at any distance beyond this radius, for any conditions up to and including the case of the MCA. Under these limitations, the incremental dose due to such release would not be a significant addition to the total radiation dose received by people outside the RCB from direct radiation or leakage-plume inhalation following a MCA. (The weighted incremental inhalation dose to the thyroid far outweighs the incremental dose to other organs.)

As stated above, in the realistic case, stack release would contribute only a very small amount to the total radiation dose received by persons outside the RCB after a MCA, because of the time required for fission product dispersion throughout the building. For conservatism, however, it is assumed in this analysis that the rate of release of activity following a MCA-type melt is extremely rapid, and that not even seconds of time elapse before the RCB air is massively contaminated. It is assumed

further that a total of 25% of the reactor inventory of each iodine isotope is dispersed uniformly into the bulk air volume ($\sim 1.2 \times 10^6 \text{ ft}^3$)* of the RCB. With the present design concept of radiation monitoring and isolation valving, within two seconds after this contaminated air reaches the radiation monitors in the building ventilation system there is a signal to isolate the RCB from the stack. Within two seconds after this the RCB is isolated. In the following analysis, it is assumed that the total stack release of unfiltered air corresponds to 3 sec of full flow at 15,000 cfm. This is a total volume of 750 ft^3 , or 0.06% of the total air volume of the RCB, containing 0.015% of the total reactor inventories of the iodines.

The calculation of the atmospheric dispersion of activity resulting from release of 750 ft^3 of air was based on the generalized Gaussian plume formula for a continuous point source,¹¹ for a total release time of 3 sec. No one standard meteorological category is worst for all distances. Therefore, at each distance a category was selected which yielded the largest radiation dose. Of the various categories, Category A is worst for short distances from the RCB. In the range from 1500 m to 4000 m, Category D appears to be the most pessimistic. Beyond that distance, until roughly 10,000 m, Category E is worst; and, beyond that, Category F is worst. All of these statements are based on an assumed effective stack height of 76.4 m (250 ft), and wind speed of 1 m/sec.

Postulating Category A conditions, the maximum air concentration (curies/ m^3) at ground level occurs at $\sim 300 \text{ m}$. There, numerically, it is $\sim 3\text{--}1/2 \times 10^{-5}$ times the stack radiation-release rate (in units of curies/sec). The corresponding total plume inhalation dose to the thyroid was calculated on the basis of the tabulation of infinite-time dose to the thyroid per iodine curie inhaled, in TID-14844.¹² For these conditions, the maximum incremental inhalation dose to the thyroid is 11 rem. This is the maximum dose within the range to 1500 m, regardless of the choice of meteorological category. Between 1500 m and 5000 m, the largest incremental inhalation dose to the thyroid is 5 rem (Category D). Beyond 5000 m, the largest incremental dose is 2 rem; this is obtained using Category F and a wind speed of 1 m/sec.

3. Effects of Forced Ventilation of the Reactor Containment Building after a MCA

It is a basic assumption of the MCA model that the Reactor Containment Building is promptly isolated from the stack, and that thereafter no use is made of the stack to reduce the overpressure and the fission product contamination of the RCB. However, some preliminary analyses have been made of the radiation-dose consequences of employing forced ventilation of the RCB through a system of filters to the stack. Results are presented here for information only.

* Ed. note: See footnote (*), Table VII-B-1, p. 284

These calculations of plume dose from stack release are based upon the methods of AECU-3066¹³, using data published by J. Z. Holland.¹⁴ (AISITE-II was not used for the stack-release calculations, although it is possible to study the effects of such release of radiation with that program.) For this analysis, it was assumed that all of the exhaust air passed through filters which were 99% efficient in trapping the halogens and the solids, but entirely ineffective in trapping the noble gases. Atmospheric-stability-Category F was assumed to apply, with a wind speed of 1 m/sec. As discussed in Section VII-C-2 above, Category F gives conservative values of the dose from stack release, at distances of 5000 m or more. Assuming a wind speed of 2 m/sec for Category E, Category E doses are smaller.

The results of these analyses are as follows:

- (1) There would be some reduction in the net total inhalation dose to the thyroid of receptors close to the RCB, provided that the forced-ventilation rate were large (e.g., of the order of 100 cfs);
- (2) There would be a large increase in the whole-body dose to receptors, at all distances, because of the exhaust of noble gases which would not be trapped in the filters.

In view of these results, it is concluded that there would be a net disadvantage to using the stack so soon after the postulated fuel melt. In the Title-II design phase, various dose-reduction devices will be evaluated, and attention will be given also to the feasibility of forced air exhaust out the stack after sufficient time delay for decay of the important noble gases.

In Table VII-C-2, a brief summary is given of the incremental whole body doses which might be expected from forced ventilation out the stack at 100 cfs for 2 hr, starting immediately after the MCA fuel melt. The dose rate to the receptor from the ventilation plume is assumed to be zero until the plume front reaches the receptor. Thus, larger 2-hour ventilation-plume doses would result if the initial time of the 2-hour dose period was delayed, and this increase in dose could be significant at distances of 5000 m or more.

4. MCA Doses from Samples Under Irradiation

It is assumed that samples of toxic materials may be under irradiation in AARR at the time of a postulated MCA. Before approval may be given for irradiation of such samples, their hazard potential for causing an accident, and their contribution to the biological hazards in the event of a reactor accident, will be reviewed. The amounts of toxic materials that will be approved for introduction into the reactor will be limited so that in the event of a MCA they would not contribute significantly to the biological doses. Criterion 5.2 (Section II) will be governing.

Table VII-C-2

Whole-Body Dose from Exposure to Ventilation Plume from the Stack
for the First Two Hours After the MCA Fuel Melt
(Ventilation Rate: 100 cfs)

Distance from RCB (m)	Whole-Body Dose (rem)		
	Plume Photons	Plume Beta-Rays	Total
100	21.	-	21.
200	21.	-	21.
500	20.	-	20.
1000	21.	-	21.
2000	26.	0.3	26.
5000	14.	1.9	16.

D. Distributions of Plume Inhalation Doses based on Statistical Treatment of Observed Meteorological Conditions

Over a period of years, Argonne National Laboratory has amassed detailed records of wind direction, wind speed, and atmospheric stability on the Laboratory's DuPage County, Illinois, site. Because the plume-inhalation doses, and especially the thyroid dose, would be the most significant radiation doses to receptors outside the RCB following a MCA, a digital computer program has been written to apply the meteorological data to obtain statistical distributions of plume-inhalation doses. This program has been applied to the case of the inhalation dose to the thyroid from the fission-product iodines, for the special case of a constant leakage rate of 1.0% of the RCB air mass per day. Depletion and decay of the individual fission-product iodine isotopes are included, but no correction is made for the time required for the plume to travel to the receptor, or for the dilution effects of turbulent air conditions in the lee of the RCB (building wake). The calculations are based on the methods of TID-14844.²

The radiation plume is assumed to originate from a point source at a fixed location. The entire circle with center at the point source is divided into eighteen 20-degree sectors. In determining the statistical 30-day inhalation dose to the thyroid for a receptor in a given sector, the recorded meteorological data have been applied as follows:

(1) All sequences of 720 consecutive hourly exposures (30 days total) were calculated, beginning with each hour of a five-year period of measurements. If, in a given period of one hour, the wind direction was recorded as being outside the given sector, zero inhalation dose was assigned. The total dose accumulated during a given 720-hour (30-day) period was calculated as the sum of the individual hourly doses.

(2) The actual wind speed, measured hourly, was included in the manner of TID-14844.

(3) Plume-dispersion parameters were selected for each hourly dose, based on Pasquill's atmospheric-stability-categories C, D, or F.

The characterization of relative stability was determined by the value, ΔT , of the temperature difference: air temperature at a height of 144 ft minus air temperature at 5.5 ft. If $\Delta T \leq -2.0^{\circ}\text{F}$, Category C was selected; if $0 \geq \Delta T > -2.0^{\circ}\text{F}$, Category D; if $\Delta T > 0^{\circ}\text{F}$, Category F. (See Table VII-B-4 above.)

Statistical analyses of this type have been carried out for: each of the eighteen 20-degree sectors; each of 9 distances: 100, 300, 500, 800, 1500, 2000, 4800, 8000, and 20,000 m; and for 1 day and 30 days. The computed data have been processed to determine 5, 10, 20, 40, 50, 70, 80, 90, 95, 98, 99, and 100th - percentile values for the plume-inhalation dose to the thyroid in each of the eighteen 20-degree sectors, at each distance.

Figures VII-D-1 through VII-D-3 present isopleths of inhalation dose to the thyroid, for various cases of particular interest in the evaluation of site acceptability. Note again that these figures are for the hypothetical case where the leakage rate is assumed to be constant, at 1.0% day. They are included to indicate the degree of conservatism of the calculations summarized in Sections VII-B and VII-C above, and to indicate the directional distribution of doses which might be anticipated in the event of a MCA. The effect of plume meandering has been included (see Section VII-B-7 above).

These special calculations also provide a comparison with dose calculations wherein gross variations in wind direction, wind speed, and atmospheric-stability conditions are not included. A separate set of calculations has been performed for the hypothetical case of constant wind direction, wind speed (2 m/sec), and atmospheric stability (Category E). Table VII-D-1 supplies ratios of the statistical inhalation doses to the Category E dose, at a distance of 4800 m from the RCB, for the special case where the half-life of the inhaled fission-product isotope is long by comparison with the total time of exposure to the plume. It may be observed that if 95th-percentile ratios are used, the effect of the expected gross variations in meteorological conditions is to reduce the inhalation-dose contributions of these isotopes by 75% or more. Some of the important bone seekers (e.g., Sr^{90} , and Ce^{144}) fall into this category.

The maximum 720-hour inhalation dose to the thyroid from fission product iodines has been determined to be 61 rem at 4800 m. This value is the absolute maximum over all radial sectors, and over a period of 5 years of recorded data. The maximum 95th-percentile value is 28 rem. For this same case of a constant RCB leakage rate of 1.0%/day, but constant conditions of wind direction, wind speed (2 m/sec), and atmospheric stability (Category E), the AISITE-II result for the inhalation dose to the thyroid is at least three (3) times as large as the above-mentioned absolute maximum, and it is at least 6 times as large as the maximum 95th-percentile value.

FIG. VII-D-1

ISOPLETHS SHOWING MINIMUM DISTANCES AT WHICH
THE 24-HOUR DOSE IS EQUAL TO OR LESS THAN 75
REM 90, 95, 98 OR 100 PERCENT OF THE TIME

TABLE VII-D-1

RATIOS OF STATISTICAL MAXIMUM INHALATION DOSE
TO CONSTANT-CONDITIONS DOSE AT A DISTANCE OF
4800 METERS FROM THE REACTOR CONTAINMENT BUILDING(a)

Percentile	Total Time of Exposure, days					
	1	2	4	8	16	30
100	1.71	1.37	0.74	0.58	0.42	0.33
98	0.66	0.53	0.44	0.33	0.27	0.23
95	0.50	0.40	0.32	0.26	0.22	0.20
90	0.35	0.31	0.26	0.21	0.18	0.17

(a) These ratios apply for isotopes with half-life long by comparison with the total time of exposure. The ratios listed are the maximum of all values in the eighteen 20-degree sectors. By "constant-conditions" is meant: Category E, wind speed of 2 m/sec, and constant wind direction.

E. Conclusions

In Section II, Criterion 19.1, the exclusion area, low population zone and population center distance for the AARR are defined in terms of the physical area controlled by the Laboratory and the population distribution, road availability for warning and evacuation, etc., in the surrounding area. The radius of the exclusion area (1500 m), the outer radius of the low-population zone (5000 m) and the population center distance (7000 m) so designated fall well within the definitions established in the Code of Federal Regulations, Title 10, Part 100.¹

The calculated radiation doses following the AARR MCA, reported in Section VII-C, above, are substantially lower than limiting values published in Reference (1), and fulfill Criterion 19.2 (Section II). The maximum two-hour whole body and plume-inhalation thyroid doses projected at the exclusion area radius (1500 m) are 1.0 and 63. rem, respectively, as compared with the limiting values of 25 and 300 rem. The infinite-time whole body and thyroid doses projected at the outer radius of the low population zone (5000 m) are <1. and 115 rem, respectively, again as compared with the limiting values of 25 and 300 rem. These doses are based upon very conservative models.

The radii at which the limiting whole body and inhalation-thyroid doses of 25 and 300 rem, respectively, are received are: maximum two-hour doses - 240 and 350 m; and infinite-time doses - 500 and 2500 m. It can be seen that the limiting two-hour-dose radii are well within the exclusion radius (1500 m) and the limiting infinite-time-dose radii are well within the low population zone outer radius (5000 m).

Other conclusions may be drawn from the doses reported. For example, the main ANL cafeteria lies some 800 m from AARR, at the center of the 200-Area, wherein almost 40% of the Laboratory population is located. At this distance, the maximum two-hour whole body and thyroid doses are 3 and 150 rem, respectively, and it should be noted that at an assumed wind speed of 1 m/sec, the plume would take ~ 13 min to reach this point. It is apparent that ample time is available for evacuation of personnel before anywhere near the limiting doses would be received.

In the innermost region of the low population zone, between its inner radius of 1500 m and a radius of 2000 m, some 700 persons reside in the northwest and southwest quadrants* (projected to 965 by 1980). It

* There are no residents in the described region in the northeast and southeast quadrants, which lie almost entirely within the Laboratory site boundary (see Figure III-A-2).

can be seen from Fig. VII-C-3 that the thyroid dose to these people would not reach the limiting value of 300 rem until ~ 4 to 9 days after the MCA, depending on their location within the zone. The 24 hr thyroid dose to these people lies between 75 and 115 rem. Indeed, the 24 hr thyroid dose to persons at the outer boundary of the low population zone (500 m) is only ~ 20 rem (see Figure VII-C-4). Similar inferences can be made with regard to whole body doses, from Figure VII-C-5. Again it is apparent that there is ample time available for communication, warning, and evacuation if necessary, before excessive doses are approached.

Although no engineered safeguards (except the Reactor Containment Building and its isolation valves) are required to mitigate the offsite consequences of the MCA, the present facility design includes systems which will (a) reduce the pressure in, and hence the fission product leakage from, the RCB and (b) remove airborne activity from the containment atmosphere, thereby reducing the fission product and/or other activity available for release. While these systems are not specifically designed to function following the MCA, there is every reason to believe that they will be or can be made operational at some reasonable time following the MCA, and the calculated doses would be reduced accordingly. As an example, for the case of one-half the air conditioning ("refrigeration") capacity continuing to operate after the MCA, the duration of overpressure (and therefore leakage) from the RCB would be reduced from 28 to 8 days, and the infinite-time thyroid dose at the outer boundary of the low population zone would be reduced from 115 to 55 rem.

To demonstrate the conservatism in the projected MCA radiation doses reported, a digital computer program developed at Argonne was used to apply measured meteorological data (recorded over a period of 5 years), to obtain statistical distributions of plume inhalation doses (to the thyroid) with distance and direction from AARR. The statistical computation gave an absolute maximum (100th-percentile) dose over all directional sectors of ~ 61 rem, or ~ 28 rem if a 95th-percentile value is used, for the infinite-time thyroid dose at the outer boundary of the low population zone. These statistically-derived values were based upon a constant RCB leakage rate of 1.0%/day (rather than upon a leakage rate varying with building overpressure), and therefore a comparison was made with an AISITE II computation based upon the same constant leakage rate and constant conditions of wind direction, wind speed (2 m/sec) and atmospheric stability (Category E). In this comparison the AISITE-II result produced thyroid doses at least 3 times the maximum (100th percentile) statistical result, and over 6 times the 95th percentile value.

In the MCA computational model, one non-conservative assumption was made, i.e., the containment isolation valves in the Containment Building ventilation system were assumed to close instantaneously at the start of the accident, and no allowance was included in the reported radiation doses for the effect of fission products released through this system to the stack in the finite period during which the exhaust air radiation detectors and the isolation valves are functioning to seal the containment. However, a separate analysis of a 3-sec full-flow release

(see Section VII-C-2) (based on detection and valve closure times of 2 sec each, which are considered reasonable to achieve) shows that the thyroid doses to receptors from this stack "puff" would amount to only a maximum of 11 rem anywhere in the exclusion area (radius 1500 m) and 5 rem anywhere in the low population zone (outer radius 5000 m). These doses are insignificant compared to those received from the Reactor Containment Building leakage plume.

It should be noted particularly that all of the projected MCA radiation doses are based upon a guaranteed maximum containment leakage rate of 1.0%/day at 7.5 psi overpressure*. As noted in Criterion 19.2 (Section II), the design maximum leakage rate is a factor of ten smaller than the guaranteed maximum rate. The detailed design and fabrication of the containment and its penetrations will be such that a leakage rate in the neighborhood of the design rate can be met in the as-built structure. Therefore it is expected that actual doses would be much smaller than those projected in Section VII-C, for this reason alone.

Special provisions are included in the current facility design to protect personnel at the AARR complex who are outside the Reactor Containment Building, in the event of the MCA. The reactor control room is located outside the Reactor Containment Building, in the Laboratory and Office Building, and specific features are incorporated into the design to protect persons in that room against radiation hazards, at least for a time sufficient to secure the plant, possibly mitigating the severity of the accident. The control room is provided with additional shielding to allow personnel to remain in this room for the period required to secure the plant and to evacuate, without receiving excessive doses. Other features of design provide for filtered ventilation and slight pressurization of the control room to minimize in-leakage of fission products which might be leaking from the RCB into the Laboratory and Office Building.

In the current design, a monolithic concrete building wall separates the Laboratory and Office Building from the outer wall of the RCB. The wall provides additional shielding against direct gamma-ray radiation, shielding which is approximately equivalent to a 1-ft thickness of ordinary concrete. Where the wall is not continuous, equivalent protection is provided by shadow shields. The thickness of the Laboratory and Office Building concrete roof is also sized to attenuate direct radiation from the RCB. Assuming that there is little exposure to direct contamination from fission products leaking from the RCB, it is estimated that persons in the Laboratory and Office Building would receive a whole-body dose of considerably less than 10 rem if they evacuated the building

* Ed. note: See footnote p. 278.

within 10 min after a MCA.

The closest neighbor to the AARR facility on the ANL site is the CP-5 Research Reactor Facility. At its nearest approach, this facility is less than 100 m from the Reactor Containment Building of AARR. From Figure VII-C-1 it may be seen that the 2-hr MCA doses at this distance exceed the emergency doses. However, certain factors should be effective in reducing the MCA doses to personnel at CP-5. These factors are:

(a) Leakage from the reactor will most likely occur at more than one point, as opposed to a single leak at ground level.

(b) Persons within the buildings of the CP-5 complex ordinarily would be protected from direct contact with the plume; fission products might be drawn into the intake of the CP-5 ventilation system, but in this case filtration and additional dilution would result, since the normal intake rate for CP-5 is 12,000 cfm, in comparison with a leakage rate significantly less than 10 cfm for the AARR MCA.

(c) The CP-5 emergency evacuation warning system will be directly tied to AARR; on this basis it is reasoned that CP-5 personnel can be evacuated within 15 min after an accident in AARR, through an exit that is most distant (175 m) from AARR. Under these circumstances, emergency doses will not be exceeded.

It is therefore concluded that the AARR facility may be constructed and operated safely at the site described herein, and that even if the postulated MCA should occur, adequate reactor containment as presently envisioned would prevent the creation of an undue hazard to the general public, or to general Laboratory personnel.

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12. Ibid. Reference 2, Table III, p. 25.
13. "Meteorology and Atomic Energy," AECU-3066 (July 1955).
14. J. Z. Holland, "Radiation from Clouds of Reactor Debris," Proceedings of the International Conference on the Peaceful Uses of Atomic Energy (1955), P/572, Vol. 13, pp. 110-118.

VIII. ANALYSES OF HYPOTHETICAL ACCIDENTS

A. Introduction

In Section VI above, analyses were presented of accidents and situations which are credible in the sense that credible mechanisms for their initiation can be visualized. The worst of these is the rupture of the main primary coolant outlet line from the reactor vessel in a location outside the Reactor Pool. This latter accident forms the basis for the Maximum Credible Accident (MCA) calculational model (which because of conservative assumptions represents a more pessimistic situation than the actual worst accident) described and analyzed in Section VII above.

Frequently during the course of a safety review, consideration is given to accidents and situations for which no credible mechanisms can be postulated, either to substantiate the conservatism in the MCA model or to demonstrate that their incredibility removes them from consideration in the MCA analysis, which by definition is intended to cover only accidents which are credible. Such accidents and situations are designated as "hypothetical accidents" in this report.

Analyses of some such hypothetical accidents are presented in this Section. It should be noted that the consequences of most of the cases analyzed are not as serious as those of the MCA.

B. Hypothetical Rapid Insertions of Reactivity to Super-Prompt Criticality

1. Introduction

In this section, the anticipated reactor response to hypothetical rapid insertions of much more than one dollar in reactivity is discussed. It is shown that even for a reactivity addition of as much as $\frac{k_{ex}}{\beta_{eff}} = 7 -$

or more - there should not be a prompt massive metal-water reaction between the fuel plates and coolant. This conclusion is important to the evaluation of consequences of reactivity accidents, for example, the assessment of the capacity of the reactor pressure vessel and other components to survive such an excursion. The containment potential of the reactor pressure vessel is reviewed in Appendix E below, with consideration for the results discussed herein.

2. Models for Reactivity Compensation

In Reference (1), the expected response of the HFIR to a hypothetical rapid rise in reactivity to well above prompt-criticality is discussed. Analysis of such excursions still is conjectural, for there is insufficient similarity between the HFIR and actual systems exposed to such reactivity transients to permit one to make verifiable, precise statements. For this reason, the model of reactor response described in Reference (1) was intentionally made very simple. In that model, all of the extra thermal energy released in the excursion was assumed to be stored in the fuel plates. Conservatively, the thermal expansion of the fuel plates was assumed to cause a corresponding uniform expansion in core volume without ejection of coolant water from the fuel zone. It was concluded that, even with the non-uniform power distribution in the core, the reactor could compensate a step of 3.3% in excess reactivity without vaporizing any of the fuel. It was argued further that if, somehow, a larger reactivity addition occurred, the addition would be relatively slow and a steam explosion would disassemble the reactor and shut it down before all of that reactivity could be inserted.

Two other models of reactivity feedback have been considered for the preliminary safety analysis of the AARR. In each model, it is assumed that thermal expansion of the fuel is compensated by buckling of the fuel plates and ejection of some coolant from the fuel zone. Model "A" postulates that, in response to a rapid rise in reactivity to well above prompt-criticality, the extra heat would be stored in the fuel plates, and the corresponding volumetric thermal expansion of the plate would eject the same volume of water from the fuel zone without a change in the spatial dimensions of the fuel zone. Since the reactor fuel zone already is undermoderated, this further increase in the metal-to-water ratio would result in a large reactivity loss. Moreover, conduction of heat to the water would reduce reactivity further, in a short-period transient.

In Model "B", it is assumed that the lateral thermal expansions of the fuel plates buckle the plates and thereby increase the metal-to-water ratio in the fuel zone. The fuel zone is assumed to retain its original total lateral area. Longitudinal thermal expansion is assumed to occur to the extent of one-third of the total volumetric plate expansion. Other negative-reactivity effects, which have not been included in Model B, are: (1) thermal conduction from the fuel plates to the coolant water would result in rapid local heating of water, with the probable formation of steam bubbles; and (2) substantial elongations of the fuel plates would extend a fraction of the fuel beyond the vertical extent of the beryllium reflector and into regions where the "black" portions of the control plates are effective.

Table VIII-B-1 summarizes the computed change in reactivity as a function of the assumed uniform volumetric expansion of the fuel plates, for Models A and B. The fractional volumetric expansion of the fuel plate, $\Delta V/V_0$, was approximated by linear functions of the incremental heat stored in the plate, ΔH . In the solid phase, $[\Delta V/V_0] = 8.8 \times 10^{-5} \Delta H$, where ΔH is in units of watt-sec/gm.

Table VIII-B-1

Change in k_{eff} Due to Fuel-Plate Expansion

Percentage Expansion	$\frac{k_{ex}^a}{\beta_{eff}}$	
	Model A	Model B
30	-10.6	-5.0
60	-25.4	-10.8
80	-39.0	-15.4

^a

The value of β_{eff} is 0.0071. Note that $k_{ex} \equiv k_{eff}^{-1}$ is not the change in "reactivity", as it is usually defined, namely,

$$\rho = \frac{k_{eff}^{-1}}{k_{eff}}.$$

In the transition phase, the additional thermal expansion of the metal is $[\Delta V/V_0] = 1.9 \times 10^{-4} \overline{\Delta H}$, where $\overline{\Delta H} \equiv \Delta H - (\Delta H)_1$, and $(\Delta H)_1$ is the heat required to raise the temperature of the plate to the melting temperature. This linear relationship is a good approximation in the liquid phase, as well:

$$[\Delta V/V_0] = 1.9 \times 10^{-4} \overline{\Delta H}, \text{ where } \overline{\Delta H} \text{ is the heat added to the molten fuel.}$$

With these models of reactivity feedback, the peak reactor power and total energy released in the hypothetical excursion were obtained with a digital-computer program. Table VIII-B-2 summarizes the computed response of the reactor for total inputs of 4 to 7 in units of k_{ex}/β_{eff} , as steps or ramps. The reactor is assumed to be operating at 100 MW at the time of the reactivity injection. It may be seen that for this range of excursions the total incremental energy release to the time of reactor shutdown is almost the same for the step and for the 10-msec ramp.

In both models A and B, it was assumed that every fuel plate receives the same incremental thermal energy. Both models ignore the loss of heat due to thermal conduction from the cladding surface into the coolant water. In view of the high thermal conductivity and small thermal time constant of the HFIR fuel plates, this results in an underestimation of reactivity feedback. The reactivity coefficient for partial voiding of coolant water in the fuel zone is negative and large ($-1\%/ \%$ void). The conduction of heat to the water also would lower the fuel temperature, and possibly the reactor would be shut down without ignition of the metal in the presence of steam.

If the reactivity excursion were large enough, it is conceivable that an effect similar to the SL-1 fuel-plate eruption might occur. In the SL-1 accident, it appeared that molten fuel was injected into the water and a steam explosion occurred which disrupted the core and shut down the reactor. The problem for the AARR is more complicated because certain motions of fuel could result in an increase of reactivity. However, it is reasonable to expect that a rapid disassembly of metal fuel plates would result also in an ejection of water, and this negative reactivity effect should be considerably larger in magnitude. The SL-1 fuel plate was more than twice as thick (120 mils) as the 50-mil-thick HFIR fuel plate, with a cladding thickness more than three times as large as that of the HFIR plate. The thermal time constant of the HFIR fuel plate is much smaller. Note that in the case of the SL-1 reactor, the destructive excursion resulted in the melting of approximately 50% of the plate area of the core but only $\sim 1\text{--}1\frac{1}{2}\%$ of the core aluminum participated in a metal-water reaction.²

3. Metal-Water Reactions

It is expected that the occurrence of a prompt massive metal-water reaction would require fuel-plate temperatures in the vicinity of

TABLE VIII-B-2

SUMMARY OF REACTOR RESPONSES TO REACTIVITY TRANSIENTS
ASSUMING ONLY FUEL-PLATE-EXPANSION FEEDBACK OF REACTIVITY

Model A (Fixed Reactor Core Volume)

Total Change in k_{ex}/β_{eff}	4			5			6			7		
Total Time for Ramp Reactivity Insertion, msec	0	10	30	0	10	30	0	10	30	0	10	30
Maximum Percentage Expansion of Average Fuel Plate	21	21	18	27	27	-	33	32	-	38	37	24
Total Incremental Energy Release, MW-sec	100	99	87	122	121	-	143	142	-	163	160	112
Ratio of Peak Power to Initial Power of 100 MW	200	194	111	321	309	-	463	442	-	625	583	210

Model B (Fixed Lateral Core Area)

Maximum Percentage Expansion of Average Fuel Plate	40	40	35	51	-	-	62	-	-	72	71	49
Total Incremental Energy Release, MW-sec	172	171	153	214	-	-	254	-	-	293	290	205
Ratio of Peak Power to Initial Power of 100 MW	323	317	201	524	-	-	766	-	-	1051	1009	377

the boiling point of aluminum. At operating pressure, this is roughly 3600°C (see Ref. 1). For the hypothetical case of a reactivity step to $[k_{\text{ex}}/\beta_{\text{eff}}] = 7$, and Model-A feedback, the peak averaged fuel-plate temperature is $\sim 1750^{\circ}\text{C}$. For so rapid a transient, the temperature at the cladding surface would rise less rapidly than the temperature at the center of the fuel plate. Also, other mechanisms of negative feedback of reactivity would tend to terminate the reactivity excursion sooner. For these reasons, it is expected that steam formation and/or a steam explosion would cause reactor shutdown with perhaps a small fraction of the fuel involved in a metal-water reaction. If the reactivity were added over a time interval of 30 msec or longer, the total nuclear energy release would be much smaller (see Table VIII-B-2) and conceivably a control-plate scram could occur to assist reactor shutdown.

It is considered incredible that all of the fuel-plate material would undergo a complete metal-water reaction. If the temperature of the fuel-plate material reached the ignition point ($\sim 1400^{\circ}\text{C}$), it is expected that the fuel would be explosively disassembled and many of the small molten particles would be sprayed against large, relatively cold metal surfaces. This would reduce the surface area in contact with steam, with consequent reduction in burning rate. Conduction of heat to the cold metal would reduce the temperature of the molten material and would tend to quench the metal-water reaction.

In spite of these arguments against the occurrence of a complete metal-water reaction with all of the fuel-plate material, it is instructive to examine the time-scale and the magnitude of a complete reaction, assuming also a conservative fragmentation of the molten material. The rate at which burning occurs depends strongly upon the state of the core subsequent to the time of initiation of the reaction. The actual degree of core subdivision is not predictable. Only a few experimental results are available as general guides in this respect. Particle-size analysis of the residue collected from the SPERT-ID core following its destructive test indicates that a surface area augmentation factor of 6 occurred for the molten portion of the core.³ Subsequent ANL experiments in TREAT, on HFIR and SPERT-ID fuel plate material, have demonstrated that under transient heating conditions causing fragmentation, the typical HFIR fuel fragments are larger, i.e., the ratio of surface area to volume is smaller.^{4,5,6} Therefore the burning rate of the aluminum content of the HFIR fuel material would be lower.

An analysis has been made of the rate of aluminum-water reaction using the following model:

a. The fuel disassembles uniformly into spherical particles such that the resulting surface to volume ratio is 6 times the value for the original fuel plate.

b. All particles thus formed ignite and burn according to the small-particle-rate law (see Appendix B below).

c. The reaction is complete, i.e., all the aluminum content of the burning particles reacts with water.

The total amount of aluminum which reacts is then 7.2×10^4 gm.⁷ The associated chemical energy release is approximately 1300 MW-sec which would be accompanied by the evolution of 9×10^4 standard liters of hydrogen.

The original ratio of surface area to volume of the fuel plates of the AARR core is approximately 15.7 cm^{-1} ; the augmented ratio is then $\sim 94 \text{ cm}^{-1}$. Particles formed by disassembly would then have a diameter of 0.07 cm based upon a metal density of 2.0 gm/cm^3 (approximate density at 2000°C , the assumed temperature of the burning aluminum). Each particle would contain 3.6×10^{-4} gm of aluminum and have a surface area of $1.5 \times 10^{-2} \text{ cm}^2$.

Applying the above-cited rate law, the aluminum content of each particle would react at a constant rate of $7.4 \times 10^{-4} \text{ gm/sec}$ and hence the total duration of the reaction would be approximately 1/2 sec. The total chemical energy release rate, approximately 2-1/2 MW-sec/msec, is much lower than the associated nuclear energy release rate for hypothesized transients having the potential for initiating metal ignition over a large area of the reactor core. For the case of a 10-msec ramp to $(k_{\text{ex}}/\beta_{\text{eff}}) = 7$, the average rate of energy release to time of reactor shutdown is $\sim 10 \text{ MW-sec/msec}$. Also, the chemical energy release is delayed, in that it occurs primarily after fission is terminated.

4. Summary and Conclusions

Neither in planned destructive reactor tests nor in reactor accidents has a substantial fraction of the fuel been involved in metal-water reaction, prompt or delayed, even though very short periods were encountered. The reactivity injected in these excursions was in the range of 3-6 dollars. To accomplish the BORAX-I destruction, for example, a control rod controlling about 4% (~ 6 dollars) was rapidly ejected to produce an asymptotic period of $\sim 2.6 \text{ msec}$.⁸ The total nuclear energy release is estimated to have been 135 MW-sec. While the amount of fuel-plate material that suffered a metal-water reaction is unknown, it is known that of the fuel that melted, a substantial fraction did not suffer a metal-water reaction.

The computations described above, for the AARR, indicate that there are important inherent mechanisms for reactor shutdown which would prevent, or at least limit, the occurrence of a prompt metal-water reaction in the fuel zone even for step reactivity changes as large as $(k_{\text{ex}}/\beta_{\text{eff}}) = 7$. For a more nearly realistic accident, with reactivity

injection occurring over a longer period of time, say 30 msec or more, probably little of the fuel-plate material would suffer any metal-water reaction.

Nevertheless, in the event of a very rapid, very large accidental reactivity addition, conceivably a large fraction of the fuel zone might burn, with a potential energy release of ~ 1300 MW-sec. This burning would proceed at a relatively slow rate, and would require $\sim 1/2$ sec for completion. This time period is sufficiently long to assure distribution of the energy over piping and other components of the primary system, not simply over the reactor pressure vessel, and the likelihood that the primary system will remain unbreached is enhanced. (See Appendix E for a discussion of the containment potential of the reactor pressure vessel.)

C. Primary System Ruptures Accompanied by Failure to Scram

As discussed in Section VI-D-2 above, it is considered incredible that control plate scram will not be accomplished following the occurrence of a major primary system rupture. The time required for full control plate insertion following the resulting pressure loss is no greater than 350 msec, including pressure sensing and plate de-latch delays. As the analysis of Section VI-D shows, this is insufficient time for the development of a means for control plate jamming due to gross fuel plate melting. This conclusion is reached even if it is assumed that the reactor continues to operate at full 100 MW power level during the time required for complete control plate travel. Hence the discussions which follow are considered to relate only to hypothetical reactor conditions following as assumed major rupture of the primary system. Nevertheless, it is instructive to examine the sequence of events and possible consequences of simultaneous scram failure, and therefore the following material is directed toward this end.

The discussions which follow are based upon examination of reactor conditions for the same initiating accidental events described in Section VI-D-2, viz. ruptures of (1) inlet line inside the Reactor Pool, (2) outlet line inside the Reactor Pool, (3) inlet line outside the Reactor Pool, and (4) outlet line outside the Reactor Pool, all coincident with failure of the reactor to scram. The anticipated reactor behavior following each of these assumed ruptures is elaborated upon below.

Case 1. Ruptured Inlet Line Inside the Reactor Pool

It is assumed that a complete pipe severance occurs such that with continued pump operation, water from the pump is discharged to the pool and pool water is drawn into the reactor vessel. The normal coolant flow rate would be reduced because of the pump characteristics under these off-design conditions, and would reach a value of approximately 5.2×10^6 lb/ft²-hr, calculated as previously described in Section VI-D-2 above. The maximum tolerable heat flux in the core under these conditions is estimated to be that corresponding to a reactor power level of approximately 68 MW. This value of maximum heat flux is estimated from the data of Fig. VI-D-4, the source of which is described in Section VI-D-2.

At the reduced values of flow rate and pressure, steam would probably form in much of the core. The formation of steam voids in turn would tend to shut the core down. However, the steam voids could then be swept out of the core, water reintroduced, and the core power raised again. The amplitude and frequency of such reactor power oscillations cannot be calculated, but it is justifiable to assume that if the core were to continue to operate without meltdown, the time averaged maximum heat flux could be no larger than that shown to be the burnout flux under these conditions of operation, i.e., the heat flux equivalent to 68 MW power level.

With continued operation of the reactor at an average power level of 68 MW, and with circulation through the Reactor Pool, primary coolant would initially be drawn from the reactor pool at 75°F, pass through the reactor and the primary heat exchangers (the secondary coolant system is assumed to

continue operating normally), and return to the Reactor Pool at some temperature above 75°F but less than the 120°F control point on the heat exchangers. Assuming for the purposes of this analysis that the water would re-enter the pool at 120°F, for a total pool water inventory of 190,000 gal at 75°F, the influx of ~8900 gpm of primary water at 120°F would begin to raise the pool temperature at a rate of 0.8°F/min.

Since the core is assumed not to have melted thus far, the containment isolation valves would not yet have been tripped automatically by the release of fission products, and therefore no building overpressure could yet develop because of pool temperature increase. However, if the containment isolation system were tripped by release of N^{16} from the primary system to the Reactor Pool, or if the system were tripped manually, by administrative decision, an overpressure could occur. Based on a final equilibrium temperature of the pool water and the building atmosphere of 120°F, the overpressure would be about 2 psi (1/2 psi air pressure and 1-1/2 psi water vapor pressure).

If at this time cooling should cease and conditions (core melting, metal-water reaction and hydrogen combustion) leading to fission product release should occur, the additional energy would result in approximately 4-1/2 psi overpressure superimposed on the existing 2 psi overpressure. This would result in a final overpressure of 6.5 psi which is still within the design pressure of 7.5 psi used for the containment shell.* However, there is no apparent cause of a delayed loss of cooling capability to occur such as just described. It is considered to be much more likely that either adequate cooling of the core would continue indefinitely and the reactor would eventually shut down without fuel melting, or that a partial core meltdown would occur shortly after initiation of the accident.

In the former case (no melting), no fission product release would occur. However, the containment isolation system would be tripped by the release of N^{16} to the Reactor Pool. In the latter case (partial core meltdown), little or no metal-water reaction would be anticipated since some forced circulation of coolant would be maintained either by the main pump motors or the pony motors. Fuel melting could be sufficient to cause shutdown by core disassembly, or, if limited, the reactor could continue to produce fission power. In either event the total fission product release is expected to be small compared to the full core inventory. Since no primary system water would be discharged into the RCB and no significant metal-water reactions would occur, at worst the resulting overpressure developed in the Reactor Containment Building would not significantly exceed the value of 2 psi cited above for the case of no fuel melting.

Case 2. Ruptured Outlet Line Inside Reactor Pool

In this case it is assumed that a rupture occurs in the reactor vessel outlet line within the Reactor Pool, and the pumps continue to operate and circulate coolant through the primary heat exchangers. As in Case 1, the pump characteristics are such as to result in reduced flow, calculated to be approximately 9.4×10^6 lb/ft²-hr. Under the accident

*Ed. note: Footnote p. 278 invalidates this statement.

conditions of pressure and flow rate, the data of Fig. VI-D-5 indicate that the reactor with a perfect core could marginally operate at a power level of 100 MW. At these values of power level and flow rate, a heat balance calculation gives an average coolant temperature at the reactor outlet of 164°F, and the pool water would eventually rise to this temperature.

If indeed the core were perfect (i.e., there were no hot spots), then no melting and fission product release would occur. However, the containment exhaust radiation detectors would trip the isolation system because of release from the reactor outlet line to the Reactor Pool of N¹⁶ which had not yet significantly decayed. With a normal core having hot spots, some melting at the hot spots could be expected. Only a minor fission product release would occur, since the core hot spots at worst represent no more than 5% of the total core volume. As in the previous case considered, no significant metal-water reactions are anticipated in the presence of continued forced circulation of the coolant.

In either case, then, the Reactor Containment Building could be expected to seal automatically, and a containment overpressure would develop because of the increase in pool temperature. Assuming (1) the air temperature instantaneously follows the pool temperature, (2) the water vapor pressure is at all times in equilibrium with the pool, and (3) the containment is ideally isolated (no heat losses and no ventilation), the pressure in the containment will approach a value of ~7 psig (2 psig air pressure and 5 psig water vapor pressure). This is only slightly below the design pressure of the Reactor Containment Building (7.5 psig);* however, since in this case only ~5% of the core is considered to have melted, the radiation dose consequences to the surroundings would be far less serious than those projected in Section VII above for the Maximum Credible Accident.

Case 3 and Case 4. Ruptured Inlet or Outlet Line Outside Reactor Pool

For the remaining two hypothetical cases of primary system rupture without scram to be examined, i.e., those involving rupture locations outside the Reactor Pool, it is considered that the initial stages of these accidents would be similar to those described for the same cases discussed in Section VI-D-2 above. The initial amounts of coolant lost from the primary system and the assumed temperature levels would not be significantly different from those previously assumed for each corresponding case of rupture location.

The major differences to be considered under the present assumption of failure to scram are that a much greater potential for rapid core temperature rise, and hence a greater potential for rapid and more extensive metal-water reactions of the fuel plates, would exist. Certainly, there is little likelihood that subsequent to loss of forced flow, natural coolant circulation could adequately cool any portion of the core fuel zone, and complete meltdown and full fission product release would be anticipated.

*Ed. note: See footnote p. 330.

To illustrate a worst-case evaluation of fuel temperature rise, insulated heating of the AARR core at 100 MW power level would produce an average rate of temperature rise of approximately 1300°C/sec. Under these conditions, ignition of the fuel plate metal could occur at a temperature level of approximately 1400°C (see Appendix B, Section B-III). If ignition should occur, complete consumption of the aluminum content of the fuel plates could be accomplished in ~11 sec based upon the original heat transfer area of the core. Fuel plate disassembly would be expected subsequent to ignition, and the burning metal would fall to the bottom of the reactor vessel.

If a significant fraction of the fuel plate aluminum were to react rapidly, it is considered likely that much of the water contained in the reactor vessel would be expelled by steam and hydrogen generated as a result of the metal-water reaction, possibly uncovering the beryllium reflector and other associated core materials and support structures. Depending upon the distribution of the chemical energy release and subsequent fission product decay heat, the possibility of further metal-water reactions involving these materials cannot be ruled out when they are not covered with water, but are supplied with steam.

The additional potential energy sources available from various reactor vessel internals are presented in Table B-1 of Appendix B. Examination of the data presented therein shows that significant additional energy sources are available through metal-water reaction of materials other than the fuel plates. This is particularly true in the case of the beryllium reflector.

Unfortunately, experimental data relevant to beryllium-water reactions are quite sparse. Those experimental results presently available are discussed and summarized in Section B-IV of Appendix B. Briefly stated, the results indicate that oxidation reactions between beryllium and water may occur at temperatures below the melting point which is approximately 1280°C. An ambient heat source is necessary to sustain such reactions, and the reaction rate is quite low for beryllium temperatures below 1000°C, as illustrated by the data of Table B-7 of Appendix B. At temperatures above the melting point, beryllium tends to ignite and is self-heated to temperatures above ambient. This reaction is energetic in water, but no data are available to indicate whether or not the reaction is self-sustaining under these conditions.

Preliminary calculations have been made to determine lower limits on the time required for beryllium temperatures to reach the melting point when exposed to the ambient heat sources provided by fission product decay and burning aluminum.

These calculations show that with normal decay heat* in an uncovered reflector and assuming no internal conduction to cooler regions, the part of the removable reflector receiving the maximum gamma heating would reach

* In these calculations it was assumed that all gamma radiation due to fission product decay is absorbed in the reflector. This overestimates the reflector decay heat source by at least 100%.

the melting point in approximately 4-3/4 min. That part of the removable reflector operating at average gamma heating rates would reach the melting point in 11 min. The permanent reflector would melt after 17.5 min at its maximum heating point and after 58 min at its average heating point.

It should be noted that the times given represent absolute minimum values because of (1) an assumption of no heat conduction, and (2) the fact that the core would not be in place after this type of accident and the gamma heating rates would be much lower than those used.

Radiant energy from burning aluminum would shorten the times listed above, but further detailed analysis is required to determine the extent of the reduction. Assuming a view factor of unity, the radiant heat flux is such that it could heat one cm³ of beryllium to its melting point in about 2 min (no conduction considered). This is in the absence of gamma heating.

Other calculations have shown that should burning aluminum contact the beryllium reflector at operating temperature, the reaction would be quenched even if the reflector were uncooled.

The degree to which all of these effects could combine is unknown and it is therefore impossible at this time to deny the possibility of a reaction under these assumed conditions. (It must be remembered that the assumed conditions are considered incredible.) However, the calculations do show that reflooding of the reactor vessel, at a rate sufficient to cover the reflector with water prior to the occurrence of beryllium temperatures near the melting point, would provide sufficient cooling to render a significant beryllium-water reaction incredible. This entire problem area requires additional study to determine more realistic estimates of potential heat sources, associated temperature increase rates and vessel reflooding rates. Continued analyses will be conducted during the detailed design phase of the AARR to determine the necessity for and type of vessel reflooding mechanisms that might be employed. A preliminary concept for vessel reflooding, utilizing Reactor Pool water is shown schematically in Fig. VI-D-2. In present concept, the valves depicted in the figure would open in response to the development of a very low pressure level in the reactor vessel, thereby allowing it to communicate directly with the pool.

With respect to structures other than the fuel plates and reflector listed in Table B-1 of Appendix B as having potential for significant chemical energy release, it is concluded that any measures taken which obviate the possibility of a beryllium-water reaction will also prevent significant reaction of the other materials. This is true since they are exposed to the same ambient heat sources, and the ignition temperature for beryllium is less than that necessary for ignition of the remaining structures.

Associated with the assumed accident conditions described above (which, again, are considered incredible), a consideration is now given to one additional problem, viz. the potential for the disassembled core to melt through the bottom of the pressure vessel and subsequently through the Reactor Containment Building floor.

In the hypothetical accident described in preceding paragraphs it is postulated that the core would melt down and accumulate at the bottom of the reactor vessel. It is considered possible that subsequently the core might melt through the bottom of the vessel and drop to the floor of the room below the reactor vessel. To determine a means for containing the molten core there, it is planned to study the effectiveness of collecting the molten material in a sturdy pan, water-cooled underneath, installed above the aforementioned floor. This is an approach similar to that taken by the designers of the Indian Point-II reactor.

For the AARR, one ft³ of fuel is involved, producing approximately one MW of heat one hour after shutdown. If natural circulation of molten fuel is neglected, the maximum thickness to which the fuel could build up is approximately 0.32 ft (based on equal heat losses from both bottom and top). At this thickness, the boiling point (4200°F) of the fuel would be reached, and fuel dispersal would commence. The corresponding maximum heat flux into the pan would be 5.5×10^5 Btu/hr-ft² which would be carried away by nucleate boiling of water beneath the pan.

Actually, if the fuel were 0.32 ft thick, it could be only 2 ft in diameter, and such a geometry seems readily amenable to slumping. Therefore, the above noted heat flux appears quite conservative, even if natural convection of molten fuel were considered. If slumping rather than the fuel boiling point controls maximum fuel height then the possibility that natural convection might increase both the maximum possible fuel thickness and the pan heat flux need not be considered.

Based upon this preliminary analysis, it appears likely that the molten core could be contained in the manner described above, and hence containment would not be violated.*

In summary, it is concluded that even in the event of failure to scram (considered incredible) following a massive rupture of the AARR primary system, the consequent accident conditions would be no worse than those assumed for the AARR MCA as described in Section VII above, provided certain minor modifications are incorporated into the facility design to render incredible the release of any significant energy due to hypothetical metal-water reactions in addition to those already assumed for the MCA. The preliminary and very conservative analyses discussed above indicate that this can readily be accomplished.

*Ed. note: This analysis was performed only to show that a means could be provided to prevent containment melt-through if the core were to melt through the reactor vessel. It was never intended that a catch-pan as described here be provided unless it were shown for the Final Safety Analysis Report that vessel melt-through was possible.

An example of such design modifications would be the reflooding valves mentioned above. Thus far, these valves have not been definitely included in the design, since it is felt that further more detailed analysis is necessary to evaluate the need for the valves as against the possible increased hazard they might create. For example, inadvertent opening of the valves during normal operation could lead to a loss-of-coolant accident of the Case 1 or Case 2 type as described in Section VI-D-2. Such detailed analysis will be carried out on a more realistic basis during Title-II design, both to evaluate and establish needs, to investigate alternate approaches, and to determine detailed design characteristics as required.

D. Hydrogen Explosion Hazards in the Reactor Containment Building

1. Disposition of Hydrogen Evolved from Core Metal-Water Reaction

The MCA for the AARR presumes the occurrence of a large primary system rupture, complete core meltdown and attendant fission product release. In addition, the melted core is assumed to undergo a complete metal-water reaction which produces heat due to the exothermic nature of the reaction. This heat is added to other MCA heat sources included in the calculation of Reactor Containment Building (RCB) overpressure (see Section VII-B-2 above).

The aluminum-water reaction of the melted core also produces an amount of hydrogen which has the potential for reacting with the oxygen content of the RCB air, thereby constituting an additional source of energy which may contribute to containment overpressure. The total hydrogen released due to a complete aluminum-water reaction of the active fuel plate material (7.2×10^4 gm of aluminum) is approximately $3,160 \text{ ft}^3$ at STP. It is considered that the hydrogen will tend to escape from the reactor vessel through the system rupture or other openings and be released to the RCB atmosphere. Three possible conditions of hydrogen disposition following this release are examined: (a) hydrogen diffusion throughout the free volume of the RCB, (b) the formation of local pockets of high hydrogen concentration and (c) continuous ignition of the hydrogen as it escapes from the primary system. These assumed conditions are elaborated upon, in turn, in the following paragraphs.

(a) Because of the high diffusion rates for hydrogen in air, the most likely disposition of the evolved gas is considered to be that which results in a concentration much below its lower limit of flammability. The free air volume of the RCB is approximately 1.2 million ft^3 .^{*} Thus if the hydrogen were uniformly distributed throughout this volume, the resulting concentration would be approximately 0.25%. This concentration is more than a factor of 16 below the lower flammability limit (4.1 volume %) in air. Therefore, it is concluded that in this case ignition of the hydrogen is impossible.

(b) If the evolved hydrogen becomes concentrated in local pockets there is a possibility of an explosive reaction with air. The minimum concentration of hydrogen in air (STP) required for detonation is 18.3 volume %, and the possibility of a severe explosion exists until the hydrogen concentration exceeds 59%.⁹ The remoteness of the possibility of achieving detonation conditions due to local hydrogen-air stratification is demonstrated through comparison of the minimum detonation concentration with the value resulting from uniform hydrogen dispersal. It is seen that a factor of approximately 75 is involved.

* Ed. note: See footnote (*), Table VII-B-1, p. 284.

It would appear that the achievement of concentrations of hydrogen necessary for detonation could only be accomplished through entrapment of the escaping gas in a small volume enclosure. However, all such enclosures of primary system components will be unsealed, and will communicate with the RCB through large openings which will provide paths for hydrogen diffusion out of such areas.

The coefficient of diffusivity of hydrogen in air is approximately $0.7 \text{ cm}^2/\text{sec}$ at 70°F . If it is assumed that the hydrogen escapes into the RCB atmosphere in the form of spherical, balloon-like bubbles of pure hydrogen, there would be a tendency for the hydrogen to rise to the containment dome. The bubbles would become lower in hydrogen concentration as they rose because of diffusion. The effective concentration "time-constant" is a function of the hydrogen bubble diameter and environmental conditions. Since the bubbles move through air of essentially zero hydrogen concentration, an estimate of the decay time of the concentration within the zone formed by the initial bubble can be made for various assumed values of initial bubble diameters. To illustrate, the following estimates of hydrogen concentration time-constants are presented, as derived from solution of the standard diffusion equation for spherical geometry with boundary values appropriate to the assumed conditions:

<u>Bubble Diameter (cm)</u>	<u>Time-Constant-τ(sec)</u>
2	0.18
4	0.67
10	4.4

Here the time-constant, τ , is used in its usual sense, i.e., the time required for the initial concentration to be reduced by a factor of $e \approx 2.7$. The values of τ listed above are for points located at the half-radius of the bubble, and to ensure that the overall hydrogen concentration within the bubble is below the lower limit of detonation, a diffusion time of approximately 2τ is required. Although it is impossible to predict bubble sizes for the escaping hydrogen, the foregoing analysis indicates that even for rather large assumed bubble sizes, the concentration decays rapidly to a level below the minimum required for detonation.

(c) The possibility of ignition of the hydrogen as it is evolved from the primary coolant system and contacts air cannot be discounted. Since the path of the escaping hydrogen is likely to be through water, the exit temperature will be far below the spontaneous ignition point and self-ignition is not likely. However, there is the possibility of contact of the gas with hot metal or an electric spark near the release point, which could lead to ignition. In this case the hydrogen would burn quietly as it contacts the air; the hydrogen would be consumed and no explosion would result.

2. Likelihood of Hydrogen Detonation in Air

A comprehensive review of the hazards associated with the use of liquid hydrogen in various experiment facilities is presented in Reference 9. Several of the studies described therein are directed toward determining the explosion potential of gaseous hydrogen and air mixtures; these results are relevant to an evaluation of the likelihood of a hydrogen detonation within the AARR RCB. In the following paragraphs, material extracted verbatim from Reference 9 is identified by quotation marks and indentation.

"The speed with which a flame moves through a quiescent mixture is a major factor in the destructive effects produced. When the flame speed is less than the local speed of sound the burning is known as a deflagration-type of combustion. However, when the flame speed is greater than the local speed of sound, combustion of the detonation-type results. In a closed vessel, deflagration is accompanied by a comparatively slow pressure rise as a consequence of the relatively low flame speed. Because a detonation produces a very rapid pressure rise in a closed vessel, its effects are highly destructive. In contrast to a deflagration wave which is propagated by heat transfer and diffusion, a detonation wave is a shock wave sustained by the energy of the chemical reaction initiated by the temperature and pressure of the wave. The limits of detonability of a hydrogen-air mixture are 18.3 and 59 volume percent hydrogen."

The results of experimental investigations indicate that a confined state of an explosive hydrogen-air mixture presents the conditions necessary to produce a detonation with all its destructive effects. The following material describes studies which provide a basis for this conclusion.

"Liquid hydrogen bubble chambers are enclosed in concrete blockhouses for studies that require radiation shielding. Release of hydrogen within the comparatively small volume of such an enclosure could result in a severe hazard. To investigate the formation of explosive mixtures at various heights above floor level, 2.7 to 65 liters of liquid hydrogen were spilled in a 21' x 13.3' x 13.5' concrete blockhouse with an open 9" x 48" beam port and one 13.3' x 13.5' Masonite wall. The first samples, taken two seconds after the 65 liter spill, showed that explosive mixtures were developed at each of five levels from 29 to 145 inches from the ground. A 41 liter-spill produced the same results and lesser quantities down to 8.3 liters developed mixtures above the lower explosive limit in two seconds at some of the sampling levels.

"To observe the results of ignition following spills, tests were made in the blockhouse, using either an electric match or an M-36 military detonator after 10 to 40 liters of liquid hydrogen were poured onto the ground by remote control.

"A spill of 21.5 liters of liquid hydrogen theoretically would supply enough gas so that the hydrogen-air combination formed would fall within the range of mixtures that can detonate. Because of losses and heterogeneous mixtures, it required a 35 liter spill to produce a detonation, with the detonator as the ignition source.* The 40 liter test, however, produced a detonation with both the strong and weak ignition sources."

The foregoing results show that even for the essentially complete confinement conditions used in these experiments, detonation is not easily produced until the hydrogen concentration is almost twice the minimum detonation limit cited above. In an unconfined state, achievement of detonation is even more difficult as illustrated by the following:

"An investigation was carried out by A. D. Little, Inc., to determine if an ignited hydrogen-air mixture would detonate in free space. A 5' diameter balloon containing 100 cubic feet of near stoichiometric hydrogen-air mixture and an 8' diameter balloon with a volume of 400 cubic feet were used. Explosive and flame sources, hot wires and sparks, at the center of the balloons, supplied the energy. Two grams of pentolite was the minimum charge necessary to produce a detonation. When the ignition energy was reduced by substituting a blasting cap with 1/2 gram of explosive, the blast yield was only 5% of that produced by the detonation. Application of energy by flame source, spark or hot wire, initiated combustion but there were no measurable pressures above ambient.*

"An 8' diameter balloon was also used to test the possibility of combustion transition from deflagration to a detonation. Only combustion resulted from the low energy initiators, and in no case did the transition occur. Because a hydrogen-air mixture in proper ratio and the presence of a sufficiently powerful shock wave source are necessary to produce a detonation, it is an unlikely occurrence in an unconfined state."

* Ed. note: Emphasis added.

On the basis of the material presented above, it is concluded that detonation of the hydrogen evolved from a complete metal-water reaction of the AARR core is incredible. Because of the large ratio of free air volume of the RCB to hydrogen volume, it is considered that essentially unconfined conditions exist. Under these conditions, a sufficiently powerful shock wave source is necessary to induce detonation. No such source can be envisioned as being present within the RCB.

3. Containment Potential of Reactor Containment Building for Hydrogen Detonation

In spite of the foregoing evidence against the credibility of a hydrogen detonation within the RCB, an analysis is presented which indicates potential effects upon the integrity of the containment. The following conservative assumptions are made:

(a) Twenty-five percent of all the evolved hydrogen collects in the dome of the RCB and forms a hydrogen-air concentration within the detonation limits.

(b) An ignition source is present of strength sufficient to initiate detonation.

(c) The detonation process is complete, utilizing all of the available hydrogen.

(d) Twenty percent of the total explosion energy is carried in the resulting blast wave. This assumption is conservative since it has been shown to be an upper limit for air-propagated blast waves emanating from detonation of high explosives (TNT).¹⁰ Half of the total blast wave energy is assumed to be directed toward and delivered to the RCB dome.

(e) Mechanical work is done on the dome by the blast wave at one hundred percent efficiency.

The thermal energy released from burning 790 cu ft of hydrogen (25% of the total evolved) is approximately 180×10^6 ft-lb. Of this total 18×10^6 ft-lb is available to do mechanical work on the RCB dome. The dome is constructed of a structural steel liner covered by reinforced concrete and has an elastic energy absorption capability of approximately 1.5 ft-lb per in.³ of "equivalent steel" material. At most, one-third of this capability is absorbed by overpressure conditions during an MCA. The dome "equivalent steel" volume is approximately 1.7×10^6 in.³, and hence to absorb the energy delivered by the blast wave, an additional strain energy absorption of $(18 \times 10^6 \text{ ft-lb} / 1.7 \times 10^6 \text{ cu in.}) \approx 10.5 \text{ ft-lb per in.}^3$ is required, of which 1 ft-lb per in.³ is provided by elastic deformation.

Using an extremely conservative value for the yield stress of the combined dome materials, 30,000 psi, the strain at yield will be approximately 0.12%. To provide the additional energy absorption required ($9.5 \text{ ft-lb per in.}^3$), plastic strain will occur. Again, conservatively assuming a constant stress of 30,000 psi for the additional strain, the total energy requirement of $11 \text{ ft-lb per in.}^3$ is achieved at a total strain of 0.5%

It is considered that the ultimate strain capability of the RCB dome materials will be at least 12%. Thus there is a large degree of confidence that rupture of the gas-tight liner cannot occur, even in the remote event (considered incredible) of hydrogen detonation as described above.

E. Total Failure of Utility and Emergency Power

As noted in Section IV-I above, the electrical power used in the AARR facility is provided in a manner that effectively precludes the possibility of a total loss of power to critical loads. For example each of the always-energized d-c pony motors which drive the main primary coolant pumps when the main a-c motors are not functioning is fed from a separate bus, and each bus in turn is furnished power from a separate battery charger. Loss of Class-III (utility) power to the battery charger results in the normally-floating battery supply for that charger furnishing power to the affected d-c bus. These measures, together with the design and safety criteria employed to implement them (see Section II above), ensure that a total simultaneous loss of both utility and emergency power can be considered incredible.

Nevertheless, the consequences of a total simultaneous loss of utility and emergency power have been considered.

Simultaneous failure of Class-III (utility) and Class-I (uninterrupted) power would result in an immediate automatic scram of the shim-safety plates. (No power is required for the insertion of these plates.) The reactor power would rapidly decrease as a result of the scram. Failure of power would result in automatic closure of the degasification system let-down valve* and its associated block valve (again no power is required for this action; these valves fail closed on loss of Class-III (utility) power.) However, in the time interval during which these valves are closing, an estimated 7 gal of the ~50 gal of water used to pressurize the primary loop would leave the system, and primary system pressure would decrease accordingly. The flow in the primary and secondary systems would coast down, reaching zero in about 30 sec and 30-40 sec, respectively.

Temperature and pressure control of the primary system would cease with loss of all power, and there is the possibility that the primary water temperature might decrease sufficiently, through heat losses to the Reactor Pool by conduction through the reactor vessel and piping walls, and to the secondary coolant in the main heat exchangers, to result in a reduction of system pressure to a value below atmospheric, through thermal contraction of the water. The primary system, however, is structurally able to accommodate this condition without collapse.

Fuel melting would be expected, but the length of time that the postulated power loss could be sustained without fuel damage is variable, depending upon the weather conditions at the time. These conditions would determine the secondary system operating conditions prevailing prior to the hypothetical loss of power.

The following is an excerpt from an Oak Ridge National Laboratory analysis of total loss of coolant flow in the HFIR,¹¹ which is applicable to the total power failure accident in AARR:

* Ed. note: See first footnote p. 249.

"An analysis of the loss of coolant flow accident...may be approached several different ways. The results show that the core will start to melt at some time between 10 seconds and 100 seconds following the start of flow coastdown. The rate of temperature rise of the core, considered on the average, will fall* between 75°F per second and 40°F per second at this time. Prediction of the sequence of events following melting is virtually impossible. The significant point is that the rate of temperature rise is not very rapid and, therefore, if dispersal of the material occurs by dripping or running out of the core it is probable that cooling will take place and little, if any, metal-water reaction would be expected. On the other hand, the material may stay together until the temperature is considerably in excess of the H₂O-Al reaction temperature and then be suddenly dispersed by a steam explosion in which case there would almost certainly be some reaction."

It should be noted that even though this accident is incredible in AARR, its consequences are not worse than those of the Maximum Credible Accident (MCA), described and analyzed in Section VII above.

* i.e. lie.

References for Section VIII

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3. R. W. Miller, et al, "Report on The SPERT I Destructive Test Program on an Aluminum, Plate-Type, Water-Moderated Reactor," IDO-16883 (June 1964).
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11. T. E. Cole to A. M. Weinberg, "High Flux Isotope Reactor - Safety Review Questions and Answers," ORNL-CF-65-11-29 (November 12, 1965), p. 20.

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APPENDIX AREACTOR PHYSICS CHARACTERISTICS*A-I. Introduction

The AARR will utilize reactor cores identical to those which have been operated successfully at 100 MW in the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory since early in 1966. The reactivity control system also will be identical to that of the HFIR. The design of the so-called "removable" and "semi-permanent" portions of the radial beryllium reflector will be based on the HFIR design. These zones supply most of the reactivity contributed by the external radial reflector.

Thus the reactor physics characteristics of the AARR will be quite similar to those of the HFIR, except with regard to differences in experiment facilities and irradiation targets in the Internal Thermal Column (ITC) and in the "permanent" beryllium zone. For this reason, where it is appropriate to do so, the physics analysis in the Preliminary Safety Analysis Report of the AARR has been based directly upon the extensive calculations and measurements performed for the HFIR by Oak Ridge National Laboratory personnel. A principal source of information is Reference 1.

Calculations have been performed to examine the differences in physics characteristics between the AARR and the HFIR, which are due to the above-mentioned differences in experiment facilities and targets. The methods of analysis are described in this Appendix, and the results of the calculations are summarized. The most important differences arise from differences in the internal flux-trap zone - the "island" as designated in HFIR, and the Internal Thermal Column (ITC) as designated in AARR. The HFIR is designed to be used as a production facility for transplutonium isotopes, and roughly one-half of the volume of the island is assigned to target-tube and target volumes. The ITC of the AARR, on the other hand, is for research production of much smaller quantities of transplutonium isotopes. The typical target material will be of higher mass number than the typical Pu-242 target of the HFIR. For effective production of the very heavy isotopes, it is imperative that the level of the thermal neutron flux in the ITC be kept high. Therefore, smaller volumes of target-tube and target material will be introduced into the ITC of the AARR. The net neutron poisoning by this material typically will be much smaller than the poisoning by the typical Pu-242 targets of the HFIR. Nevertheless, because there is a smaller metal content in the typical ITC insert of AARR,

* Other discussions pertaining to reactor physics characteristics of the AARR appear in Appendix C, Dynamic Behavior of Reactor Core, below and in Sections VI-B and VIII-B above. Other references which contain information of this category are References 2-5, reports on HFIR.

there will be a small net loss in reactivity relative to the island loading of the HFIR.

There are more beam tubes and through tubes in the permanent beryllium reflector of the AARR than in the HFIR, to provide a larger number of facilities for basic physics research. In spite of the use of a thicker beryllium reflector in the AARR, the additional streaming of neutrons in these tubes effectively removes a larger fraction of the neutrons which leak initially from the fuel zone, and therefore there is an additional reactivity loss. It is concluded, therefore, that the net available excess reactivity of the AARR reactor at the beginning of a core cycle will be somewhat less than that of the HFIR. It is estimated that the reduction will be roughly 1% in reactivity. The reactor shutdown margin will be enhanced by approximately this amount.

A-II. Methods of Reactor Physics Analysis

Differences in the reactor physics characteristics of the typical AARR system and the HFIR system were calculated in diffusion theory, using a 16-group cross section set based on the Hansen-Roach group partition of the neutron energy range.⁶ With the exception of updating of cross sections for some of the materials, the basic Hansen-Roach cross sections⁶ were used for groups 1-14, comprising neutrons of energies greater than 0.4 eV. The THERMOS⁷ program was used to calculate group cross sections for the two thermal-energy groups: Group No. 15 (0.1 eV - 0.4 eV), and Group No. 16 (0 - 0.1 eV).

A-III Reactivity-Control Worths of Control Plates

The data in Table A-1 have been taken from Reference 1, where appropriate. It is not anticipated that the reactivity-control worths of the control blades will be much affected by the differences in the composition of the ITC and of the permanent beryllium reflector. Therefore, the values listed in Reference 1 are reproduced here. It is expected that the shutdown margins will be somewhat larger in the AARR, as indicated in Footnote (a) of Table A-1.

The reactivity-control worths of the control plates in HFIR are estimated to be: (1) 0.169 Δk , with all plates fully inserted: and (2) 0.130 Δk , with the four shim-safety plates inserted and the shim-regulating control plate fully withdrawn. (See Reference 1, pg. 7-8). The largest combined differential reactivity-control worth of the 5 control plates occurs with the control plates symmetrically positioned with respect to core midheight. This worth is 0.021 $\Delta k/in.$ (Reference 1, pg. 7-8).

A-IV. Summary of Reactivity Accountability

(See Reference 1, Section 7.3.7.)

The operating core life of the HFIR has been demonstrated to be ~ 22 days at 100 MW, although the calculations summarized in Reference 1 indicate a core life of ~ 14 days. For this reason, with the exception of

TABLE A-1
SHUTDOWN MARGINS FOR THE HFIR^(a)

Reactor Loading	Shutdown Margin	
	All Control Plates Fully Inserted	Four Shim-Safety Plates Inserted
1. Maximum reactivity case, exclusive of accidental reactivity additions, with a new fuel element ^(b) and the following conditions:	0.059	0.020
(A) Extreme permissible loadings and distributions of fuel and burnable poisons resulting in maximum positive reactivity		
(B) Beryllium not poisoned		
(C) Maximum target reactivity		
(D) Zero power at 70°F		
2. Typical nominal case, with a new fuel element and the following conditions:	0.071	0.032
(A) Nominal loadings and distribution of fuel and burnable poison		
(B, C, and D) as in case 1		

(a) Shutdown margins are in units of Δk . The shutdown margins for the AARR are expected to be larger by roughly 0.01 Δk , at room temperature.

(b) "Fuel element" refers here to the two elements which comprise the core.

the values of the reactivity controlled by Xe-135 and Sm-149, only beginning-of-core-life values of the parameters are reproduced in Table A-2, below. One of the parameters in question is the reactivity effect of poisoning of the beryllium by buildup of Li^6 ; this too has been deleted from Table A-2.

As stated in footnotes to Table A-2, certain values listed in Table 7.3.5, Reference 1, have been replaced by values appropriate to the AARR. Additional detailed calculations will be performed for the Final Safety Analysis Report for the AARR.

A-V. Effects of Target Loadings in the ITC

The HFIR "island" zone, which is the "ITC" region of the AARR, typically is loaded with 31 target tubes containing pre-determined static targets. As a result of this heavy loading of target volume, there is much less water in the island, and the maximum reactivity gain in the event of a partial voiding of water in the island is estimated to be $0.013 \Delta k$. The AARR Internal Thermal Column is intended for research-production of heavy isotopes, and it is conceivable that a particular ITC insert will contain almost zero metal volume, when the highest irradiation flux is essential. Under these circumstances, on the basis of HFIR experiments, it is estimated that a reactivity gain of as much as $0.032 \Delta k$ (~ 4.5 dollars) could result from an optimum partial voiding of the ITC water. It is for this reason that the bulk-fluid velocity in the ITC has been set at 4 fps, to prevent accumulation of voids but yet offer time for control-plate action.

A calculation has been made of the difference in peak-power densities in the fuel when a 100%- H_2O ITC is used, in contrast to the island loading of the HFIR. It is calculated that the peak power density would be increased by $\sim 10\%$. The effects of this change on reactor safety will be evaluated.

For void fractions smaller than 0.2 in the 100%- H_2O ITC, the void coefficient of reactivity was measured to be $+0.06\%/ \%$ void (see Reference 1, pg. 7-6). The magnitude of the void coefficient is smaller when targets are present in the ITC. A one-dimensional computation by ANL yielded $+0.051\%/ \%$ void for the case of the 100%- H_2O ITC, but it is recognized that such calculations tend to underestimate the magnitude of the void coefficient in the ITC because they do not include the positive reactivity contribution of an increased effectiveness of the axial reflectors. For the typical ITC insert anticipated for AARR, with $\sim 21\%$ of ITC volume occupied by target tubes and targets, the void coefficient calculated by ANL is $+0.036\%/ \%$ void in the remaining ITC water. This value was obtained by one-dimensional calculations, and the correct value is somewhat larger. Nevertheless, the void coefficient is significantly reduced by the displacement of some of the ITC water with metal.

On the basis of calculations by Oak Ridge National Laboratory and by ANL, it is expected that the void coefficient of reactivity in the ITC is essentially unaffected by the degree of fuel burnup during the core

TABLE A-2

SUMMARY OF ESTIMATED REACTIVITY ACCOUNTABILITY FOR FRESH AARR

<u>Parameter</u>	<u>Reactivity, in units of Δk</u>
Clean reactor: no boron burnable poison; no targets; no Be poisoning; zero power at 70°F; no burnup	0.132 ^(a)
Boron burnable poison (2.8 gm B ¹⁰)	-0.05
Temperature deficit: zero power at 70°F, to operating conditions at 100 MW	-0.004 ^(b)
Typical target loading in ITC	0.006 ^(c)
Xe-135 and Sm-149 (100 MW), average over core life	~-0.052
Beam-tube flooding	~0
Fuel-loading tolerance ($\pm 1\%$)	± 0.001
Boron-loading tolerance ($\pm 11\%$)	± 0.004
Fuel-distribution tolerance ($\pm 10\%$)	± 0.005
Boron-distribution tolerance ($\pm 25\%$)	± 0.002
Minimum $k_{eff}-1$ (clean core, 100 MW)	0.066 ^(d)
Typical nominal $k_{eff}-1$ (clean core, 70°F)	0.088 ^(e)
Maximum $k_{eff}-1$ (clean core, 70°F)	0.100
Shutdown margins for typical nominal case: all control-plates inserted	0.081 ^(f)
shim-regulating plate withdrawn; 4 shim-safety plates inserted	0.042

(a) An estimated reduction of 0.008 in reactivity has been included.

(b) Best current estimate.

(c) All ITC targets in place; poisoning of target corresponding to poisoning by $\lesssim 1/2$ gm boron in the ITC.

(d) All ITC targets out (removal of target tubes and targets reduced reactivity by ~ 0.006); no Xe-135 or Sm-149; beam tubes and through tubes in place; Be poisoning not included.

(e) See Table A-1 for description of "typical nominal" case.

(f) An additional shutdown margin of 0.010 has been included. This is an estimated value, including the differences in worth of the beryllium and worth of the ITC at room temperature.

life.

A-VI. Temperature and Void Coefficients of Reactivity

The temperature coefficients of reactivity in the AARR with the HFIR-type fuel will be close to those of the HFIR under similar ITC conditions. The values quoted here are from Section 7.3.4 of Reference 1. Most of the coefficients were determined experimentally.

At 108°F, and with a 100%-H₂O ITC, the over-all isothermal temperature coefficient of reactivity is 0.0. The change in k_{eff} associated with a uniform increase in reactor water temperature from 79°F to 108°F is +0.0007, in the clean reactor. No experimental values of temperature coefficient of reactivity were obtained for conditions other than for the clean reactor. Analytical results for the HFIR indicate that, with the HFIR target in the island, the temperature coefficient of reactivity from the fuel-zone alone remains negative but becomes somewhat smaller in magnitude over the life of the core.

In the event of a rapid reactivity excursion, the fuel-zone would be heated preferentially, providing a negative feedback coefficient of reactivity. The calculated isothermal fuel-zone temperature coefficient of reactivity in the clean reactor is $-6.0 \times 10^{-5}/^{\circ}\text{F}$ at 79°F and $-11.7 \times 10^{-5}/^{\circ}\text{F}$ at 155°F for the case of a 100%-H₂O ITC. (See Table A-3.)

Reactivity coefficients of uniform voidings in coolant water in the fuel regions have been measured. The void coefficients in the inner and outer fuel elements are -0.188%/coolant void, and -0.384%/coolant void, respectively.

A-VII. Variations of Flux and Power Distributions During the Fuel Cycle

The following paragraphs have been reproduced from Section 7.3.3, "Fuel Cycle Analysis," of Reference 1, pg. 7-5. In actual operation, the reactivity worth of the Xe-135 was found to be in agreement with the calculated value for the HFIR. However, the experienced core life has been longer. For this reason, there are uncertainties about the calculated time-dependence of k_{eff} . For the Final Safety Analysis Report, Argonne National Laboratory will perform an independent analysis of the fuel-cycle history, for AARR conditions of ITC and beryllium facilities. Nevertheless, the present HFIR calculations are informative and valuable in describing various effects of reactor operating history. The four figures referred to in the quoted paragraphs, below, Figs. A-1 to A-4, are included for completeness.

"Fuel Cycle Analysis"

"Fuel cycle calculations were made to help determine the proper fuel and burnable-poison loadings and distributions and to investigate variations with time in such

Table A-3

Temperature Coefficients of Reactivity for the Clean Reactor

(Reference 1, Table 7.3.2)

Temperature Conditions (100%-H ₂ O ITC)	Temperature Coefficient of Reactivity ($\Delta k/k$)/°F	
	At 79°F	At 155°F
Over-all isothermal	$+2.3 \times 10^{-5}$	-2.1×10^{-5}
Fuel zone only	-6.0×10^{-5}	-11.7×10^{-5}
Non-fuel zones (by difference)	$+8.3 \times 10^{-5}$	$+9.6 \times 10^{-5}$

parameters as neutron flux and power distribution and reactivity. Figure A-1 shows typical radial power distributions at different times in the fuel cycle; Fig. A-2 shows how the average thermal-neutron flux in a typical island target varies with time; and Fig. A-3 shows, in addition to several clean-core radial flux distributions, a comparison between clean-core and end-of-cycle radial thermal-neutron flux distributions. As indicated by these curves, the island flux and the peak power density are a maximum at the beginning of the fuel cycle and decrease 2 or 3% during the fuel cycle.

"The radial power distribution curves in Fig. A-1 are typical for an average position along the length of the core. At the horizontal midplane the relative power density near the outer edge of the outer annulus is somewhat greater because of a symmetrical longitudinal flux peaking that takes place as the control plates are withdrawn from the core. Also, at the outer corners of the core the relative power density at the end of the fuel cycle tends to be greater than shown in Fig. A-1. During a major portion of the fuel cycle, the thermal-neutron flux at these corner regions is depressed by the presence of the black regions of the control plates, resulting in a relatively low local percentage of fuel burnup. At the end of the fuel cycle, when the plates are fully withdrawn, both the fuel concentration and neutron flux are high, resulting in a somewhat higher power density than indicated. Additional calculations have been made which indicate that these longitudinal

FIG. A-1
TYPICAL RADIAL POWER
DISTRIBUTIONS DURING A CYCLE

FIG. A-2
AVERAGE THERMAL-NEUTRON FLUX
IN THE TARGET DURING A CYCLE

FIG. A-3
TYPICAL RADIAL THERMAL-NEUTRON-FLUX DISTRIBUTION

UNCLASSIFIED
ORNL-DWG 63-8106

FIG. A-4
REACTIVITY VARIATIONS DURING A CYCLE

peaking effects will not produce relative peak power densities greater than those shown in Fig. A-1.

"Reactivity variations during the fuel cycle are shown in Fig. A-4. As indicated by curves A and D, at the beginning of the fuel cycle the burnable poison controls about $0.04 \Delta k$;* at the end of the core life it is worth about $0.007 \Delta k$, thus resulting in a loss of about two days of core lifetime.

"The inclusion of a burnable poison in a reactor core often gives rise to a situation where the reactivity of the core will increase as some of the poison burns out. Curve A of Fig. A-4 shows that this is not true in the case of HFIR operation. Curve A, however, does not represent the maximum possible reactivity. During the first seven days of full-power operation, the maximum k_{eff} , with the control plates out, occurs approximately four days after a shutdown from full power, at which time the combined poisoning effect of xenon and samarium is at a minimum. Curve B of Fig. A-4 is a plot of k_{eff} achieved in this manner vs the time in the cycle at which the power was stepped from 100 to 0 Mw. Even in this case, k_{eff} is never greater than at the beginning of the fuel cycle. If, after being stepped from 100 to 0 Mw, the core is maintained down for ten days or more instead of just four days, the xenon is essentially gone and the promethium (Pm^{149}) has nearly all decayed to Sm^{149} , resulting in somewhat lower values of k_{eff} as depicted by curve C."

A-VIII. Miscellaneous Nuclear Parameters

Thermal neutron fluxes computed by Oak Ridge National Laboratory for certain hypothetical HFIR conditions are reproduced in Table A-4. Also in Table A-4 are values of the mean-prompt neutron lifetime, and the effective delayed-neutron fraction. These values should be little affected by the differences between HFIR and AARR.

Argonne has calculated values for the peak thermal neutron flux in the ITC for the idealized case where the control plates are withdrawn

* Note that there is some inconsistency between the value of $0.04 \Delta k$, shown in Fig. A-4 as the reactivity controlled by boron in the fresh reactor, and the value of $0.05 \Delta k$ listed in Table A-2. The latter value ($0.05 \Delta k$) is the best present estimate. Figure A-4 should be corrected accordingly. (ANL footnote)

TABLE A-4
CALCULATED NEUTRON FLUXES
AND NEUTRON-KINETICS PARAMETERS IN AARR(a)

(Operating Conditions; 100 MW)

Thermal Neutron Flux, neutrons/cm ² -sec	
Maximum in 100%-H ₂ O ITC	5.5 x 10 ¹⁵
Maximum in unperturbed reflector	
Beginning of fuel cycle	1.1 x 10 ¹⁵
End of fuel cycle	1.6 x 10 ¹⁵
Mean Prompt-Neutron Lifetime, μ sec	
Beginning of fuel cycle	35
End of fuel cycle	70
Effective Delayed-Neutron Fraction, β_{eff}	0.0071

(a) See Table 7.3.1, Reference 1. These values are assumed to be the same as for the HFIR, for the same conditions.

and "white" (aluminum) followers are in place. For the fresh reactor, the peak thermal neutron flux (0 to 0.4 eV) in the 100%-H₂O ITC is 4.4×10^{15} neutrons/cm²-sec. Other calculations have been performed for the case where all "black" portions of the control plates are inserted. This resulted in a tilting of the radial power distribution toward the ITC, and the peak thermal neutron flux in the ITC rose to approximately 6×10^{15} . When a typical ITC loading is introduced (see Section A-V, above), the calculated value of the peak thermal neutron flux in the ITC is reduced by approximately one-fourth (computations by ANL).

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3. T. E. Cole to A. M. Weinberg, "High Flux Isotope Reactor - Safety Review Questions and Answers," Memorandum ORNL-CF-65-11-29 (November 12, 1965).
4. ibid., Ref. 3, Supplement No. 1 (February 1, 1966).
5. ibid., Ref. 3, Supplement No. 2 (August 15, 1966).
6. G. E. Hansen and W. H. Roach, "Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies," LAMS-2543 (December 1961).
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APPENDIX B

SURVEY OF METAL-WATER REACTION

DATA RELEVANT TO AARR

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APPENDIX BSURVEY OF METAL-WATER REACTION DATA RELEVANT TO AARR*B-I. General

The Argonne Advanced Research Reactor is a light water moderated and cooled, beryllium reflected, aluminum-U₃O₈ cermet fueled reactor. Because of the combination of materials involved, there exists the possibility, under abnormal operating conditions, of the occurrence of several energy- and hydrogen-generating reactions. Reactions between beryllium and water, between aluminum and water, and between aluminum and U₃O₈ must be considered.

Experimental information does not exist over the entire range of conditions of interest. In the following survey the available data have been summarized briefly and extrapolations made where necessary. Attempts have been made to make the more conservative choice where alternatives existed.

No discussion of specific incidents has been included here. Such questions as which parts of the reactor may become hot enough to react extensively, what surface areas are involved, how quenching, limited steam flow, and external cooling may be effective, and many others, must be answered on the basis of careful study of each individual case. This survey is intended solely to provide guidance in calculating rates of reaction, heat generation, and hydrogen evolution once estimates have been made of the reacting area and the sample temperature.

B-II. Aluminum-Water and Aluminum-U₃O₈ (Thermite) ReactionsA. Introduction

The reactor contains 2.1×10^5 gm (460 lb) of aluminum chiefly divided between the core and the control rods (see Table B-1). The initial surface areas are also given in Table B-1.

The reaction between aluminum and water is described by the equation

* The material presented in this Appendix was prepared by L. Leibowitz and R. E. Wilson of ANL, Chemical Engineering Division.

TABLE B-1
SUMMARY OF CORE DATA

<u>SECTION I</u>	<u>WEIGHT</u> <u>(lb)</u>	<u>AREA</u> <u>(in.²)</u>	<u>POTENTIAL CHEMICAL</u> <u>ENERGY</u> <u>(MW-sec) (a)</u>	<u>POTENTIAL H₂</u> <u>EVOLUTION</u> <u>(liters STP)</u>
Aluminum in Meat and associated clad of fuel plates	160.0	66,962	1,277	0.90×10^5
<u>SECTION II</u>				
Other parts in core: combs, side plates, and end fittings				
Al	136.3	20,253	1,088	0.767×10^5
<u>SECTION III</u>				
Control Rods				
Al	154.7	14,500	1,234	0.871×10^5
Eu ₂ O ₃	38.1	-	-	-
Ta	23.9	-	24	0.033×10^5
<u>SECTION IV</u>				
Support Grid, ring adapter, control cyl. supports				
304 Stainless Steel	115.5	2,527	31	0.268×10^5
<u>SECTION V</u>				
Reflector				
Be	2942	35,000	53,000	3.32×10^5
Al can	9	1,400	72	0.05×10^5

(a) The reaction products are assumed to be Al₂O₃, U₃O₈, Eu₂O₃, Ta₂O₅, (Fe, Ni, Cr)₃O₄, BeO, and H₂ as appropriate.

Some relevant physical properties¹ of the metal and its oxide are listed in Table B-2.

Table B-2

Properties of Aluminum and Aluminum Oxide¹

	<u>Al</u>	<u>Al₂O₃</u>
Melting point, °K (°F)	932 (1220)	2315 (3700)
Boiling point, °K (°F)	2740 (4480)	- ^a
Heat of fusion, kcal/mole	2.55	28.3
Heat of vaporization, kcal/mole	70.7	456

^aVapor pressure reaches 1 atm at 3800°K (6400°F). The vapor does not consist of Al₂O₃ molecules.

The heat generated by reaction at one atmosphere is about 4200 cal/gm Al. This is roughly true up to about 2800°K (4580°F). At that temperature the heat liberated falls rapidly as the temperature increases. This is due to the formation of hydrogen atoms and other dissociation products at the high temperatures.

B. Slow Oxidation Reactions

Measurements have been made of the rate of reaction of molten aluminum and steam at one atmosphere.² Below the melting point the reaction rate is so low that it may be completely ignored. At temperatures up to 1300°C (2370°F) the oxide film which forms is highly protective and the rate of metal-water reaction decreases rapidly with time. Between 1400°C (2550°F) and 1600°C (2910°F) the reaction rate is constant, the oxide providing no protection to the underlying metal. In the higher temperature range of 1400°C (2550°F) to 1600°C (2910°F) the reaction rate is described by the equation:

$$\text{Rate} = 2.95 \times 10^8 \exp (-73,500/RT) \frac{\text{mgm Al}}{\text{cm}^2\text{-min}},$$

$$R = 1.987 \frac{\text{cal}}{\text{mole-deg K}}.$$

As an illustration, Table B-3 gives experimentally-measured amounts of metal reacted with pure steam at one atmosphere in 10 min at a series of temperatures.

Table B-3
Reaction of Aluminum with Steam in 10 Minutes at
Various Temperatures

<u>Temperature</u> <u>°C (°F)</u>	<u>Al Reacted</u> <u>in 10 Minutes</u> <u>(mgm/cm²)</u>
1200 (2190)	0.76
1300 (2370)	1.37
1400 (2550)	1.28
1500 (2370)	3.19
1600 (2910)	7.7

Experimental errors are fairly large at the lower temperatures. Thus, if upon melting of the reactor fuel there is a twofold increase in the aluminum area available for reaction with steam, there will be a total of $(66,962 \times 2)$ or 133,920 in.² (8.6×10^5 cm²) reacting. At 1500°C (2370°F), for example, the rate of reaction will thus be about 2.76×10^2 gm Al/min or 0.61 lb Al/min.

C. Ignition and Combustion

At temperatures above 1600°C (2910°F), aluminum samples were found to ignite in steam. Experimental ignition delay times are listed in Table B-4 for 0.8 cm diameter spheres.

Table B-4
Ignition Delays of 0.8 cm Diameter

<u>Surface Temperature</u> <u>°C (°F)</u>	<u>Time to Ignition</u> <u>(min)</u>
1600 (2910)	18
1650 (3000)	9
1700 (3090)	5
1750 (3180)	0

Thus, as soon as an aluminum sample reaches 1750°C (3180°F) it will ignite and burn. The burning seems to be in the vapor phase with generation of white oxide smoke. The combustion time of the pellets is given by the equation

$$D_0 = 0.16 t + 0.336$$

where D_0 is the initial diameter of the pellet in centimeters and t is the lifetime in seconds. This is simply an experimental equation which was fitted to measured burning times. No physical significance should be given to the constants in the above equation.³ The equation for lifetime of particles shows the reaction rate to be $0.96 \text{ gm Al}/(\text{cm}^2) (\text{min})$. This rate law was verified experimentally in the range 0.96 to 0.55 cm diameter.

Experiments with extremely small aluminum particles (0.036 cm diameter) using laser beam heating,⁴ have indicated that for very small particles the ignition temperature is not 1750°C , as was found with the larger ones, but was very close to the melting point of Al_2O_3 (2050°C , 3720°F). Furthermore, that work also indicated that the rate of reaction is controlled simply by gaseous diffusion. The theoretical reaction rate (confirmed by one experimental point) is $0.206/D \text{ gm Al}/(\text{cm}^2) (\text{min})$, where D is the particle diameter in centimeters. This is the form of the rate law obeyed by simple hydrocarbon fuels.

Thus, there are two laws describing the rate of combustion of aluminum particles in water. The "large particle" law supported by experiment down to a diameter of 0.55 cm, and a "small particle" law supported by experiment at a particle diameter of 0.036 cm. Somewhere between 0.55 and 0.036 cm a change in the reaction mechanism seems to occur. It is impossible at present to decide where and how this takes place. Some decision must be made, however, in order to carry out a safety analysis of aluminum combustion. One possibility for dealing with this problem is to find the diameter at which the two rates are equal. That is

$$0.96 = \frac{0.206}{D}$$

and

$$D = 0.21$$

It would then be possible to use the large particle law, rate = $0.96 \text{ gm Al}/(\text{sq cm}) (\text{min})$, and an ignition temperature of 1750°C for particles larger than 0.21 cm diameter, and the small particle law, rate = $0.206/D \text{ gm Al}/(\text{sq cm}) (\text{min})$, and a 2050°C ignition temperature below 0.21 cm diameter.

This should not be taken to imply that large particles will burn down to 0.21 cm diameter and then go out. Self-heating would be enough to raise the temperature above 2050°C. Indeed, measured burning temperatures of large samples have indicated temperatures above 2000°C. Table B-1 summarizes energy release and hydrogen evolution data for various metal components in the reactor. These are listed roughly in decreasing order of likelihood of reaction. The heats listed are simply those for formation of H₂ and the metal oxide from the metal and water.

D. Aluminum-U₃O₈ (Thermite) Reaction

Use of a U₃O₈-aluminum mixture leads to a fuel element in which there is the potential for a highly exothermic reaction. It is necessary to evaluate the safety implications of this potential energy source.

The overall reaction involved may be described by the following equation:⁵

This reaction may be further divided into two steps:

Rough calculations indicate that enough heat is evolved in the overall reaction, if released adiabatically, to cause a temperature increase of about 2000°C for 75 to 85 w/o U₃O₈ specimens. About three-fourths of this heat is produced by the reduction of the U₃O₈ to UO₂. Very high temperatures have indeed been reported by Fleming and Johnson⁶ at high U₃O₈ concentrations. Work at ANL has not been able to completely reproduce Fleming's results. Some rapid reaction was found, however, with 75 to 85 w/o U₃O₈ pellets.

With 40 w/o U₃O₈ pellets which were plunged into preheated furnaces, self-heating rates of only 100 to 200°C/sec were observed. Sample temperatures reached only a few hundred degrees above the furnace temperature of 1250°C or 1000°C. Several HFIR fuel samples were similarly tested in the laboratory furnace;⁷ there was no evidence of self-heating due to the thermite reaction.

It is concluded that despite the energy potentially available, the reaction is sufficiently slow that no hazard is involved, particularly at the lower U₃O₈ concentrations involved in typical reactor fuels (about 27 w/o U₃O₈ including clad). The potential heat available from the thermite reaction is included in the core aluminum-water reaction heat given in Table B-1, since upon complete oxidation the reaction products are the same.

B-III. A Summary of TREAT Results with Aluminum-U₃O₈ Cermet (HFIR) Fuel Plates

A. Introduction

A number of experiments have been carried out in the TREAT reactor using aluminum-U₃O₈ cermet (HFIR) fuel plates.^{7,8} Several important findings from these experiments are summarized briefly below.

B. Fuel Meltdown

In an initial series of twelve experiments, HFIR fuel samples in water were exposed to neutron bursts in TREAT at fission energies ranging from 133 to 1082 cal/gm⁷. It was found that "the samples retained their plate-like shape at fission energy inputs as high as 440 cal/gm, even though 230 cal/gm of energy is sufficient to melt the aluminum cladding and the aluminum in the matrix of the fuel." It was estimated that these fuel plates reached peak temperatures of up to about 1400°C. (It is the fuel plate temperature that must be considered in judging the consequences of an excursion). The reactor period ranged from 152 to 100 msec for these relatively unreacted samples. It should be pointed out, however, that if a shorter period is to be considered, a temperature of 1400°C would be reached at energies lower than 440 cal/gm since heat losses would be smaller.

Damage to the fuel plates appeared to be progressively more severe as the fission energy input increased, but the plates still maintained very nearly their original shape. Moreover, of the seven fuel samples which received 440 cal/gm of energy or less, only one reacted to an extent greater than 2.5%. That sample, which gave 11.8% reaction, was exposed in 285°C water with 1,000 psia overpressure (964 psia H₂O and 36 psia He).

"The samples subjected to energies greater than about 500 cal/gm lost their resemblance to plates.For those exposed in 30°C water (20 psia He), the residues consisted of (1) a single large globule containing most of the U₃O₈, (2) some smaller particles which were oxide-coated metallic aluminum, and (3) a fine powder which was principally alumina (α Al₂O₃).

"The changes in the samples that occurred in the high energy experiments in heated water were different in character from those that occurred in the corresponding experiments in 30°C water. ...The samples in these runs were almost completely reacted and were fused into single large globules with large internal void spaces. Small quantities of very finely divided alumina were also found". The conclusions from these results is that these cermet fuel plates maintain their shape until ignition and then extensive reaction and fuel slumping occurs.

A second series of five experiments⁸ in which high speed motion pictures were taken during the TREAT burst supports this conclusion. It was observed that the HFIR fuel plates do not fuse into globules until

extensive vapor-phase combustion of the aluminum has begun.

C. Fuel Plate Break-Up

From high speed motion pictures taken during TREAT transients, it has been found that in 30°C water fragmentation occurs only after vapor phase combustion of the aluminum has begun. The fragmentation apparently results from the motion of the water column. The water is initially forced upward by the formation of vapor bubbles. The return of the water collapses the bubbles and causes the sample to fragment. A similar phenomenon has been observed in the laboratory when small metal particles at the bottom of a water column were heated with a laser beam.⁹ Prior to these photographic observations, fragmentation was thought to occur before ignition in TREAT experiments. In heated water, motion pictures have not been taken during TREAT transients. Examination of residues from tests in heated water shows that even when 90% reaction occurs, most of the residue is found in a single fused mass. The violence of the water vapor bubble collapse must be much less in heated water.

D. Ignition

It was found in laboratory experiments² that 0.8-cm dia. aluminum spheres would ignite after an 18 min delay, if heated to 1600°C. In the TREAT experiments, ignition of HFIR plates would occur if enough fission energy were supplied to heat the samples to about 1400°C. In both cases, chemical reaction provided sufficient energy to raise the sample temperature to about 1750°C, at which temperature immediate ignition occurred. There are two reasons why the fuel plates in TREAT need not be heated as much as the laboratory samples for ignition to take place: (1) chemical reaction causes greater temperature rises since the thin fuel plates have a much greater surface-to-volume ratio than the spheres and (2) the fast heating of the nuclear pulse allows less time for heat losses than does the slower induction heating in the laboratory experiment. The two values for ignition temperature appear to be consistent when these differences are taken into account. The most reasonable temperature to choose for ignition of these fuel plates under fast transient conditions is 1400°C if only nuclear heating is included. For low heating rates, such as would occur in a loss-of-coolant incident, an ignition temperature between 1600 and 1750°C (with delays*) would be more appropriate.

B-IV. Beryllium-Water Reactions

A. Introduction

The reactor contains 1.33×10^6 gm (2942 lb) of beryllium in several sections as the reflector. Many cooling channels are provided which, in addition to their normal function, result in a large surface area being available for reaction with water under abnormal conditions. The total surface area potentially available for chemical reaction with water is 2.26×10^5 cm² (35,000 in²).

* See Table B-4 above.

The reaction of beryllium with water is described by the equation

Some physical properties¹ of the metal and its oxide are given in Table B-5.

Table B-5

Properties of Beryllium and Beryllium Oxide¹

	<u>Be</u>	<u>BeO</u>
Melting point, °K (°F)	1556 (2340)	2823 (4620)
Boiling point, °K (°F)	2757 (4508)	- ^a
Heat of fusion, kcal/mole	2.8	15.1
Heat of vaporization, kcal/mole	71.1	143.5

^aThe pressure above BeO reaches 1 atmosphere at about 4123°K (6960°F). The vapor does not consist simply of BeO molecules.

The heat generated by metal-water reaction at the melting point of beryllium is 9450 cal/gm Be.

At high temperature (above 1300°C, 2370°F) in excess steam, small amounts of the volatile compound Be(OH)₂ will form. This reaction may be ignored in the present case.¹⁰ In discussing the heat generated by the metal-water reaction, care must be exercised that the correct products are specified. For example, H₂ will dissociate into H atoms quite extensively at high temperatures. Similarly BeO will form other substances at sufficiently high temperatures. The heat of reaction and the volume of gas produced will be markedly altered. One cannot extrapolate indefinitely without considering the details of each case. Failure to appreciate this point has led several workers into errors.

The Be-H₂O reaction will be discussed in two sections; first, slow oxidation reactions which are observed up to roughly the melting point of the metal and second, ignition and combustion which may occur above that temperature.

B. Slow Oxidation Reactions

Some work on the reaction between beryllium and water was carried out by Aylmore, Gregg, and Jepson¹¹ at temperatures between 500°C (1020°F) and 750°C (1380°F). Their work was done chiefly in pure water vapor at a pressure of 12 Torr (0.016 atm). They report that at 550°C (1020°F) and 600°C

(1110°F) the reaction rate was initially high but decreased rapidly with time. At 650°C (1200°F) and above, however, the rate of reaction first decreased and then increased, indicating breakaway and subsequent non-protective oxidation. (Breakaway is a term used to indicate an increase in metal oxidation rate resulting from a loss of protectiveness on the part of the oxide. Reaction rates after breakaway are higher than before breakaway but do not necessarily produce excessive corrosion.)

A study of the beryllium-water reaction was carried out by Blumenthal and Santy¹² at higher pressures and temperatures. This work was done in pure water vapor at pressures between 150 Torr (0.2 atm) and 650 Torr (0.9 atm) and at temperatures from 780°C (1440°F) to 1010°C (1850°F). This range of conditions is closer to what may be of importance in reactor safety studies.

At the lowest temperature used in the Blumenthal and Santy experiments, a breakaway type of reaction was found. No information is available at 650°C (1200°F) at water vapor pressures near one atmosphere. Based on the experimental data at hand, it should probably be assumed that breakaway oxidation will occur at 650 C and above even at the higher pressures.

Data are presented by Blumenthal and Santy for rates of reaction at 780°C (1440°F), 830°C (1530°F), 930°C (1710°F) and 1010°C (1850°F) at 650 Torr (0.9 atm) water pressure. The results of the experiment at 1010°C (1850°F) are virtually identical with those at 930°C (1710°F). The authors state (p. 66), "At temperatures above 1010°C the beryllium-water reaction became too rapid to follow quantitatively in the steady flow system." Because of this statement and the fact that the results at 930°C (1710°F) and 1010°C (1850°F) are almost identical, the values given by Blumenthal and Santy at 1010°C (1850°F) are suspect. It may be that even at that temperature the limits of measurement by the equipment being used had been reached. In the following discussion the 1010°C (1850°F) experiment has been rejected as unreliable.

The experiment at 780°C (1440°F) shows the rate to fall rapidly as the temperatures are lowered much below 800°C (1470°F). This is also indicated by data of Aylmore, et al.¹¹ For purposes of extrapolation to temperatures above 930°C (1710°F) the reaction rate data of Blumenthal and Santy at 830°C (1530°F) and at 930°C (1710°F) have been extrapolated, using an Arrhenius plot, to the melting point of Be. In Table B-6 are given some of these reaction rates. It should be kept in mind that this table is based on only two experimental rate values. Unfortunately, this is the only information presently available.

Table B-6Extrapolated Rate of Reaction Between Beryllium and Steamat 650 Torr at Selected Temperatures

<u>T</u> <u>°C (°F)</u>	<u>mgm Be/cm²-min</u>
800 (1470)	1.16
900 (1650)	4.23
1000 (1830)	12.7
1100 (2010)	32.2
1200 (2190)	72.4
1284 (2340)	132.0

The reaction rate was shown by Blumenthal and Santy to be dependent on the pressure of water vapor. At 780°C (1440°F) the rate is proportional to the 0.6 power of water vapor pressure while at 830°C (1530°F) and 930 C (1710 F) it is roughly proportional to the first power of pressure. If this pressure dependence holds up to the melting point of Be, the rates tabulated must be increased by $76/65 = 1.17$ for application at one atmosphere of water vapor pressure.

Using this information it is fairly simple to calculate the time required for 1% of the Be in the reactor to be oxidized at one atmosphere at various temperatures. This calculation has taken the surface area available to be $2.26 \times 10^5 \text{ cm}^2$ (35,000 in. ²) and the amount of Be present to be $1.33 \times 10^6 \text{ gm}$ (2942 pounds). It is also possible to estimate the rate of energy liberation by reaction at any temperature. Some illustrative calculations are given in Table B-7.

Table B-7Reaction Rates in Steam (at One Atmosphere) of Berylliumin AARR Based on Original Surface Area

<u>T</u> <u>°C (°F)</u>	<u>Be Reacted</u> <u>(lb/min)</u>	<u>Time for 1% of</u> <u>Be to React</u> <u>(min)</u>	<u>Energy Liberated</u> <u>(MW-sec/min)</u>
800 (1470)	0.67	43	12.0
1000 (1830)	7.36	4.0	132.0
1284 (2340)	76.6	0.38	1375.0

The surface area used in all these calculations was that which was initially available. There are many small openings and channels involved and the availability to water vapor of the underlying Be could be quite markedly lowered by the formation of a relatively small quantity of oxide. The molar volume ratio of oxide to metal is 1.68.¹³ There is thus a substantial increase in the volume on oxidation. The oxide has been described by Blumenthal and Santy as follows: "...the samples were coated with a thick crusty outer layer of beryllium oxide... the oxide material itself holds together fairly well and does not easily tend to powder." The degree to which the oxide can be depended upon to hinder further reaction in small channels cannot be estimated at present. A conservative approach, in the absence of further information, seems to require the assumption that the oxide is not effective in this respect.

C. Ignition and Combustion

Studies of ignition of massive pieces of beryllium have been reported by Blumenthal and Santy¹² in hydrogen/oxygen, carbon monoxide/oxygen, methane/air, and hydrogen/air flames. The most pertinent of these is the hydrogen/oxygen flame, which, after combustion gives an atmosphere composed of steam plus either hydrogen or oxygen depending on the original stoichiometry of the fuel. Ignition of small samples has been reported, under various conditions by Kuehl,¹⁴ Maček,¹⁵ and others. Relatively little work has been done with massive samples in systems containing appreciable amounts of water. The work of Kuehl and of Blumenthal and Santy indicates that beryllium will ignite in water vapor at about its melting point of 1284°C (2340°F).

Kuehl measured ignition temperatures using electrically-heated 0.020 in. diameter beryllium wire in 100 psia oxygen-water vapor mixtures. He found that the pure oxygen ignition temperature of about 2500°C (4500°F) was lowered to roughly 1400°C (2550°F) by the addition of 25% water vapor. Increasing the amount of water to 100% lowered the ignition temperature slightly to about 1300°C (2370°F), roughly the melting point of beryllium. Unfortunately, he did no work at 1 atm in these mixtures.

Blumenthal and Santy carried out experiments in which 3/16 in. diameter, 1/2 in. long beryllium cylinders were exposed to a hydrogen-oxygen flame. Fuel-rich ($O_2/H_2 = 0.33$) and oxygen rich ($O_2/H_2 = 0.99$) flames were used. In one experiment only was a stoichiometric flame ($O_2/H_2 = 0.50$) used.

They found that in the fuel-rich experiments, "as the sample temperature approached the melting point of beryllium, a significant release of airborne beryllia occurred at the hot spots...(which) were as much as 600°C (1000°F) above the bulk temperature of the sample, but never obtained a temperature of greater than 2000°C (3600°F). In the oxygen-rich flames the beryllium samples were found to ignite and burn with a very hot vapor phase diffusion flame under all but the lowest initial heat flux conditions. Samples were ignited under conditions in which the thermal balance excluding the energy generated by chemical reaction, would leave the sample at a steady

state temperature below the beryllium melting point...when general ignition and vapor phase combustion occurred, a significant fraction of the total beryllium sample was converted to airborne beryllium oxide smoke in a matter of a few seconds." The results presented for the experiment with $O_2/H_2 = 0.5$ show that this stoichiometric flame is much more reactive than the fuel-rich flame.

Taking into account the fact that the low temperature rate of reaction of beryllium with water is higher than that with oxygen,¹¹ one must conclude that if the metal is allowed to be heated to near its melting point it will ignite and burn in steam. The rate of reaction will very likely be controlled by vapor phase diffusional processes. It is unfortunate that no experimental results in pure steam are available.

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APPENDIX C
DYNAMIC BEHAVIOR OF REACTOR CORE

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APPENDIX C

DYNAMIC BEHAVIOR OF REACTOR CORE

(Theoretical Model and Survey Calculations)

C-I. Introduction

The transient behavior of the AARR (HFIR) core has been examined extensively through calculation programs accomplished at ORNL. These results have been obtained using a theoretical model which comprises instantaneous relationships between neutronic behavior and coupled reactivity effects arising from transient thermal conditions in the reactor. The following discussions are largely extracted from relevant material found in References (1) and (2) and the addenda and supplements thereto. Identification of extracted material is made throughout this Appendix by quotation marks and indentation.

Major departures from exact replication of ORNL work are those occasioned by: (1) lack of detailed definitions at this time of philosophies for operating conditions which depart from the normal mode to be used in the AARR facility and, (2) a different concept of the hypothesized ITC voiding accident caused by differences in the use of the ITC for irradiation experiments.

With respect to the first of the foregoing differences in the utilization of the two facilities, two additional operating modes are provided for HFIR wherein the reactor vessel head is opened and the reactor is operated in a critical condition. These are described as mode numbers two and three in References (1) and (2). Although it is expected that AARR operation at low power, without pressurization and perhaps without forced coolant flow will be employed, detailed definitions of operating conditions and reactor protective measures have not been completed at this time. Thus, the only currently available HFIR data relevant to AARR are those obtained for operation under mode number one, which for the purposes of the safety analysis may be briefly outlined as follows. Mode 1 is for pressurized operation (approximately 600 psi) at a power level compatible with coolant flow over the range between full coolant flow and ten percent flow. At full flow, the power level can range from zero to 100 MW. The power level at ten percent flow can range from zero to 10 MW (utility power outage condition).

The differences in concept of a hypothesized ITC voiding accident (item 2 above) arise primarily from the differences in design flow rates of the bulk of the ITC coolant. The HFIR facility operates with the ITC zone of the reactor heavily loaded with target material, thereby imposing a substantial heat removal load upon ITC coolant. This requires that the coolant flow rate therein be similar to that in the fuel zone of the reactor core. Consequently, if voids were to be present within the bulk reactor coolant, it is possible that they could be swept rapidly into the ITC thereby producing a reactivity addition of a maximum magnitude of $1.3\% \Delta k/k$ in

approximately 30 msec.

A converse situation exists for the planned utilization of the ITC in the AARR facility, wherein a comparatively light loading of irradiation samples will be exposed to ITC fluxes. These samples will be contained within gas or hydraulic rabbit tubes or sample baskets and only a small volume of high speed coolant will be employed to provide necessary heat removal. Thus, the bulk of the ITC water in the AARR will experience minimal heat removal requirements and the flow rate requirement is correspondingly reduced. Therefore, the bulk flow of water through the ITC in the AARR has been selected so as to minimize the rate at which reactivity could be added to the system by sweeping voids into that region with the bulk coolant. The velocity selected, 4 fps, is sufficiently high to ensure that voids that enter that region will continue to be swept along with the coolant and would not accumulate. With this velocity established, the reactivity gain associated with uniform voiding of the bulk ITC coolant in this manner would occur in a time span of approximately 375 msec. This exceeds the corresponding HFIR time by a factor of ten. However, because of the greater anticipated water content of the ITC in AARR, the potential for a larger total reactivity addition exists. To illustrate (it is not intended to operate AARR under this condition), if the ITC contains only water and void, then a maximum reactivity gain no greater than 3.2% $\Delta k/k$ is achieved with a uniform void fraction of 0.70. With normal loading of the ITC with experiment facilities and materials, the maximum possible reactivity gain associated with uniform voiding of the remaining water is reduced. The consequences of an ITC voiding accident for AARR operation are discussed in detail in Section VI-B above.

Following sections of this Appendix present a detailed description of a mathematical model used in analog calculations to predict the transient behavior of the HFIR. Comparisons of calculated results with SPERT measurements are also provided. Survey calculations for HFIR, which were used in providing a design basis for the reactor safety system, are described.

C-II Method of Analysis

"Two methods of analysis have been used to examine the behavior of the HFIR when subjected to various reactivity additions. The method used most extensively was a mathematical analysis, using an analog model of the HFIR; the other method involved the extrapolation of SPERT experimental data to HFIR conditions. This latter analysis is discussed in greater detail in Section C-III of this Appendix.

"The reactor model selected for the analog studies was based on the usual set of heat removal and reactor kinetics equations. Wherever practical to do so, the constants and assumptions used were selected on a conservative basis. In the following paragraphs an effort is made to explain in a qualitative way the extent of the conservatism. A completely quantitative

analysis of the degree of conservatism has not been possible because of a lack of appropriate experimental data.

"The analog calculations were performed in such a way that the average HFIR core and very localized hot channel, hot streak, and hot spot portions of the average core were considered simultaneously. However, in so doing it was assumed that the transient behavior of the reactor was dependent only on average conditions, local net deviations from the average being neglected insofar as feedback is concerned. This means that no advantage was taken of negative reactivity feedback associated with strongly weighted moderator density changes in the high power density regions.

"Variations in reactivity resulting from temperature changes throughout the average core were accounted for by dividing the core into four radial regions: flux-trap target, water annulus between target and fuel, fuel, and reflector-control. Each of these regions had individual reactivity coefficients and heating and heat removal rates."

Some of the later calculations employed spatial representation of five axial divisions within the fuel zone of the core to provide additional detail in the representation of transient thermal processes. Reactivity variations resulting from possible void formation in the fuel region coolant were not included in early calculations because of the uncertainty associated with the worth and quantity of the voids. Later studies were made using a model for void formation based upon a correlation proposed by Griffith, Clark and Rohsenow.³ It was found that the degree of conservatism associated with ignoring void contribution to reactivity compensation is greater for low power operation and for short period reactivity accidents. Although the nominal maximum plate surface temperatures exceeded the saturation temperature before the peak power occurred and voids would occur in the fuel region, the calculated void fractions were so small for high coolant flow rates as to be essentially negligible in terms of negative reactivity feedback.

"Reactivity temperature coefficient data used in calculations were obtained in large part from HFIR critical experiments.⁴ In these experiments, the temperature coefficients for the island and for the fuel region were determined as a function of moderator temperature up to the normal full power moderator temperature for the reactor. Coefficients for the other less important regions were obtained from normalized calculations. Since it was not possible to obtain experimental coefficients for the end-of-core-life

conditions, calculations were also used to predict the ratio of beginning-of-cycle to end-of-cycle coefficients for the fuel region, the only region for which this ratio is significant. The end-of-cycle coefficient for the fuel region was calculated to be one-half that at the beginning of the cycle.

"The fuel region temperature coefficient was found to be somewhat sensitive to the moderator temperature. The coefficient becomes more negative as the moderator temperature increases. This effect was included in some of the calculations, while in others an average coefficient was used. From the standpoint of establishing the peak power and temperatures during a fast transient the variation in temperature coefficient with temperature was not really important; for all cases of interest the increase in average bulk moderator temperature up to peak power was less than 30 F.

"A fuel plate expansion coefficient of reactivity was also included in the analog calculation. Its value was determined from the fuel region void coefficients measured in the HFIR critical experiments.⁴ When the coolant flow rate and thus the water film heat transfer coefficient are low, the fuel plate expansion coefficient is equally as important as the moderator temperature coefficient in terminating an excursion."

In calculating the reactivity effect of fuel plate expansion it was assumed that only the expansion in plate width and thickness is effective in reducing core water content. Restraints imposed upon longitudinal expansion of the fuel plated would provide additional core water loss and additional reactivity reduction but this beneficial effect has been neglected in the analog analysis.

"Reactivity control by means of control plates was characterized by a power-level or rate-trip signal, a delay between signal and plate release, a linearly decreasing safety plate acceleration (to 1 g at the end of the first 6 in. of travel), and a control plate differential worth that was a function of plate position.

"The heat transfer mechanism simulated in the analog model was that associated with a flat-fuel-plate, rectangular-coolant-channel geometry. Heat transfer from the plates to the coolant was assumed to take place by conduction and convection and by direct deposition of gamma and neutron energy in the coolant.

"Resistance to the flow of heat from the plates to the water was provided by the aluminum fuel plate, an aluminum oxide film between the plate and water, and the fluid film: the latter two resistances were assumed to have negligible heat capacity. The inlet temperature of the coolant water was assumed to be constant during the transient because of the short duration of the transient relative to the primary coolant cycle time.

"Fuel region afterheat was included in the analysis and was assumed to have a maximum value of 7% of the steady-state power.

"The kinetic equations used in the analog studies to describe the transient behavior of the AARR core in a manner consistent with the above approximations and assumptions are as follows: the nomenclature is set forth in Table C-1.

"Nuclear equations:

$$\frac{dP}{dt} = \frac{(1 - \beta) k - 1}{\ell^*} P + \sum_{i=1}^{i=5} \lambda_i C_i \quad (C-1)$$

$$\frac{dC_i}{dt} = -\lambda_i C_i + \frac{\beta_i k P}{\ell^*} \quad (C-2)$$

$$k = k_o + F(t) + \sum_{j=1}^n \alpha_j \left[\bar{T}_j - \bar{T}_j(0) \right] + \alpha_{jF} \left[\bar{T}_F - \bar{T}_F(0) \right] \quad (C-3)$$

TABLE C-1

NOMENCLATURE FOR APPENDIX C

a	Ratio of power produced in fuel plates to total power
A	Heat transfer surface area
b	Ratio of power produced in coolant to total power
C_F	Total heat capacity of metal, Btu/°F
C_w	Total heat capacity of coolant, Btu/°F
C_i	Delayed neutron precursor for ith group (units and normalization consistent with use of power instead of neutron density in Eqs. C-1 and C-2)
$F(t)$	Control plate and accident reactivity variations with time, Δk
g	Acceleration due to gravity, in./sec ²
h	Heat transfer coefficient, Btu/hr ⁻¹ ft ⁻² (°F) ⁻¹
k	Effective neutron multiplication factor
k_0	Effective neutron multiplication factor at time zero
ℓ^*	Prompt neutron lifetime, sec
P	Total reactor power, Btu/sec
S	Distance control plate moves after release, in.
t	Time, sec
T_1	Oxide-metal interface temperature
T_2	Oxide-water interface temperature
$\bar{T}_F$	Average metal temperature over region
$\bar{T}_F(0)$	Average metal temperature during steady state
T_i	Average bulk water temperature at core inlet
$\bar{T}_j$	Average bulk water temperature throughout jth region
$\bar{T}_j(0)$	Average bulk water temperature throughout jth region during steady state
T_o	Average bulk water temperature at region outlet
$\bar{T}_w$	Average bulk water temperature throughout region
α_j, α_{jF}	Temperature coefficients of reactivity for jth region of core, for bulk water and metal, respectively, $\Delta k/^\circ\text{F}$
β_i	Delayed neutron fraction for ith group
λ_i	Decay rate of ith delayed neutron precursor
λ_1	Rate at which heat is transferred out of metal, Btu-sec ⁻¹ (°F) ⁻¹ (Determined from steady-state conditions)
λ_2	Ratio of heat transfer rate per unit temperature drop across oxide film to λ_1 (determined from steady-state conditions)
τ	One-half residence time of coolant in region

"Heat removal equations:

$$C_F \frac{d\bar{T}_F}{dt} + \lambda_1 (\bar{T}_F - T_1) = aP \quad (C-4)$$

$$\bar{T}_F - T_1 = \lambda_2 (T_1 - T_2) \quad (C-5)$$

$$C_w \frac{d\bar{T}_w}{dt} - \frac{C_w}{\tau} (T_i - \bar{T}_w) - hA(T_2 - \bar{T}_w) - bP = 0 \quad (C-6)$$

$$C_w \frac{dT_o}{dt} - \frac{C_w}{\tau} (\bar{T}_w - T_o) - hA(T_2 - \bar{T}_w) - bP = 0 \quad (C-7)$$

"Control plate movement equations:

$$\frac{d^2 S}{dt^2} = 4 \text{ g} - \frac{1}{2} \text{ g S in.}^{-1} \quad (0 \leq S \leq 6 \text{ in.}) \quad (C-8)$$

$$\frac{d^2 S}{dt^2} = \text{g} \quad (S > 6 \text{ in.}) \quad (C-9)$$

"In Eqs. (C-6) and (C-7) the quantity h is the heat transfer coefficient of the coolant film. During a transient, in which the flow rate remained constant, the h value was maintained constant for heat fluxes less than the steady-state burnout heat flux. When the latter heat flux was reached, the h value associated with that location was set equal to zero and was presumed to remain at zero until the power level was reduced to about one-half of the power level that corresponded to the steady-state burnout heat flux. In the analog analysis, incipient boiling heat fluxes were used in lieu of burnout heat fluxes because adequate correlations for the latter were not available. This approach is not considered unduly conservative because at high power a narrowing of the coolant channel tends to make the incipient boiling point and the burnout point nearly equal."

It should be noted that the use of the incipient boiling heat fluxes as the level corresponding to interruption of heat transfer from the fuel plate essentially removes from consideration the possibility of prior flow maldistribution. Significant loss of cooling due to flow instability prior to the onset of incipient boiling is not considered possible and hence fuel plate insulation coincident with incipient boiling is conservative in this respect.

"The absolute value of the average-core heat transfer coefficient prior to the film blanketing condition was taken as the minimum predicted by the applicable steady-state correlation.⁵ This was conservative since the use of a low h value resulted in less negative reactivity feedback associated with moderator expansion. At the hot spot a nominal h value, corresponding to steady-state hot spot conditions,⁶ was used.

"When the h value at the hot spot is set equal to zero, the ratio of hot spot temperature to surrounding plate temperature increases, and thus the radial heat transfer away from the hot spot (5/64 in. diam) should increase. In the analog analysis no increase in radial transfer was permitted after the h value was set equal to zero. Therefore, in this respect the calculated hot spot plate temperatures are higher than would actually be expected.

"The hot spot conditions used in the analog model were the same as those used in the steady-state analysis of the core⁶; that is, all conceivable tolerances and abnormalities (blisters, fuel segregation, etc.) within specifications were considered to exist in a consistent manner at the same time and place. Although much factual statistical data on the distribution of abnormalities in the fuel plates are still not available, it appears that the probability of the existence of the worst combination of heat removal conditions is very small. This needs to be kept in mind when using the calculated results to evaluate core damage and the ability of the HFIR to handle reactivity accidents.

"An accurate evaluation of the calculational results in terms of core damage becomes quite questionable when the predicted plate temperatures approach and exceed the melting temperature of the fuel plates. There are, however, some SPERT data which indicate that for calculated temperatures well in excess of the melting point no actual melting or even plate warpage will occur. For this reason, it appeared conservative to assume, for purposes of evaluating the calculated results, that no melting would occur until in the calculation the melting temperature was reached and enough additional heat was added (heat

of fusion) to actually melt the aluminum. In the calculation the heat of fusion was represented by an increase in plate temperature equal to the heat of fusion divided by the specific heat. Therefore, the calculated melting point temperature was equal to

$$\begin{aligned} 1950^{\circ}\text{F} &= 1210^{\circ}\text{F} \text{ (melting temperature)} + \frac{167 \text{ Btu lb}^{-1}}{0.226 \text{ Btu lb}^{-1}\text{ }^{\circ}\text{F}^{-1}} \\ &= 1210^{\circ}\text{F} + 740^{\circ}\text{F}. \end{aligned}$$

Calculated temperatures above 1210°F should be interpreted accordingly.

"From an operating point of view, the occurrence of a very small fission product leak is probably not the most serious type of core damage resulting from a transient. A permanent buckling of the fuel plates could significantly reduce the coolant velocity adjacent to a plate containing hot spots. Theoretically, this would reduce the burnout power level margin for normal full power operation. This might mean that the particular core, after experiencing a severe transient, would have to be operated at reduced power or discarded. Calculations indicate that for peak transient hot spot heat fluxes less than the steady-state burnout heat flux excessive permanent buckling of the fuel plates should not occur.

"Furthermore, the occurrence of accidents that would produce heat fluxes in excess of the above are considered to be so infrequent that the rejection of a core upon such an occasion would be of little concern.

"The safety system level and rate trips selected for the analog studies are shown in Table C-2.

Table C-2

Safety System Level and Rate Trips

Level trip (MW)	13-130
Rate trip (MW/sec)	20
Minimum change in power to actuate rate trip (MW)	5

"The level trip is proportional to the coolant flow rate, but the rate trip is fixed. The level and rate trips selected are as low as appears practical for flexibility in operation and for freedom from false scrams due to instrument noise.

"It was necessary in the analysis to consider the changing characteristics of the core as a function of time in the fuel cycle. At the beginning of the fuel cycle, when the control plates effectively separate most of the Be-H₂O reflector from the fuel region, the neutron lifetime is just one-half of the neutron lifetime at the end of the fuel cycle. However, the advantages of the longer neutron lifetime at the end of the fuel cycle are negated to some extent by the existence of a low thermal conductivity aluminum oxide film at the hot spot, greater fuel plate deflection, and, generally speaking, a lower safety plate differential reactivity worth. In many cases the end-of-cycle conditions result in higher hot spot plate temperatures than beginning-of-cycle conditions. Therefore, both extreme conditions were considered in the study.

"Additional pertinent characteristics associated with the analog model are listed in Table C-3."

Representation of delayed (γ, n) reactions in the beryllium reflector of AARR and their effects upon AARR transient behavior have not been included in the mathematical model described herein. Although these effects may be duly accounted for through the addition of appropriate additional delay group relationships in the neutron kinetics equations as developed by Keepin ⁷, the resulting added complexity was not considered warranted. In these studies, the major concern is the behavior of the reactor under super-critical conditions for which the contribution of delayed photoneutrons to the overall neutron balance is very small.

However, it is recognized that when the operating reactor is made subcritical, photoneutron production will influence the decay of fission power. This effect may be of significance, especially for small reactivity shut down margins:⁸ when required, proper consideration is given thereto as, for example, in the analysis of emergency cooling requirements for a loss of primary coolant flow accident.

C-III. Significance of SPERT Data

The information contained in the following paragraphs, which constitutes an evaluation of the ability of the analog model described above to predict SPERT test results, is a verbatim reproduction of material presented in Appendix C of Reference (2).

TABLE C-3

FULL POWER, FULL FLOW, STEADY STATE CHARACTERISTICS OF THE ANALOG MODEL

	Target Region	Target Annulus Region	Fuel Region						Control Region
			Start of Cycle			End of Cycle			
			Nominal	Hot Streak	Hot Spot	Nominal	Hot Streak	Hot Spot	
Metal heat capacity, C_F , Btu/°F	1.49	0	35.7	35.7	41.4	35.7	35.7	41.4	13.1
Metal heat generation, aP_i , Btu/sec	853	0	92,400	132,500	242,300	92,400	132,500	242,300	627
Mean metal temperature, $\bar{T}_f$, °F	365	125	230	349	428	238	469	627	152
Metal-oxide interface temperature, T_1 , °F	199	125	222	337	407	230	457	606	150
Oxide-water interface temperature, T_2 , °F	199	125	222	337	407	222	344	417	150
Water inlet temperature, T_i , °F	120	120	120	121	121	120	121	121	120
Water outlet temperature, T_o , °F	136	130	189	257	257	189	255	255	128
Water heat generation, bP_i , Btu/sec	144	264	2721	2775	2775	2721	2775	2775	283
Water transit time, 2τ , sec	0.0417	0.2514	0.0397	0.0403	0.0403	0.0397	0.0418	0.0418	0.0901
$\lambda_1 = aP_i/(\bar{T}_f - T_1)$	5.203		11,555	11,326	11,377	11,555	11,326	11,377	241
$\lambda_2 = (\bar{T}_f - T_1)/T_1 - T_2)$	∞	-1	∞	∞	∞	1.333	0.1034	0.1131	∞
Temperature coefficient, $\alpha_j, (\Delta k/^\circ\text{F}) (\times 10^5)$	+3.2	+2.7	-9.7 ^(a) -1.0 ^(b)			-4.8 ^(a) -0.5 ^(b)			+1.1

TABLE C-3 (continued)

Reflector flow rate, gpm	840	Fuel region water film heat transfer coefficient, h, Btu hr ⁻¹ ft ⁻² (°F) ⁻¹	
Reflector power level, MW	3		
Primary system flow rate, gpm	15,000	Average core	12,000
Prompt neutron lifetime, λ^* , μ sec		Hot spot	15,000
Beginning of cycle	35		
End of cycle	70		
Effective delayed neutron fraction, β_{eff}			
Beginning of cycle	0.0070		
End of cycle	0.0071		
Steady state burnout heat flux ^(c) , Btu hr ⁻¹ ft ⁻² (°F) ⁻¹	2.47 x 10 ⁶		

Low Power, Full Flow

Temperatures and heat generation rates reduced to be consistent with initial steady state power.

Temperature coefficients different, depending on initial temperatures.

If initial bulk water temperature less than 120°F, burnout heat flux^(c) will be $> 2.47 \times 10^6$; assumed = 2.47×10^6 .

Heat transfer coefficients assumed the same.

Low Power, 10% Flow

Temperatures and heat generations appropriate

Temperature coefficients appropriate

Burnout heat flux^(c) = 4.70×10^5 .

Fuel region water film heat transfer coefficient, h.

Btu hr⁻¹ ft⁻² (°F)⁻¹

Average core	1970
Hot spot	2460

Control System Characteristics

Initial acceleration was 4 g for all calculations discussed in this report.

Safety system delay time: 10 msec

Scram signal: Rate and level

(a) Water

(b) Fuel plates.

(c) Burnout heat fluxes tabulated are actually calculated steady state incipient boiling heat fluxes. Actual burnout heat fluxes are somewhat greater.

"In attempting to make use of SPERT data for partially analyzing the transient behavior of the HFIR, two approaches were taken. The first was to use the HFIR analog model, with appropriate SPERT input data, to calculate specific SPERT cases. The purpose in doing this was to determine whether or not the HFIR model was conservative. Since the applicable SPERT tests had relatively low coolant flow rates, were unpressurized, had only self-shutdown mechanisms, and started at essentially zero power level, a direct comparison with the HFIR at power could not be made.

"The results of the comparison with the available SPERT data indicated that the HFIR model was quite conservative for the low power conditions. Furthermore, there were indications that the absence of voids in the analog model was responsible for a significant portion of the conservatism. Judging from a comparison of calculated water film temperatures for the low and high power cases, it was concluded that voids would also be present in similar transients at high power; thus it was further concluded that the model should be conservative for the corresponding initial-high-power cases. Since the degree of conservatism is not known, one cannot really take advantage of it in a quantitative sense. However, an indication of conservatism provides additional confidence in an analytical method that would have been used unhesitantly in the absence of the experimental data.

"The SPERT core and test calculated was that identified as SPERT IV DU-12/25;⁹ physical characteristics of this core and the HFIR core are compared in Table C-4.

"The particular SPERT experiments were conducted at atmospheric pressure, using effective step changes in reactivity of approximately 0.00973 and 0.0126 $\Delta k/k$. Calculations were made with these same step changes, but no void coefficients were included. In order to compare the "experimental" results with the calculated results, the SPERT data were extrapolated to a pressure of 2500 psi to obviate the necessity for including voids in the calculation; the result was a 30% increase in peak power, a 100% increase in energy release, and about a 60% increase in plate temperatures.¹⁰ These data are compared with the calculations in Table C-5. Also included in Table C-5 are the low pressure experimental data.

TABLE C-4

CHARACTERISTICS OF SPERT (DU-12/25) AND HFIR CORES

	SPERT, DU-12/25	HFIR
Core geometry	Square parallelepiped	Cylindrical flux trap
Type of fuel element	Aluminum clad plates, U-Al alloy meat	Aluminum clad plates, U ₃ O ₈ -Al dispersion meat
Fuel loading (kg)	3.8	9.4
Core dimensions (cm)		
Outside diameter		44.0
Inside diameter		14.0
Length of side	38.0	
Height (active)	61.0	50.8
Effective core volume (liters)	62.2	50.8
Fuel plate surface area (cm ²)	1.483×10^3	2.774×10^3
Surface-area-to-volume ratio (cm ⁻¹)	0.017	0.055
Fuel plate thickness (cm)	0.152	0.127
Coolant channel thickness (cm)	0.455	0.127
Metal-to-water ratio (based on fuel plates only)	0.335	1.0
Overall metal-to-water ratio	0.66	1.0
Fuel plate heat capacity (Btu/°F)	23.2	35.7
Prompt neutron lifetime (μsec)	57.0	35.0 ^(a)
Moderator temperature coefficient in fuel region (Δk/°F)	-8.33×10^{-5}	-5.8×10^{-5} ^(b) -8.7×10^{-5} ^(c)
Fuel plate temperature coefficient (Δk/°F)	-0.33×10^{-5}	-1.0×10^{-5} ^(b) -1.6×10^{-5} ^(c)
Void coefficient, fuel region average (Δk/%ρ)	0.0024	0.0029
Effective delayed neutron fraction	0.007	0.0071
Power density ratio, nominal maximum-to-average	2.4	1.45

(a) Clean core condition

(b) Clean core, 68°F moderator temperature

(c) Clean core, 155°F moderator temperature

TABLE C-5

COMPARISON OF EXPERIMENTAL AND CALCULATED SPERT (DU-12/25) RESULTS

Reactivity Step Insertion ($\Delta k/k$)	Experimental Stable Period (msec)	Peak Power (MW)			Total Energy Release During Transient (MW-sec)			Maximum Plate Temperatures ^(a) (°F)					
								At Peak Power			Maximum		
		Experimental		Calculated	Experimental		Calculated	Experimental		Calculated	Experimental		Calculated
		10 psi	2500 psi		10 psi	2500 psi		10 psi	2500 psi		10 psi	2500 psi	
0.00973	20.7	169	220	440	9.6	19.1	38	300	435	1340	307	450	1960
0.0126	10.1	505	657	1625	12.2	24.4	66	364	540	3000	378	560	4440

(a) Does not include hot spot conditions. Calculated values do not consider heat of fusion; temperatures above 1210°F should be corrected for this fact.

"As indicated by the comparison of results in Table C-5, the calculation is conservative by a considerable amount. In the actual experiments, there was no evidence of core damage. The calculations, however, indicate melting for both cases, even though hot spot factors were not included.

"Part of this conservatism probably results from the use of a water film heat transfer coefficient (h) that is too small and/or a percentage heat generation rate in the coolant (b) that is too small. Increasing the values of these two parameters would reduce the plate temperatures, peak power, and total energy release because of less heat generation in the plate, a smaller resistance to heat transfer out of the plate, and because there would be greater negative feedback.

"Evidence of low h and b values is obtained by comparing plate temperatures at the peak power and at the peak temperature. The experimental data show a small percentage difference between these two temperatures, while the calculated percentage difference is relatively large.

"The h and b values used in all of the HFIR calculations have purposely been low so as to result in conservative plate temperatures. However, in previous HFIR analog studies¹¹ and also during the SPERT-HFIR studies, the effect of varying h was investigated. The results indicated that a 40% increase in h reduces the peak power by only 3% and the maximum plate surface temperature by only 14%. Furthermore, increasing b from 2.5% to a reasonable upper limit of about 6% does not make a very significant difference in the lack of agreement between experiment and calculation. Thus, at least for the high pressure cases, it appears that there is an additional feedback mechanism that is not being accounted for in the calculation. It is interesting to note, however, that for the high pressure comparison the "experimental" temperatures were not above the saturation temperature. Therefore, if one discounts the possibility of significant void formation in hot spot regions that were not instrumented, and if the method used for extrapolating from low pressure to 2500 psi was reasonably correct, he must conclude that at least at high pressure there was a shutdown mechanism not associated with void formation in the usual sense. Whether or not the same or another unaccounted for feedback mechanism exists at higher power (~100 Mw) can only be speculated upon. If the mechanism is associated with voids, it probably exists at higher power levels also.

"The other approach used in trying to make use of SPERT data involved a direct analytical comparison of the SPERT and HFIR transients resulting from the same reactivity additions. A comparison of the physical characteristics of the two cores indicated that the HFIR, with no safety system other than the self-shutdown mechanisms, should experience a less severe transient than the particular SPERT core of interest, when both cores are subjected to the same reactivity addition. The reason for this is that the HFIR, while having about the same void and temperature coefficients, has almost 2.5 times the surface-area-to-volume ratio and just a little over one-half the water-to-core volume ratio as the SPERT core. This means that for a given increase in fuel plate temperature the increase in moderator temperature and thus the negative reactivity feedback are considerably greater for the HFIR. This assumes of course that the greater flow resistance associated with the narrower HFIR coolant channels will not impede the density change to the extent that the potential improvement will be negated.

"Another feedback mechanism which favors the HFIR is the fuel plate expansion reactivity coefficient. This coefficient for the HFIR is about three times that for the particular SPERT core. Since the plate coefficient is really a moderator expulsion coefficient, the above comment relating to the increased flow resistance of the narrower HFIR channel applies here, too.

"The particular SPERT core being compared with the HFIR has one clear advantage: it has a longer prompt neutron lifetime when compared to the HFIR clean core condition (57 compared to 35 μ sec). For the end-of-cycle condition, the neutron lifetime for the HFIR is about 70 μ sec.

"Results from the SPERT-HFIR comparison calculations are shown in Table C-6. When evaluating the comparison, several things need to be kept in mind. The calculations did not include any hot spot factors but did include the nominal maximum-to-average power density ratios (2.4 for SPERT and 1.45 for the HFIR). (Plate temperatures for these "maximum" power density spots and for the average core are tabulated.) Neglecting the possible effect of power distribution on reactivity feedback, a comparison of the average plate temperatures is probably more indicative of the differences to be expected from the different SPERT

TABLE C-6

COMPARISON OF SPERT AND HFIR CALCULATED TRANSIENT CHARACTERISTICS

Reactivity Step Insertion ($\Delta k/k$)	Coolant Velocity (ft/sec)	h (Btu hr ⁻¹ ft ⁻² (°F) ⁻¹)	Peak Power (MW)	Peak Average Power Density (MW/liter)	Energy Release Per Unit Core Volume at Time of Peak Power (MW sec/liter)	Total Energy Release Per Unit Core Volume (MW sec/liter)	Peak Maximum ^(a) Plate Temperature ^(b) (°F)	Peak Average Plate Temp. ^(b) (°F)	Peak Maximum Heat Flux ^(c) (Btu hr ⁻¹ ft ⁻²)
SPERT									
0.00973	12	3.5×10^3	440	7.1	0.29	0.61	1960	890	6.4×10^6
0.0126	12	3.5×10^3	1625	26.1	0.56	1.06	4440	1980	15.0×10^6
0.0126	12	4.93×10^3	1575	25.3	0.53	0.98	3800	1700	17.7×10^6
HFIR									
0.00973	5	2.1×10^3	420	8.3	0.20	0.39	950	450	1.6×10^6
0.00973	12	4.4×10^3	410	8.1	0.20	0.35	760	380	2.6×10^6
0.00973	20	6.4×10^3	410	8.1	0.20	0.37	660	340	3.3×10^6
0.00973	42	10.5×10^3	420	8.3	0.20	0.39	560	310	4.5×10^6
0.00973	42	2.1×10^3	480	9.4	0.22	0.47	1100	510	2.0×10^6
0.0126	5	2.1×10^3	1600	31.5	0.39	0.73	2000	890	3.8×10^6
0.0126	12	4.4×10^3	1550	30.5	0.35	0.67	1650	760	6.2×10^6
0.0126	20	6.4×10^3	1540	30.3	0.33	0.65	1450	690	8.0×10^6
0.0126	42	10.5×10^3	1510	29.7	0.33	0.63	1225	610	10.8×10^6

(a) Includes only nominal maximum-to-average power density ratio; does not include hot spot effects.

(b) Calculated values do not consider heat of fusion, and temperatures above 1210°F should be corrected for this fact.

(c) Assumes constant h value.

and HFIR core characteristics.

"In the above calculations all reactivity feedback was based on average core conditions. The inclusion of feedback from properly weighted "hot" areas of the cores, where the reactivity importance might be greater, should decrease the extent of the transient. However, because the SPERT core has a greater maximum-to-average nominal power density ratio than that for the HFIR, it is possible that inclusion of such a factor would show that on a relative basis SPERT transient temperatures and power densities would not be so much higher than those for the HFIR.

"Two other simplifications used in these HFIR-SPERT calculations involved the use of a constant h value, and the heat of fusion of the fuel plates was not accounted for. Had h been reduced to essentially zero when the burnout heat flux was reached, the corresponding calculated temperatures would have been relatively higher for SPERT. To account for the heat of fusion, all plate temperatures between 1210°F and 1950°F should be 1210°F , and all temperatures above 1950°F should be reduced by 740°F .

"The application of the above "normalizing" conditions and second order corrections does not alter the conclusion that plate temperatures and heat fluxes for the HFIR would actually be significantly lower than in the particular SPERT core, when subjected to the same reactivity addition. Since the SPERT core suffered no apparent damage when subjected to a $0.0126 \Delta k/k$ step change, it is further concluded that the HFIR would not be damaged by the same reactivity addition, which is about as large as would be considered "reasonable." This latter conclusion is, of course, applicable to the initial-low-power, low-flow-rate condition only. However, there is no apparent reason why the conclusion would not also be applicable for higher powers and flow rates. Even so, since there is insufficient high power SPERT data to compare with, one can at best conclude that the method for analyzing the high power cases is conservative.

"In view of the physical differences in the SPERT and HFIR cores, a further evaluation of the calculated results is of interest. It is observed in Table C-6 that the peak average power density calculated for the HFIR was about 1.7 times that calculated for SPERT, indicating that the negative feedback per unit power density was less for the HFIR in spite of its lower water volume and greater heat transfer surface area per

unit core volume. The reasons for this are that the HFIR has about 2.0 times as much aluminum (fuel plate) per unit core volume to store heat in during a transient, and the HFIR power rise is quicker because of the shorter neutron lifetime. The effect of the latter can be factored out to some extent by comparing total energy release per unit volume at the time the transient is turned around. As indicated in Table C-6, the total energy release for the HFIR is about 30% less than for SPERT. Of course the peak power density and the total energy density, per se, are of little consequence, since plate temperatures are more indicative of core damage."

C-IV Survey Calculations of Core Transient Behavior

A. Kinetic Behavior of the Reactor

In order to establish the design basis for the reactor safety systems, the kinetic behavior of the reactor was investigated by analog simulation.

"It was found that when at high power, the HFIR is strongly stabilized by thermal feedback in reactivity. This is due in part to the large net negative temperature coefficient of the system and in part to the extremely short time constants associated with the transfer of heat between fuel and coolant. The average fuel-to-coolant time constant at full flow is ~ 25 msec, which is somewhat shorter than that of many pressurized-water reactors. This tight thermal coupling between the fuel and moderator together with the fuel plate prompt negative temperature coefficient provide the prompt negative reactivity feedback that is effective in quenching power excursions: in particular, it is effective in mitigating the consequences of a startup accident. In this connection it should be noted that, although the temperature and void coefficients of the coolant in the flux trap are positive, this is a region of low-power density relative to the fuel; thus, the net overall effect of increased temperature on reactivity is a stabilizing one at any power level."

Some of the results described below demonstrate the adequacy of the safety system under various abnormal conditions. The analog simulation includes most of the known mechanisms which provide internal reactivity feedback and is believed to be conservative, since it is virtually certain that additional negative reactivity factors are operative.

Core performance data presented herein are based upon early calculations performed for the HFIR, using as a basis reactivity induced

accidents associated with the concept of ITC voiding appropriate to the HFIR. As mentioned previously this approach becomes very conservative when applied to the AARR.

In order to judge the adequacy of the safety system for the HFIR, it was necessary to establish certain performance criteria. These criteria are based not only upon certain physical boundary conditions, but also to some extent on the experience and judgment of the designers, particularly in regard to instrument response and noise. The significant criteria used to judge the effectiveness of the safety system for fast excursions are as follows:

"Nondamage Criterion - The hot-spot heat flux does not exceed the steady-state value which corresponds to the minimum burnout or incipient boiling heat flux calculated for the conditions present at the hot spot. Should this condition be reached, it is probable that some distortion of the fuel plates may occur; but no release of fission products to the primary coolant is expected to result.

"Damage Criterion - The fuel cladding at the hot spot reaches the melting temperature and contains just sufficient heat to actually initiate melting. In simulating this situation, it has been assumed that when the transient hot-spot heat flux exceeds the steady-state burnout or incipient boiling heat flux, the heat transfer at that spot vanishes and remains zero until the excursion is terminated. Under these conditions, it is further assumed that the damage will be limited to melting in small areas accompanied by a minor release of fission products to the primary coolant but that no mechanical damage other than warping and distortion of the fuel plates will occur. This criterion is consistent with the observed results in the SPERT test series for cases where only a small amount of melting took place.

B. Safety System Speed of Response

"In the past, reactor safety systems have often been designed primarily for protection against the startup accident (continuous motor-driven rod withdrawal from the shutdown condition). Analysis indicates that a fast-acting external safety system for this purpose is not required in the HFIR (see Section VI-B above). On the other hand, a more conceivable accident would involve the introduction of a significant reactivity increase while the reactor is at power. Perhaps the most obvious way in which this could be brought about is by the introduction of a void into the coolant in the target region. Whereas the geometry of the

target region and its coolant is such that it is highly improbable for such a void to be swept in, the amount of reactivity and its possible rate of addition appear to represent the maximum conceivable reactivity accident to the reactor. For this reason, the "optimum void" incident was selected as a basis for investigating the performance of the safety system.

"Investigations made early in the design phase indicated that the temperature coefficients alone were sufficient to handle the initial peak of all excursions resulting from a stepwise insertion at full power of up to $0.0075 \Delta k/k$. It was found that to handle significantly greater amounts of reactivity, it would be necessary for the safety system to have a short release time and an initial acceleration in excess of $1 \times g$. As a result of these studies, it was decided that the goal of the safety system design effort should be the development of the fastest safety system considered to be practical with known techniques."

The final design and an account of the considerations leading thereto are described in Reference (1). Level protection at constant 100% flow is provided by a trip at 1.3 times maximum allowable steady-state power. To gain an additional margin of safety, and to provide protection against the insertion of large increments of reactivity a rate trip is provided that initiates a scram for power increase rates equal to or greater than 20 MW/sec.

"Based upon experience with the release mechanism originally developed for the Oak Ridge Research Reactor it appeared likely that a safety plate release time of 0.01 sec or less and an initial acceleration of $4 \times g$ or greater could be achieved and accordingly these performance goals were established.

"Later simulator investigations utilized better information on reactor characteristics, and were performed in order to test the ability of the safety system to cope with incidents occurring at full power. In these studies various amounts of reactivity were introduced on a 30-msec ramp. (The approximate coolant transient time in the target region is 40 msec; however, the time was reduced to better approximate the reactivity buildup.) The damage and nondamage criteria were used to assess the results. Investigations were also made to determine sensitivity to the safety-plate-release delay time, to the initial plate acceleration, and to the position of the safety plates at the initiation of a scram. The case in

which one plate fails to scram was also considered. As previously stated, since the rate trip is most effective in a fast excursion, only this was used, and it was set to trip at 20 Mw/sec. The rate trip is activated at about 1.05 times normal full power if the excursion starts at full power and is very fast.

"At the beginning of an operating cycle, the safety plates have a group reactivity worth of about $0.007 \Delta k/k$ per inch. As can be seen from Fig. C-1, a 10-msec release time and a $4 \times g$ initial acceleration satisfies the nondamage criterion for the specified reactivity increase rate up to $0.0071 \Delta k/k$ inserted; the damage criterion is met for reactivity increases up to $0.0142 \Delta k/k$. It was determined that the temperature coefficients alone can handle the excursion peaks from reactivity insertions of 0.0058 and $0.0115 \Delta k/k$ within the nondamage and damage criteria respectively.

"As the operating cycle proceeds and the control plates are withdrawn, their worth per unit travel will decrease. The situation corresponding to about one and one-half days before the end of cycle is shown in Fig. C-2. When these curves are compared with those of Fig. C-1, it should be realized that the characteristics of the core have changed during the cycle. The prompt-neutron lifetime will have increased from about $35 \mu\text{sec}$ to about $70 \mu\text{sec}$, and an oxide film will have built up on the fuel plates, particularly at the hot spot. For design conditions in this case, the nondamage and damage criteria yield allowable rapid reactivity increases of 0.009 and $0.0136 \Delta k/k$ respectively.

"The worst condition studied, minimum safety plate differential worth, occurs at the end of the cycle when the plates are fully withdrawn (Fig. C-3). It was assumed that the withdraw limits were set so that the initial worth of the four shim-safety plates would be no less than $0.0007 \Delta k/k$ per inch. Moreover, in this case it has been assumed that one of the plates fails to scram. The nondamage and damage criteria yielded acceptable reactivity changes of 0.0072 and $0.0017 \Delta k/k$ respectively.

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FIG. C-1
PERMISSIBLE $\Delta k/k$ VS SAFETY SYSTEM DELAY
BEGINNING OF CYCLE

FIG. C-2

PERMISSIBLE $\Delta k/k$ VS SAFETY SYSTEM DELAY
ONE AND ONE-HALF DAYS BEFORE END OF CYCLE

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FIG. C-3
PERMISSIBLE $\Delta k/k$ VS SAFETY SYSTEM DELAY
END OF CYCLE

"Another series of investigations was made to determine what requirements, if any, should be placed on the safety system in order to cope with accidents that might occur at low reactor power. The accidents studied included the startup accident and various excursions resulting from step increases in reactivity at low power. The initial power level used in these studies was 0.05 w, which is the estimated minimum power at source level. The simulation was performed both at 100% and at 10% coolant flow with the level safety system only, with the rate safety system only, and with all safety action delayed (i.e., with no fast-acting safety system).

"At 100% coolant flow it was found that the reactor could satisfy the nondamage criterion during a continuous rod withdrawal of $0.005 \Delta k/k$ per second or a step increase of $0.0098 \Delta k/k$ even with delayed safety action. The close coupling between the fuel and the coolant and the net negative temperature coefficient provide the initial self-shutdown mechanism.

"At 10% coolant flow the reduced heat transfer coefficient at the fuel plate surfaces makes the self-shutdown mechanism somewhat less effective. Moreover, the permissible hot-spot heat flux is considerably lowered by the reduced flow. With delayed safety action and 10% flow, a reactivity step of $0.0076 \Delta k/k$ can be sustained without exceeding the nondamage criterion. If the safety system is operating this is increased to $0.009 \Delta k/k$."

C. Influence of Initial Reactor Power Upon Transient Behavior

For cases of very rapid reactivity addition rates and assuming no safety system action it is to be expected that the consequences of a given reactivity addition will be essentially independent of initial reactor power level until the initial value approaches the range necessary for significant inherent reactivity feedback. Thereafter, the consequences increase in severity with increasing values of initial power level. This is inferred from the neutron kinetics equations for step function reactivity insertions in excess of the effective delayed neutron fraction. In the absence of feedback, the instantaneous reactor period decreases following application of a reactivity step and approaches an asymptotic value which depends only upon the reactivity added and the assumed constant neutronic parameters. The initial departure of the reactor period from the asymptotic value is due to the influence of delayed neutrons which, during the short time span involved for the completion of super-prompt critical excursions, act as an essentially constant neutron source of strength proportional to the initial neutron level. The effect of such a source upon

the reactor period is inversely proportional to the instantaneous neutron level. Hence, the period departs from the asymptotic value by less than one per cent whenever the neutron level is greater than one hundred times its initial value. Thus, since the AARR core develops little reactivity feedback until the power level approaches 10 MW (assuming full coolant flow rates), the reactor behavior in the power range above this level will be the same for all such transients beginning at an initial power level below 0.1 MW. As the initial power level is increased above 0.1 MW, the reactor period that exists when the transient power enters the feedback range shortens. This causes an increase in the time lag between reactor power and reactivity feedback causing a greater power overshoot and energy release. Thus, in this range of initial power levels, the consequences of a given reactivity insertion increase in severity with increasing initial power level.

The assumption of a step function reactivity insertion used in the foregoing discussion simplifies the analysis inasmuch as it assures that the reactor is subjected to the same net total reactivity insertion in all cases. Clearly, this situation changes for ramp function reactivity insertions because of thermal feedback effects, such that for a given ramp rate, the net maximum reactivity imposed upon the reactor will be a function of initial power level.

However, for reactivity ramp rates sufficiently high compared to reactor response rates, the reactor behavior will approximate a step function response. For lesser ramp rates, there will be a worst condition of initial power corresponding to each insertion rate. Additional calculations are in progress to assess this effect for ramp function insertions occurring in a time span of 30 msec.

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APPENDIX DASPECTS OF AARR RESEARCH AND DEVELOPMENT PROGRAMIMPORTANT TO SAFETY

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APPENDIX DASPECTS OF AARR RESEARCH AND DEVELOPMENT PROGRAMIMPORTANT TO SAFETYD-I Introduction

Since the majority of the AARR core design parameters and reactor components such as the fuel elements, control plates, control drives and inner reflector are identical to the corresponding parameters and components in HFIR, only a small amount of research and development will be required for the core. The major R&D effort will be for the development of the experiment facilities. This Appendix describes that portion of the AARR R&D program which concerns the safety of the reactor.

D-II Transient Heat Transfer Tests

The transient heat transfer test program is intended to provide a means by which the calculated effects of positive reactivity additions on heat transfer and hydrodynamics in the core may be compared with a physical model that approximates the behavior of a fuel plate and the cooling media during a simulated reactivity-induced power transient. An equally important aspect of the tests is to provide the basic information required to improve the methods of analyzing reactor power transients.

The physical model used is an electrically heated test section conforming closely to the geometrical shape of the core flow channel in which representative flow conditions for the AARR core are used. Initial tests will study the design conditions of concern in the safety analysis dynamic studies. However, the continuing program of dynamic analysis for the final safety analysis and these tests will have mutual feedback which will yield new information requiring incorporation into the testing program. The final tests will incorporate this new information and provide substantiation of the operating limits of the reactor with regard to inadvertent reactivity additions.

The basic plan for the test program is to conduct the tests on an approximate AARR channel in which the power is varied with time in accordance with the predictions of the dynamic analyses. It is planned to use a specific test section for 100 MW testing in which the approach to test section failure by heat transfer or hydrodynamic phenomena will be made in small steps in order to adequately study the channel behavior. Beginning at the equivalent of 100 MW initial power, the tests will proceed from conditions representative of low reactivity insertion cases to the maximum test facility capability which is estimated to be about 75¢ in reactivity added linearly in 33 msec. Depending on the results and operating experience, the tests will then proceed at higher initial power levels and the same procedure repeated in order to determine, if possible with the equipment available, the actual burnout point of the test section. Void measurements are also planned.

The conditions at which the threshold of fuel melting is reached and the character of pressure pulses during non-melting transients will be the main areas of interest. This heat transfer and hydrodynamic information will be compared with the safety analysis model and used to make improvements in the model.

D-III System Dynamics (Reactor Kinetics) Studies

A. Control Systems Dynamic Model

An analog computer program will be developed to simulate the reactor core, primary and secondary process systems and associated mechanisms for the control of power level, system pressure and coolant flow rate and temperature. Analog simulation will be used to develop control design information for the architect-engineer and for verification of control systems stability and performance capabilities.

B. Transient Dynamic Model Development

Analytical investigations are in progress, designed to culminate in an overall mathematical model which describes the transient behavior of the AARR and an appropriate digital computer program to implement necessary calculations. Specific problem areas under investigation are the following:

(1) Transient heat transfer and hydraulics models will simulate those conditions in the AARR fuel plates and coolant channels, important to the determination of inherent reactivity feedbacks and transient heat transfer limitations. Development of these mathematical models will be guided by the results of the transient heat transfer tests planned for the AARR.

(2) Core and primary system damage models are intended to provide a means for calculating reactor damage during power transients due to local boiling pressure pulses and pressure pulses arising from rapid melting of fuel. Studies will include the following:

(a) The possibility of core damage resulting from pressure pulses induced by coolant boiling will be studied and incorporated in a core damage model if significant.

(b) The effects of large pressure pulses, which originate in the reactor core due to rapid melting of fuel, upon the boundaries of the primary system will continue to be analyzed. In particular, emphasis will be placed upon the calculation of threshold values of transient pressures leading to primary system rupture (MCA conditions).

(3) The primary and secondary process systems will be modeled and coupled with the foregoing core models such that interactions between process and core variables may be studied. Safety systems' performance in response to abnormal reactor conditions will be modeled and analyzed.

C. FSAR Transient Model and Calculations

Whenever applicable the above mathematical models will be combined in an overall computer code which will be used for transient calculations to simulate hypothesized reactor accidents. Results of these calculations will be presented in the FSAR for the AARR.

D-IV Component Development and Testing

A. Reactor Vessel Design Study

A complete analytical review of the AARR reactor pressure vessel will be performed. It may include both an extensive stress analysis of the critical areas and a thermal analysis of the vessel to obtain the temperature distribution through the wall. Because of the dependence on geometry, the review will be performed on the selected design configuration approved for fabrication.

The midsection of the pressure vessel with its numerous beam tube nozzle penetrations and the flat head closure at the top make these regions very crucial from a structural standpoint. Although stresses in the vicinity of the nozzle penetrations can be calculated by approximate methods, none of them are capable of considering in sufficient detail the complex geometry of the junction resulting from the multiplicity of penetrations. The proximity of penetrations results in significant overlapping of stresses between adjacent nozzles. In addition, a fillet, or even a weld bead, can significantly affect the stiffness of the junction and result in discrepancies between the actual and calculated stresses. Because the magnitude of the departure of the calculated stresses from the actual are unknown, and may be great unless the configuration is accurately defined, it is imperative that the review be performed on the approved vendor's design. The analytical review will serve as a verification of the results obtained by the designer's and fabricator's final vessel design.

B. Gas and Hydraulic Rabbit Facility Development

The success of the AARR to the Laboratory is dependent upon the ability to utilize the neutron flux within the core safely and conveniently. Many prime experimental programs require that samples irradiated in high neutron flux zones be made immediately available to the experimenter. Since other programs require steady state operation of the reactor, devices which provide for insertion and removal of samples from the reactor complex during operation are therefore essential.

Suitable gas and hydraulic rabbit devices must be developed and tested to assure (1) that they can be fitted to the reactor, and (2) that they satisfy the stringent safety parameters of heat transfer and reactivity effects encountered in the penetration of the relatively small core and beryllium reflector regions of high neutron and gamma flux.

In order to assure satisfactory and safe operation, complete full-scale prototypes of the various rabbit facilities will be fabricated. The prototypes will include the hydraulic and gas rabbits for the ITC and the reflector positions. The test program for the prototypes will include:

- (1) Remote installation and removal.
- (2) Simulation of hot cell assembly and disassembly.
- (3) Tests of coolant flow and reversal.
- (4) Measurement of time of travel and velocity.
- (5) Positioning and position indication studies.
- (6) Heat transfer investigations.
- (7) Other tests as may be required to satisfy the design parameters.

C. Blind Beam and Through Tube Development

The complexity, safe operation, and cost of the horizontal facilities require development and refinement of the current conceptual design prior to final design and fabrication of the actual beam tubes. To implement necessary design changes, improvements, or verification of design, one blind beam tube and one through tube prototype will be fabricated and tested. Remote assembly and disassembly of the facilities, destructive tests, and nondestructive tests will be performed on the prototypes to assure proper and safe design. The prototype beam tube assembly test program will include the following:

- (1) Mechanical design tests to check dimensions, confirm tolerance, and evaluate the overall design.
- (2) Proof tests to demonstrate that assembly and disassembly can be adequately accomplished.

In addition to tests to be performed on the complete prototype assemblies, a group of integrity tests is planned for the purpose of further demonstrating the adequacy of the design:

- (1) Destructive Tests
 - (a) Collapse tests will verify the calculated collapse pressure and the status of seals following a beam tube failure.
 - (b) Bending tests to verify tube characteristics and seal capability under this type of loading.
- (2) Non-destructive Tests
 - (a) Hydrostatic tests to verify performance of seals and diaphragms under simulated service conditions.
 - (b) Coolant flow tests to determine flow characteristics of beam tube and vessel nozzle coolant channels.

D. Static Irradiation Facility Development

A prototype of a typical static irradiation facility will be fabricated and tested in order to facilitate required further development, refinement or verification of current conceptual designs prior to final design and fabrication of the actual facility.

To ensure that a safe and usable design will emerge, the test program of the prototype will include:

- (1) Simulated remote installation and removal tests.
- (2) Assembly and disassembly tests both under hot cell and reactor conditions.
- (3) Heat transfer tests.

In addition, tools and fixtures for performing the required tasks will be developed as part of the test program.

APPENDIX EREACTOR VESSEL CONTAINMENT POTENTIAL

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APPENDIX EREACTOR VESSEL CONTAINMENT POTENTIAL*E-I. Vessel Dimensions and Material Properties

The basic dimensions used in the preliminary analysis of the reactor vessel containment potential for an Inconel-600 vessel are listed below:

Inner radius	48 in.
Main shell thickness	3 1/2 in.
Height of main shell (approx)	15 ft
Volume	754 ft ³

The mechanical properties of the vessel material are tabulated in Table E-1. The stress-strain curves (actual and assumed for Inconel-600) are shown in Figure E-1.

Table E-1Mechanical Properties of Inconel Alloy-600

<u>Form and Condition</u>	<u>Temp (°F)</u>	<u>Yield Strength 0.2% offset (psi)</u>	<u>Tensile Strength (psi)</u>	<u>Elongation (%)</u>	<u>Source</u>
Forging, small grain soft	75	40,500	92,000	44	Ref. (1)
Plate	75	36,500	90,500	47	Ref. (1)
Plate	600	31,000	90,500	46	Ref. (1)
Coupon	75	45,100	94,560	50	Ref. (2)
Coupon	600	33,200	87,700	47	Ref. (3)

* Ed. note: This Appendix was written at the time when the reactor vessel material was Inconel-600. The work described herein has not been repeated for the stainless steel vessel described in Section IV-D above.

FIG. E-1
ESTIMATED STRESS-STRAIN RELATION FOR INCONEL-600

E-II. Strain Energy of the Vessel

For the purpose of analysis, it is assumed that the Inconel-600 alloy has a minimum tensile strength of 90,500 psi and a minimum ultimate strain (ϵ_u) of 45%. The stress-strain relation for power law strain hardening can be expressed by

$$\sigma = 132,000 \epsilon^{0.5}$$

In terms of radial displacement, the assumed equation of state is

$$\left(\frac{PR_i}{X_i} \right) = 132,000 \left(\frac{R-R_i}{R_i} \right)^{0.5} \quad (E-1)$$

where	P	= the resisting pressure
	R _i	= undeformed radius (48 in.)
	R	= deformed radius
	X _i	= shell thickness (3 1/2 in.)

The deformation of the vessel for a given radial strain can be expressed by⁴

$$V-V_o = v = 1.44 V_o \left(\frac{R-R_i}{R_i} \right) \quad (E-2)$$

where	V _o	= undeformed volume (754 ft ³)
	V	= deformed volume
	v	= incremental volume.

Substituting Eq. (E-1) into Eq. (E-2) and expressing P in lb/ft², one obtains

$$v = CP^2 \quad (E-3)$$

where	C	= 5.69 x 10 ⁻¹⁰ .
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The strain energy (E_s) in the vessel can now be obtained from the expression,

$$E_s = \int_0^P P dv = .667 P v \text{ ft-lb.} \quad (\text{E-4})$$

For $\epsilon = \epsilon_u$ and $\epsilon = 1/3 \epsilon_u$, the strain energies of the vessel are 408 MW-sec and 79 MW-sec, respectively. The associated bursting pressures are 6,420 psi and 3,730 psi. The use of 1/3 the ultimate strain has been recommended in Reference (5) for the safe design of reactor pressure vessels for TNT-equivalent detonations. However, in the AARR vessel, the belt section of the shell, the nozzles, and piping are adequately reinforced to preclude premature failure at these locations. Therefore, it is believed that the marginal strain will be much greater than 1/3 ϵ_u and the vessel will undergo a gross ductile deformation.

(Recent preliminary design studies currently in progress by the vessel vendor indicate that the vessel wall may be reduced to 2-7/8 in. above and below the midsection containing the beam tube nozzles. The effect of such a reduction in wall thickness on the vessel containment potential has been determined. The revised vessel strain energies would be 340 MW-sec and 66 MW-sec for full ultimate strain and 1/3 ultimate strain, respectively.)

E-III. Containment Potential

The effect of a large nuclear excursion would be to melt, and perhaps vaporize, a portion of the aluminum in the fuel plates. Consequently, a nuclear excursion could initiate an aluminum-water reaction. The effect of such a reaction has been considered in the preliminary analysis of the AARR reactor pressure vessel containment potential. It was postulated that the destructive effects of such an excursion would appear in two stages. The first stage, with duration measured in milliseconds, would involve rapid generation of superheated steam and perhaps an exothermic reaction of any vaporized aluminum with steam to Al_2O_3 , hydrogen gas, and more steam. This first stage would most probably disperse the remaining (or all) molten aluminum in the water.

The second stage, measured in hundreds of milliseconds to seconds, involves heat transfer- and diffusion-limited burning of the molten aluminum particles that exceed the ignition temperature of about 3000°R under transient heating conditions (see Appendix B, Section B-III above). In Section VIII-B above it was predicted that burning of metal would proceed at a slow rate and require approximately one-half second for full ignition and release of 1300 MW-sec.

The mechanical features of the primary coolant system are not sufficiently defined at this time to justify a determination of the distribution of energy absorption throughout the system or the total energy absorption capability of the system. Actually, the time period of one-half second for full ignition in the second stage is sufficiently long to permit

certain factors to enter which would significantly reduce the likelihood that the integrity of the primary system would be breached as a result of reactivity excursions (see Section VIII-B above). Factors that would tend to mitigate the destructive potential of the second stage reaction include plastic expansion of the vessel, piping, and components, flow of water through the pressure relief system, collapse of steam bubbles, and dissolution of evolved hydrogen gas in the water. However, even though the ignition of aluminum in the second stage is slow enough to result in a static load throughout the primary system, only the pressure vessel was assumed to absorb the strain energy. This approach, which takes no credit for absorption of energy by the remaining primary system, is conservative because it essentially assumes that the energy release is prompt.

The same effect from a nuclear excursion was assumed for AARR as was previously derived for HFIR. That is, the same fraction of the excursion energy transmitted to strain energy in the vessel was used as given in Reference (6).

Figure E-2 shows the strain energy imparted to the reactor vessel by various steam explosions not accompanied or followed by metal-water reactions. Figure E-3 shows similar data for steam explosions accompanied by prompt reaction of vaporized aluminum with water.

The maximum energy releases calculated to be containable by the AARR reactor vessel are summarized in Table E-2. A nuclear excursion of approximately 3,000 MW-sec can be contained, based on the ultimate strength of the vessel, if there is no aluminum-water reaction. The allowable excursion to cause $1/3$ ultimate strain in the vessel is about 800 MW-sec. If the nuclear excursion causes prompt reaction of any vaporized aluminum, the allowable nuclear excursion is reduced to about 850 and 425 MW-sec for ultimate strain and $1/3$ ultimate strain, respectively.

FIG. E-2
STRAIN ENERGY IN AARR REACTOR VESSEL FOR
NUCLEAR EXCURSIONS WITHOUT METAL-WATER REACTION

FIG. E-3

STRAIN ENERGY IN AARR REACTOR VESSEL FOR
NUCLEAR EXCURSIONS PLUS PROMPT METAL-WATER REACTION

Table E-2Containable Energy Releases in AARR Reactor Vessel

<u>Type of Event</u>	<u>Maximum Containable Energy Release^a (MW-sec)</u>	
	<u>Nuclear</u>	<u>Al-H₂O Reaction</u>
Nuclear excursion, "steam explosion"	3000	-
With 1/3 ultimate strain	800	-
Nuclear excursion plus prompt Al-H ₂ O reaction	850	1500
With 1/3 ultimate strain	425	350

^a Based on limiting values of 408 and 79 MW-sec of strain energy deposited in the vessel wall, assuming ultimate strain and 1/3 ultimate strain, respectively.

Based on the analysis in Section VIII-B above a nuclear excursion of 163 MW-sec can be accompanied by a release of approximately 1300 MW-sec from the ignition of the molten aluminum. Under these conditions ultimate strain can be assumed and the containment potential of the reactor vessel is more than adequate to withstand the total energy release.

References for Appendix E

1. "Engineering Properties of Inconel Alloy 600," Technical Bulletin T-7, Huntington Alloy Products Div., International Nickel Company (1964).
2. D. H. Coplin, et al., "Effect of Irradiation on the Tensile Properties of Zircaloy-2, Incoloy-800 and Inconel-600," GEAP-5100-9 (June 1966).
3. B. C. Beaudreau, et al., "Tensile Properties of Fuel Cladding Materials Irradiated in Superheat Environments," GEAP-4754 (April 1966).
4. M. A. Salmon, et al., "Studies of Reactor Containment Structures," IITRI-578P22-14.
5. W. R. Wise, Jr. and J. F. Proctor, "Explosion Containment Laws for Nuclear Reactor Vessels," NOLTR-63-140 (August 1966).
6. T. E. Cole, "High Flux Isotope Reactor - Safety Review Questions and Answers," ORNL-65-11-29, Supplement No. 2 (August 15, 1966).

APPENDIX FCONTAINMENT OVERPRESSURE CALCULATION METHODContents

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APPENDIX F

CONTAINMENT OVERPRESSURE CALCULATION METHOD

F-I Introduction

To analyze the overpressures which could be developed in the AARR Reactor Containment Building as the result of a postulated accident and to provide a sound basis for selecting the design pressure rating for the building, a method has been developed to enable calculation of the Reactor Containment Building overpressure at any desired time after occurrence of an assumed accident. This method takes into account the time-dependent heat transfer among the containment atmosphere, the containment building structure, various structures and equipment within the building, the Reactor Pool water, the spilled primary water, and the outside surroundings.

A description of the method developed is presented here. Details, results and analyses of the cases studied will be found in Section VII-B-2 above.

F-II Primary Descriptive Equations

To conveniently represent, in mathematical terms, the heat transfer processes treated herein, the containment system studied is separated into the following subsystems: (1) containment atmosphere which includes both the contained dry air and the contained water vapor, (2) dome, (3) wall above grade, (4) wall below grade, (5) basement floor, (6) internal structures, equipment, and concrete, (7) Reactor Pool structure, (8) Reactor Pool water, and (9) spilled primary water.

The primary descriptive equations are based on the following assumptions: (1) air obeys the perfect gas law, (2) there is perfect and instantaneous mixing of air and water vapor, and (3) there are no temperature gradients within the containment atmosphere or the pool water or spilled water. As a consequence of the latter assumption, it is implicitly assumed that any heat transferred is instantaneously and uniformly distributed within any gas or fluid subsystem. Transient heat conduction is accounted for in each solid subsystem.

The primary equations describing the energy balance for each subsystem at the beginning and end of each time interval are given as follows (see Table F-1 for nomenclature, which is a combination of conventional usage and FORTRAN computer usage):¹⁻⁴

TABLE F-1

NOMENCLATURE FOR APPENDIX F

Symbol	Description	Units
a	Constant in decay heat equation	none
c	Specific heat	Btu/lb-°F
c_{v_a}	Specific heat of air at constant volume	Btu/lb-°F
h	Convective heat transfer coefficient	Btu/hr-ft ² -°F
$h_{a \rightarrow 1}$	Convective heat transfer coefficient from the containment atmosphere to the dome	Btu/hr-ft ² -°F
$h_{a \rightarrow 2}$	Convective heat transfer coefficient from the containment atmosphere to the wall above grade	Btu/hr-ft ² -°F
$h_{a \rightarrow 3}$	Convective heat transfer coefficient from the containment atmosphere to the wall below grade	Btu/hr-ft ² -°F
$h_{a \rightarrow 4}$	Convective heat transfer coefficient from the containment atmosphere to the internals	Btu/hr-ft ² -°F
$h_{a \rightarrow 9}$	Convective heat transfer coefficient from the containment atmosphere to the reactor pool enclosure	Btu/hr-ft ² -°F
$h_{a \rightarrow p}$	Convective heat transfer coefficient from the containment atmosphere to the reactor pool water	Btu/hr-ft ² -°F
$h_{a \rightarrow L}$	Convective heat transfer coefficient from the containment atmosphere to the spilled primary water	Btu/hr-ft ² -°F
$h_{p \rightarrow 9}$	Convective heat transfer coefficient from the reactor pool water to the reactor pool enclosure	Btu/hr-ft ² -°F
$h_{L \rightarrow 7}$	Convective heat transfer coefficient from the spilled primary water to the basement floor	Btu/hr-ft ² -°F
k	Thermal conductivity	Btu/hr-ft-°F
P_a	Air pressure inside the containment	psia
P_{over}	Containment overpressure	psig
P_{sat}	Saturated vapor pressure computed at the average water temperature for the time interval	psia
P_v	Vapor pressure computed at the average air temperature and average water vapor inventory for the time interval	psia
$q_{a \rightarrow 1}$	Convective heat transfer rate from containment atmosphere to dome	Btu/hr
$q_{a \rightarrow 2}$	Convective heat transfer rate from containment atmosphere to wall above grade	Btu/hr
$q_{a \rightarrow 3}$	Convective heat transfer rate from containment atmosphere to wall below grade	Btu/hr
$q_{a \rightarrow 4}$	Convective heat transfer rate from containment atmosphere to internals	Btu/hr
$q_{a \rightarrow 9}$	Convective heat transfer rate from containment atmosphere to pool enclosure	Btu/hr
$q_{a \rightarrow p}$	Convective heat transfer rate from containment atmosphere to reactor pool water	Btu/hr
$q_{a \rightarrow L}$	Convective heat transfer rate from containment atmosphere to spilled primary water	Btu/hr
$q_{p \rightarrow 9}$	Convective heat transfer rate from reactor pool water to pool enclosure	Btu/hr

TABLE F-1 (continued)

Symbol	Description	Units
$q_L \rightarrow 7$	Convective heat transfer rate from spilled primary water to basement floor	Btu/hr
t	Time after shutdown	sec
t_a	A containment atmosphere temperature	°F
t_{a1}	Containment atmosphere temperature at the beginning of the time interval	°F
t_{a2}	Containment atmosphere temperature at the end of the time interval	°F
$t_{a2(i-1)}$	Containment atmosphere temperature at the end of the time interval determined by the (i-1) iteration	°F
$t_{a2(i)}$	Containment atmosphere temperature at the end of the time interval determined by the i th iteration	°F
t_L	A spilled water temperature	°F
$t_{L2(i-1)}$	Spilled water temperature at the end of the time interval determined by the (i-1) iteration	°F
$t_{L2(i)}$	Spilled water temperature at the end of the time interval determined by the i th iteration	°F
t_p	A pool water temperature	°F
$t_{p2(i-1)}$	Pool water temperature at the end of the time interval determined by the (i-1) iteration	°F
$t_{p2(i)}$	Pool water temperature at the end of the time interval determined by the i th iteration	°F
o_{tg}	Inner surface temperature of the pool enclosure	°F
g_t	Temperature of the gas or liquid next to the surface of a solid subsystem at the beginning of the time interval	°F
$g_{t_{\Delta\tau}}$	Temperature of the gas or liquid next to the surface of a solid subsystem at the end of the time interval	°F
i_t	Solid subsystem temperature at a node point adjacent to a surface node point at the beginning of the time interval	°F
$i_{t_{\Delta\tau}}$	Solid subsystem temperature at the node point adjacent to a surface node point at the end of the time interval	°F
$n-1_t$	Temperature of the (n-1) interior node point at the beginning of the time interval	°F
$n-1_{t_{\Delta\tau}}$	Temperature of the (n-1) interior node point at the end of the time interval	°F
n_t	Temperature of the n th interior node point at the beginning of the time interval	°F
$n_{t_{\Delta\tau}}$	Temperature of the n th interior node point at the end of the time interval	°F
$n+1_t$	Temperature of the (n+1) interior node point at the beginning of the time interval	°F
$n+1_{t_{\Delta\tau}}$	Temperature of the (n+1) interior node point at the end of the time interval	°F
s_t	Surface temperature of a solid subsystem at the beginning of the time interval	°F
$s_{t_{\Delta\tau}}$	Surface temperature of a solid subsystem at the end of the time interval	°F
s_{t_1}	Inner surface temperature of the dome	°F
s_{t_2}	Inner surface temperature of the wall above grade	°F
s_{t_3}	Inner surface temperature of the wall below grade	°F

TABLE F-1 (continued)

Symbol	Description	Units
s_{t4}	Surface temperature of the internals	$^{\circ}\text{F}$
s_{t7}	Inner surface temperature of the basement floor	$^{\circ}\text{F}$
s_{t9}	Outer surface temperature of the pool enclosure	$^{\circ}\text{F}$
w	Evaporation rate for water	lb/hr
A	Constant in decay heat equation	none
$A_{a \rightarrow 1}$	Effective dome area for convective heat transfer to or from the containment atmosphere and to or from the outside atmosphere	ft^2
$A_{a \rightarrow 2}$	Effective area of the wall above grade for convective heat transfer to or from the containment atmosphere and to or from the outside atmosphere.	ft^2
$A_{a \rightarrow 3}$	Effective area for convective heat transfer between the wall below grade and the containment atmosphere	ft^2
$A_{a \rightarrow 4}$	Effective area for convective heat transfer between the containment atmosphere and the internals	ft^2
$A_{a \rightarrow 9}$	Effective area for convective heat transfer between the containment atmosphere and the reactor pool enclosure	ft^2
$A_{a \rightarrow p}$	Effective area for convective heat transfer between the containment atmosphere and the reactor pool water	ft^2
$A_{a \rightarrow L}$	Effective area for convective heat transfer between the containment atmosphere and the spilled primary water	ft^2
$A_{p \rightarrow 9}$	Effective area for convective heat transfer between the reactor pool water and the pool enclosure	ft^2
$A_{L \rightarrow 7}$	Effective area for convective heat transfer between the spilled primary water and the basement floor	ft^2
Area	Surface area of a solid subsystem	ft^2
E	Decay heat energy transferred directly to the containment atmosphere during the time interval	Btu
L	Mass leakage rate	w/o/day
P_0	Reactor operating power	MW
P_1	Reactor decay power due to residual fission after shutdown	MW
P_2	Reactor decay power due to heat emission from fission products	MW
$Q_{a \rightarrow 1}$	Heat transferred by convection from containment atmosphere to dome during time interval	Btu
$Q_{a \rightarrow 2}$	Heat transferred by convection from containment atmosphere to wall above grade during time interval	Btu
$Q_{a \rightarrow 3}$	Heat transferred by convection from containment atmosphere to wall below grade during time interval	Btu
$Q_{a \rightarrow 4}$	Heat transferred by convection from containment atmosphere to internals during time interval	Btu
$Q_{a \rightarrow 9}$	Heat transferred by convection from containment atmosphere to reactor pool enclosure during time interval	Btu
$Q_{a \rightarrow p}$	Heat transferred by convection from containment atmosphere to reactor pool water during time interval	Btu
$Q_{a \rightarrow L}$	Heat transferred by convection from containment atmosphere to spilled primary water during time interval	Btu
$Q_{p \rightarrow 9}$	Heat transferred by convection from reactor pool water to reactor pool enclosure during time interval	Btu
$Q_{L \rightarrow 7}$	Heat transferred by convection from spilled primary water to basement floor during time interval	Btu
Q_{refrig}	Heat removed from containment atmosphere by refrigeration during time interval	Btu
T_0	Reactor operation time at full power	sec
Tolerance	Convergence criterion	none

TABLE F-1 (continued)

Symbol	Description	Units
U_{a1}	Internal energy of the contained air at the beginning of the time interval	Btu/lb
U_{a2}	Internal energy of the contained air at the end of the time interval	Btu/lb
U_{L1}	Internal energy of the spilled primary water at the beginning of the time interval	Btu/lb
U_{L2}	Internal energy of the spilled primary water at the end of the time interval	Btu/lb
$U_{L@t_a}$	Internal energy of the water at the average containment atmosphere temperature of the time interval	Btu/lb
U_{p1}	Internal energy of the reactor pool water at the beginning of the time interval	Btu/lb
U_{p2}	Internal energy of the reactor pool water at the end of the time interval	Btu/lb
U_{v1}	Internal energy of the water vapor in the containment atmosphere at the beginning of the time interval	Btu/lb
U_{v2}	Internal energy of the water vapor in the containment atmosphere at the end of the time interval.	Btu/lb
$U_{v@t_a}$	Internal energy of the water vapor at the average containment atmosphere temperature of the time interval	Btu/lb
$U_{v@t_L}$	Internal energy of the water vapor at the average spilled water temperature of the time interval	Btu/lb
$U_{v@t_p}$	Internal energy of the water vapor at the average pool water temperature of the time interval	Btu/lb
Volume	Total free volume of the containment atmosphere	ft ³
W_a	Air inventory at the beginning of the time interval	lb
W_c	Amount of water vapor condensed during the time interval	lb
W_L	Inventory of spilled primary water at the beginning of the time interval	lb
W_p	Inventory of pool water at the beginning of the time interval	lb
W_v	Inventory of water vapor at the beginning of the time interval	lb
W_x	Amount of spilled primary water evaporated during the time interval	lb
W_y	Amount of pool water evaporated during the time interval	lb
β	Delayed neutron fraction	none
$\Delta\tau$	A time interval	sec
$\Delta x/2$	Half interval on each side of a node point in a solid subsystem	ft
λ	Decay constant of fission products	sec ⁻¹
ρ	Step change in reactivity, or Material density	\$ lb/ft ³
$\left(\frac{k}{\Delta x}\right)_{n-1 \rightarrow n}$	Conductivity and distance parameter between the (n-1) node point and the <u>n</u> th node point	Btu/hr-ft ² -°F
$\left(\frac{k}{\Delta x}\right)_{n \rightarrow n+1}$	Conductivity and distance parameter between the (n+1) node point and the <u>n</u> th node point	Btu/hr-ft ² -°F
$\left(\rho c \frac{\Delta x}{2}\right)_{n-1 \rightarrow n}$	Heat storage parameter of the half-thickness toward the (n-1) node point associated with the <u>n</u> th node point	Btu/ft ² -°F
$\left(\rho c \frac{\Delta x}{2}\right)_{n \rightarrow n+1}$	Heat storage parameter of the half-thickness toward the (n+1) node point associated with the <u>n</u> th node point	Btu/ft ² -°F
$\left(\rho c \frac{\Delta x}{2}\right)_s$	Heat storage parameter of the half-thickness on a surface associated with the surface node point	Btu/ft ² -°F

Containment Atmosphere

$$\begin{aligned}
& W_a U_{a1} + W_v U_{v1} + W_x U_{x@t_L} + W_y U_{y@t_p} \\
& + W_c U_{v@t_a} - \left(Q_{a \rightarrow 1} + Q_{a \rightarrow 2} + Q_{a \rightarrow 3} \right. \\
& \left. + Q_{a \rightarrow 4} + Q_{a \rightarrow 9} + Q_{a \rightarrow p} + Q_{a \rightarrow L} + Q_{\text{refrig}} \right) \\
& + E = W_a U_{a2} + (W_v + W_x + W_y) U_{v2} \\
& + W_c U_{L@t_a} + (W_c - W_x - W_y) \left(\frac{144 p_v}{778.2} \right) \left(\frac{\text{Volume}}{W_v} \right)
\end{aligned} \tag{F-1}$$

where

$$U_{a1} - U_{a2} = c_{v_a} (t_{a1} - t_{a2}) \tag{F-1a}$$

Pool Water

$$\begin{aligned}
& (W_p - W_y) U_{p1} + W_y U_{p1} + Q_{a \rightarrow p} - Q_{p \rightarrow 9} \\
& = (W_p - W_y) U_{p2} + W_y U_{v@t_p} + W_y \left(\frac{144 p_v}{778.2} \right) \left(\frac{\text{Volume}}{W_v} \right)
\end{aligned} \tag{F-2}$$

Spilled Water

$$\begin{aligned}
& (W_L - W_x) U_{L1} + W_x U_{L1} + W_c U_{L@t_a} \\
& + Q_{a \rightarrow L} - Q_{L \rightarrow 7} = (W_L - W_x) U_{L2} \\
& + W_x U_{v@t_L} + (W_x - W_c) \left(\frac{144 p_v}{778.2} \right) \left(\frac{\text{Volume}}{W_v} \right)
\end{aligned} \tag{F-3}$$

Solid Conducting Subsystem

a. Exterior Node Points

$$\begin{aligned} & \left\{ h \left[\frac{1}{2} (g_t + g_{t_{\Delta\tau}}) - \frac{1}{2} (s_t + s_{t_{\Delta\tau}}) \right] \Delta\tau \right. \\ & \quad \left. + \frac{k}{\Delta x} \left[\frac{1}{2} (i_t + i_{t_{\Delta\tau}}) - \frac{1}{2} (s_t + s_{t_{\Delta\tau}}) \right] \Delta\tau \right\} \text{ (Area)} \\ & = \left\{ \left[\rho c \frac{\Delta x}{2} \right]_s (s_{t_{\Delta\tau}} - s_t) \right\} \text{ (Area)} \end{aligned} \quad (\text{F-4a})$$

b. Interior Node Points

$$\begin{aligned} & \left\{ \left(\frac{k}{\Delta x} \right)_{n-1 \rightarrow n} \left[\frac{1}{2} (n_{t-1} + n_{t_{\Delta\tau}}) - \frac{1}{2} (n_t + n_{t_{\Delta\tau}}) \right] \Delta\tau \right. \\ & \quad \left. + \left(\frac{k}{\Delta x} \right)_{n \rightarrow n+1} \left[\frac{1}{2} (n_{t+1} + n_{t_{\Delta\tau}}) - \frac{1}{2} (n_t + n_{t_{\Delta\tau}}) \right] \Delta\tau \right\} \text{ (Area)} \\ & = \left\{ \left[\left(\rho c \frac{\Delta x}{2} \right)_{n-1 \rightarrow n} + \left(\rho c \frac{\Delta x}{2} \right)_{n \rightarrow n+1} \right] (n_{t_{\Delta\tau}} - n_t) \right\} \text{ (Area)} \end{aligned} \quad (\text{F-4b})$$

F-III Secondary Descriptive Equations

Many of the parameters contained in the primary descriptive equations are described by additional equations. These equations are given in the following:

Heat Transfer Rates

The various heat transfer rates found in the primary equations are given as follows (units of Btu/hr):

$$q_{a \rightarrow 1} = h_{a \rightarrow 1} A_{a \rightarrow 1} (t_a -^s t_1) \quad (\text{F-5})$$

$$q_{a \rightarrow 2} = h_{a \rightarrow 2} A_{a \rightarrow 2} (t_a -^s t_2) \quad (\text{F-6})$$

$$q_{a \rightarrow 3} = h_{a \rightarrow 3} A_{a \rightarrow 3} (t_a -^s t_3) \quad (\text{F-7})$$

$$q_{a \rightarrow 4} = h_{a \rightarrow 4} A_{a \rightarrow 4} (t_a -^s t_4) \quad (\text{F-8})$$

$$q_{a \rightarrow 9} = h_{a \rightarrow 9} A_{a \rightarrow 9} (t_a -^s t_9) \quad (\text{F-9})$$

$$q_{a \rightarrow p} = h_{a \rightarrow p} A_{a \rightarrow p} (t_a - t_p) \quad (F-10)$$

$$q_{a \rightarrow L} = h_{a \rightarrow L} A_{a \rightarrow L} (t_a - t_L) \quad (F-11)$$

$$q_{p \rightarrow 9} = h_{p \rightarrow 9} A_{p \rightarrow 9} (t_p - t_9) \quad (F-12)$$

$$q_{L \rightarrow 7} = h_{L \rightarrow 7} A_{L \rightarrow 7} (t_L - t_7) \quad (F-13)$$

$$\text{where } Q = q \Delta \tau \quad (F-14)$$

Enthalpies and Saturated Vapor Pressure

Equations fitted to handbook values⁵ are used to determine water and water vapor internal energies and saturated vapor pressure as functions of the appropriate temperatures.

Specific Heat of Air

The specific heat of air at constant volume as a function of temperature is given⁶ as follows:

$$c_{v_a} = 0.243 - \frac{0.889}{\sqrt{t_a + 459.69}} - \frac{58.964}{(t_a + 459.69)} + \frac{22348.05}{(t_a + 459.69)^2} \quad (F-15)$$

Air and Vapor Pressure

Air and vapor pressures are determined according to the following equations

$$p_a = \frac{(53.30) (W_a) (t_a + 459.69)}{144 (\text{Volume})} \quad (F-16)$$

$$p_v = \frac{(85.81) (W_v) (t_a + 459.69)}{144 (\text{Volume})} \quad (F-17)$$

Water Evaporation

The evaporation rate for water is given as follows:⁷

$$w = (0.156) (A_{a \rightarrow L}) (p_{\text{sat}} - p_v)^{1.2} \text{ lb/hr} \quad (\text{F-18})$$

Decay Heat

The rate at which decay heat is generated in the core is given in two parts.⁸ One part, the residual reactor fission power remaining following reactor shutdown, is given as follows:

$$P_1 = (948) (P_o) \left(\frac{\beta}{\beta - \rho} \right) \left[\exp \left(\frac{\rho \lambda t}{\beta - \rho} \right) \right] \frac{\text{Btu}}{\text{sec}} \quad (\text{F-19})$$

The other part is the fission product decay energy given by

$$P_2 = 4.74 P_o \left[A t(t)^{-at} - A(T_o + t)(T_o + t)^{-a(T_o + t)} \right] \quad (\text{F-20})$$

where $A = 12.05$, $a = 0.0639$ for $0.1 \leq \text{time} \leq 10 \text{ sec}$
 $= 15.31$, 0.1807 $10 < \text{time} \leq 150 \text{ sec}$
 $= 26.02$, 0.2834 $150 < \text{time} \leq 4 \times 10^6 \text{ sec (46.3 days)}$
 $= 53.18$, 0.3350 $4 \times 10^6 < \text{time} \leq 2 \times 10^8 \text{ sec (6.34 years)}$

Leakage

The leakage rate is given by

$$L \propto \sqrt{p_{\text{over}}} \quad (\text{F-21})$$

F-IV Method of Solution

The primary equations and supporting secondary equations for the gaseous and liquid subsystems are solved for each time interval using a numerical iterative procedure. All initial values are known for each time interval, being either input values or determined as end values from calculations for a preceding time interval. Calculations then proceed as follows:

A. First Iteration in a Time Interval

1. For a first guess of an acceptable solution, the following quantities are calculated in order using initial temperatures and water vapor inventory: p_{sat} , p_v , W_x , W_y , $Q_{a \rightarrow 1}$, $Q_{a \rightarrow 2}$, $Q_{a \rightarrow 3}$, $Q_{a \rightarrow 4}$, $Q_{a \rightarrow 9}$, $Q_{a \rightarrow p}$, $Q_{a \rightarrow L}$, $Q_{p \rightarrow 9}$, and $Q_{L \rightarrow 7}$.

2. Decay energy is calculated.

3. Equations F-1 and F-1a are solved to obtain t_{a2} .

4. Equations F-2 and F-3 are solved to obtain U_{L2} and U_{p2} .

(The quantities $U_{v@t_L}$ and $U_{v@t_p}$ are evaluated at the initial water temperatures as a first guess.) The first guess for the final water temperatures is then obtained from U_{L2} and U_{p2} using inversely fitted equations based on handbook values.⁵

B. Subsequent Iterations in the Time Interval

1. The following quantities are calculated in order using the previously computed values obtained for the interval averages of the temperatures and water vapor inventory: P_{sat} , P_v , W_x , W_y , $Q_{a \rightarrow 1}$, $Q_{a \rightarrow 2}$, $Q_{a \rightarrow 3}$, $Q_{a \rightarrow 4}$, $Q_{a \rightarrow 9}$, $Q_{a \rightarrow p}$, $Q_{a \rightarrow L}$, $Q_{L \rightarrow 7}$, $U_{v@t_L}$, and $U_{v@t_p}$.

2. A value is obtained for U_{v2} using the last computed value of t_{a2} .

3. Equations F-1 and F-1a are solved to obtain a better value for t_{a2} .

4. Equations F-2 and F-3 are solved to obtain better values for U_{L2} and U_{p2} . The final water temperatures (better values) are then obtained from the inversely fitted equations.

5. The solid subsystem initial surface temperatures are used throughout these iterations and new surface temperatures are not computed during this set of iterations.

C. Convergence

1. Convergence is satisfied, and the computations proceed to part D below when the following criteria are met:

$$\left| \frac{t_{L2(i)} - t_{L2(i-1)}}{t_{L2(i)}} \right| \leq \text{Tolerance} \quad (\text{F-22a})$$

$$\left| \frac{t_{p2(i)} - t_{p2(i-1)}}{t_{p2(i)}} \right| \leq \text{Tolerance} \quad (\text{F-22B})$$

$$\left| \frac{t_{a2(i)} - t_{a2(i-1)}}{t_{a2(i)}} \right| \leq \text{Tolerance} \quad (\text{F-22c})$$

2. If convergence is not satisfied after a reasonable number of iterations, then the problem is terminated at the point where it fails to converge.

D. Subsequent Computations Following Convergence in a Time Interval

1. At this point values have been obtained for the temperatures and pressures of the gaseous and liquid subsystems at the end of the time interval and for the amount of water evaporated during the time interval.

2. The containment atmosphere is checked to determine if it is supersaturated at its final interval temperature. If so, a certain amount of water vapor is allowed to condense out. The heat released by vapor condensation is allowed to go into the containment atmosphere, thus raising its temperature somewhat, and the end result is that the containment atmosphere is saturated at its adjusted, slightly higher final interval temperature.

3. End-of-interval surface and interior temperatures are computed for the solid subsystems using an implicit computation method applied to the equation set F-4.

4. Overpressure is calculated as the difference between the total pressure within the containment and the outside, atmospheric pressure.

5. The total leakage for the time interval is computed using equation F-21, and the weights of dry air and water vapor in the containment atmosphere are adjusted downward to account for this leakage.

E. Computations in the Next Time Interval

Final values computed in parts A-D above become initial values for subsequent calculations in the next time interval.

F. Computer Utilization

A computer program has been written for the CDC-3600 computer which carries out the calculations herein described. It is named Program PTHISTRY and has the ANL number designation RE-360X.

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PART 2

QUESTIONS AND ANSWERS ON THE ARGONNE ADVANCED
RESEARCH REACTOR PRELIMINARY SAFETY ANALYSIS REPORT

Compiled and Edited

by

I. Charak, Safety Coordinator

Reactor Engineering Division

PART 2

QUESTIONS AND ANSWERS ON THE ARGONNE ADVANCED

RESEARCH REACTOR PRELIMINARY SAFETY ANALYSIS REPORT

I. INTRODUCTION

As noted in the Preface to this report, Part 1 represents the bulk of the original PSAR text as transmitted to the AEC. In the course of the PSAR review by the Commission, numerous questions were formally raised and responses to those questions generated by the Laboratory. Extensive additional documentation containing the questions and responses was prepared during the 19-month review period. It is the purpose of this part of the report to consolidate all the questions and answers between a single pair of covers for a complete historical record of the PSAR review.

The questions listed in Section III below have been modified only to the extent that they have been numbered sequentially for easy reference, and references in the questions to Laboratory-prepared material have been made consistent with the present report where feasible. Also, the AEC organization originating the question is given. The following abbreviations are used:

CH - Chicago Operations Office
RDT - Division of Reactor Development and Technology
DRL - Division of Reactor Licensing

In general, where the answer to a question refers to the text of Part 1 above, that answer was incorporated in the revised PSAR where it was deemed appropriate. In the interest of historical accuracy, no attempt has been made to remove inconsistencies between Part 1 and the questions and answers, since the PSAR review extended over a long time period and numerous changes in design as well as design philosophy were reflected in major revisions to Part 1. It should be emphasized, however, that the present version of Part 1 accurately describes the facility that was approved for construction by the Commission.

Page, figure and table numbers in the questions usually refer to obsolete material in early draft versions of the PSAR. The reader is advised not to refer to any quoted sections of the PSAR unless the reference specifically mentions Part 1 or a question number.

Section II is an attempt to index the questions using a few key words, so that questions relating to specific areas may be readily identified. Section IV is a listing of the communications in which the various questions and Laboratory responses were contained. Finally, Appendix A is a reproduction of the ACRS letter to the Commission on the AARR PSAR review and Appendix B presents the seismic design criteria which were to be employed in the design of Class-I (earthquake-resistant) components, systems and structures.

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III. QUESTIONS AND ANSWERS

A. Questions from Reference (1).*

1. *Criterion 1:*

(a) *It appears that it will be most difficult to show that a particular structural design for control room, etc., will be unaffected by tornado winds of up to 300 miles per hour. Also that emergency heat removal systems will continue to function after direct tornado hit. (CH)*

Answer:² The control room, relay rack room, and instrument cable room of the Laboratory and Office Building and the battery and generator floor of the Electrical Equipment Building have all been located below grade so as to be unaffected by wind pressures and missiles generated by a tornado. Floor slabs over these areas are to be designed to resist uplift in event that the eye of a tornado with its negative pressure should pass directly over them. "Hardhatting" of these areas in this manner will provide additional assurance that personnel and equipment located in these areas will survive a direct tornado hit on the facility.

The cooling tower will perhaps be severely damaged by a direct tornado hit but the main secondary cooling pumps will not be damaged due to the protection afforded them by the concrete cubicle in which they are housed. The cooling tower basin is structurally designed to be unaffected by tornadoes, and the basin is also designed to provide an additional amount of depth, with the subsequent increase in water inventory which can be used as a heat sink after destruction of its superstructure. Also, provision is made for direct make-up of water to the basin using Laboratory Water as well as Canal Cooling Water, in the event of loss of tower basin water by suction from a passing tornado. (See also Part 1, Section VI-F-2c.)

(b) *Paragraph 1.5 states that the upper parts of the Reactor Containment Building will be designed to withstand wind pressure developed by basic winds up to 115 mph. Paragraph 1.6 implies that this building will be required to withstand wind velocities up to 300 mph. This apparent difference in meaning should be clarified. (CH)*

Answer:² The Reactor Containment Building with its massive shielding walls is inherently capable of withstanding structurally the forces imposed upon it by winds with velocities up to 300 mph. Reinforcement in the walls and dome, dictated by the design parameters of internal overpressure (7.5 psig) and seismic loading, is more than adequate to supply the needs of reinforcement required for this high wind loading. Criterion 1.5 will be changed to read as follows:

* Ed. note: Answers to these questions were furnished in References 2-4, and they are accordingly superscripted.

"The Reactor Containment Building structure, excluding insulation and roof covering, will be designed to withstand the total force that can be developed by a wind with a basic velocity of 300 mph at 30 feet above grade and applied to the projected elevation of from 0 to 50 feet above grade and for the total force developed by a wind with a basic velocity of 340 mph applied to the projected elevation of from 50 to 75 feet above grade."

Criterion 1.6 has been revised to reflect the changes made to Criterion 1.5.

2. *Criterion 2:*

(a) *A discussion should be included retarding the possibility of a metal-water reaction involving beryllium and water. What effect would such a reaction have on containment design pressure? (CH)*

Answer:³ See Part 1, Section VIII-C.

(b) *It is noted that the emergency cooling system will serve to limit the extent of metal-water reactions. It can also be postulated that emergency cooling water can be a source of water for metal-water reaction. (CH)*

Answer:⁴ The intent of the original statement was to indicate that the emergency cooling system served as a deterrent to the inception of metal-water reactions by keeping metal temperatures below the reaction point.

3. *Criterion 5:*

(a) *This implies that gross voiding of the ITC will not be possible. Previous discussions with the Laboratory have not given good assurance that this can be done. (CH)*

Answer:² The Internal Thermal Column (ITC) will contain rabbit tubes, each of which is individually cooled by primary coolant. The rabbit systems are arranged such that simultaneous voiding of more than one tube, or of all the tubes and the volume between the tubes (referred to as the bulk ITC volume), is considered incredible. The reactivity worth of voiding a single rabbit tube is estimated to be no greater than 25 cents, whether or not the sample remains in place. The velocity of the bulk ITC water is low, so that the maximum speed at which voids can be introduced into the bulk ITC volume is low enough to be controllable by control rod action. The design is such as to render incredible the complete blocking or voiding of all water inlets to the ITC bulk volume.

(b) Reasonably detailed criteria as affects design should be included to limit the experimental program scope so that postulated accidents associated with the experimental program do not result in more severe effects than the maximum accident considered.
(CH)

Answer:³ See Part 1, Section II, Criterion 5.2.

4. Criterion 6:

Burnable poison addition effects and peak heat flux considerations should be discussed under Fuel Stamina. (RDT)

Answer:³ See Part 1, Section IV-B-1.

5. Criterion 9:

(a) Discuss whether or not the backup shutdown system is capable of shutting the reactor down in the event of loss of normal scram capability coincident with rupture of the primary outlet line.
(CH)

Answer:⁴ The current AARR design is similar to the HFIR, with respect to the backup shutdown system, in that the poison solution is delivered to the intake of the primary system pressurization pumps (including the emergency pressurization pump) and is forced into the primary coolant loop by those pumps. This is adequate for all credible situations, since during both normal operation and operation under failure of Class-III (utility) power, (1) at least one of the main pressurization pumps or the emergency pump will be in operation at all times, assuring capability to transfer the poison solution into the primary loop, (2) both primary coolant pump pony motors will be operating at all times, assuring capability to distribute the poison throughout the reactor and primary system, and (3) the quantity of poison solution available is sufficient to shut the reactor down from any normal operating condition, assuming distribution of poison throughout the entire primary coolant loop.

In the event of a major rupture of the primary outlet line from the reactor, primary flow would probably cease $\sim 1/2$ min after rupture. As indicated in Part 1, Section VI-D-2, control plate insertion resulting from automatic reactor scram would be complete before any significant fuel melting had occurred, so that there should be no need for soluble poison injection. (The complete failure to scram of all four shim-safety plates simultaneous with primary system rupture is not considered credible.) However, if poison injection were attempted following a rupture, even if the injection were started simultaneously with the rupture, only about 5% of the poison solution (10% if both main pressurization pumps were operating) would be injected before cessation of primary flow. Thereafter, there would be no means of transporting the poison from its point of injection to the reactor.

As indicated in Part 1, Section IV-D, some consideration is currently being given to injecting the poison solution directly into the reactor vessel. However, this would require one or more small lines under full system pressure in the Reactor Pool, and damage to these vulnerable lines could lead to system ruptures. During Title-II design, an evaluation will be made of the relative advantage of ability to inject soluble poison directly into the vessel, as opposed to the hazard of small high pressure lines in the Reactor Pool, and a design change will be made if deemed advisable.

- (b) *The following additional criteria should be included:*
- (1) *This system should be of different principle and independent from the primary control system.*
 - (2) *This system should be designed to provide for testing, or monitoring, to establish the operability of all of the equipment.*
 - (3) *The conditions when this system is to be used and the method of initiation. (RDT)*

Answer:^{3,4} See Part 1, Section II, Criteria 9.1 through 9.3 and Section IV-G-1g.

It is difficult to postulate a realistic situation which would require actuation of the backup shutdown system, since the reactor is designed to possess an adequate shutdown margin of reactivity even with one control plate inoperable and in a position of minimum reactivity control effectiveness (see Part 1, Section II, Criterion 8.2), and the simultaneous failure of actuation of more than one control plate upon generation of a scram signal is not considered credible. This system is included in the facility only to provide added design conservatism. It may be stated generally that the backup shutdown system would be actuated manually from the control room, at such times when, after a manual or automatic power reduction of the plant, there were indications that insufficient negative reactivity had been inserted into the reactor to effect the shutdown to the cold condition with adequate shutdown margin.

6. *AARR Criteria 9.2 states that the backup shutdown system will have sufficient worth to shut down the plant if the control rods do not move from their original position. This criteria should be that the negative reactivity worth of this system exceeds the maximum loaded reactivity of the core; backup shutdown systems on all known PWR plants have this capability. (RDT)*

Answer:³ See Part 1, Section II, Criterion 9.2.

7. *Criterion 10:*

- (a) *In the event of an MCA involving a ruptured outlet line, there would be no reservoir of water supply for the emergency cooling system. In brief, it indicates that redundant components of the emergency cooling system will be provided, however, there is no redundant supply of water. (CH)*

Answer:² The query is specific to the original emergency shutdown and cooling system prior to its revision September 28, 1966. In the revised system (see question 18, below) rupture in the primary system renders the emergency shutdown and cooling system inoperable; therefore, the question relating to redundancy of water supply specifically for the continuance of emergency cooling is inapplicable.

(b) *The criteria include provisions for designing against rapid uncovering of the core in the event of a system rupture. The MCA analysis is based on cooling system being voided in less than one minute. These are mutually contradictory considerations; i.e., either the system is rapidly voided or the core stays covered with water. When the core stays covered with water in an MCA, the analysis should consider the extent of water boil-off and the effect on containment design pressure resulting from a coincident lack of scram. It is noted from the analysis (Appendices) that bulk boiling occurs when the system has depressurized to 275 psig; the bulk water temperature at this time will be considerably above 220°F. (CH)*

Answer:⁴ Criterion 10.3 (Section II) calls for design to minimize the possibility of uncovering the core in the event of primary system rupture. As indicated in Part 1, Section VI-D-2, we do not believe that there is a valid reason to assume that control plate scram would not occur in the event of rupture; therefore, the situation of coincident rupture and lack of scram is not considered credible.

In all four cases of primary system rupture (with scram) discussed in Part 1, Section VI-D-2, the reactor core remains covered with water. In the case of rupture within the Reactor Pool, primary coolant flow continues, and decay heat is removed through the secondary system to the cooling tower. For rupture outside the pool, decay heat would be released to the Reactor Containment Building atmosphere, partly by boil-off of water in the vessel, and partly by conduction through the vessel wall to the pool. The release of decay heat to the Containment Building throughout the MCA overpressure period is included in the MCA computational model, the consequences of which are projected in Part 1, Section VII.

The hypothetical (incredible) cases of coincident rupture and lack of scram are discussed in Part 1, Section VIII-C. In cases of rupture inside the Reactor Pool, without scram, the reactor heat would continue to be removed through the secondary system. However, because of rising Reactor Pool temperatures, overpressure might develop in the Containment Building, if sealed. These overpressures are shown not to exceed the Containment Building design pressure of 7.5 psig. In cases of rupture outside the pool, without scram, the cessation of primary coolant flow would cause a very rapid rise in fuel metal temperatures, and complete meltdown, full fission product release (100% of rare gases, 50% of halogens and 1% of solids), and probably metal-water reactions could be expected to occur in a matter of seconds after the rupture, and the resulting core disassembly would render the reactor subcritical. Therefore, there would be no prolonged water boil-off, except from decay heat and energy from aluminum-water reactions, all of which are included in the MCA computations.

8. *Criterion 11:*

Appropriate analysis and discussion of NDT of reinforcement steel in the containment structural concrete should be included in the PSAR. (CH)

Answer:² The application of the NDT temperature concept (concerning safety from brittle fracture of metallic structures and vessels) to the reinforcement steel used in the AARR-type of containment does not appear appropriate. This conclusion is based upon the fact that there are important distinctions between a reinforced concrete containment vessel and one constructed of welded steel plate. The most important distinctions between a welded steel plate and a reinforced concrete structure are:

- 1) the constraint or rigidity of reinforcing bars is slight, and the load is transferred from bar to bar through the concrete; and
- 2) there is virtually no possibility for extended brittle fracture propagation in reinforcing steel in concrete, as the load is resisted by hundreds or thousands of discrete bars, each bar independent of the others. While a failure of a single bar would cause a slight increase in the stresses in adjoining bars, the failure itself (a crack) does not induce a stress concentration in these adjoining bars. This, of course, is not true in a monolithic steel plate, where the crack may propagate very rapidly, once initiated.

If the structure consisted of a concrete frame containing only two or three bars, of course, the stress increase following a single bar failure would be significant and might lead to failure. Indeed, the only documented case we have found where brittle failure of reinforcing steel was thought to cause failure of the complete structure involved a frame which employed just a few bars.*

Although many thousands of reinforced concrete structures have been built in very cold climates, we have been unable to find any reports of failures due to brittle failure of the reinforcing steel, except in the one case cited above. We are endeavoring to probe deeper into that isolated failure, as the information we have is lacking in certain detail.

9. (a) *Is cathodic protection of the containment steel liner necessary and will it be provided? (CH)*

Answer:² At the present time, cathodic protection of the containment steel liner is not considered necessary. There will be a minimum of 18 in. of concrete protection between the backfill and the liner and this concrete will be damp-proofed. Since there is an elaborate system of sump drains around the containment building there will be little, if any, water in contact with the concrete and hence, essentially no constant water contact with the steel liner. However this problem will

* S. R. Davies, "Failure of Reinforced Concrete Due to Brittle Fracture of Reinforcement," Civil Engineering and Public Works Review, September, 1965.

receive further study during the Title-II phase of the design work to either confirm this initial position or to establish the need for a sacrificial anode system. This study will consider, among other factors, the free water in the exterior concrete walls.

(b) *With respect to galvanic action, is protection against galvanic cell action provided within the reactor vessel in view of the material (i.e., Be, Al, Fe) being used? (CH)*

Answer:² Corrosion and material compatibility tests with the AARR primary system materials have been performed at ORNL. These tests were conducted under simulated water conditions, material ratios, and material-water volume ratios expected in the AARR. The isothermal corrosion test conducted on crevice and galvanically-coupled specimens of aluminum, beryllium, and stainless steel was terminated in July, 1966. The samples had been exposed to deionized water at 200°F for 3655 hours. Although the examination has not been complete as yet, a preliminary review of the test results does not reveal any indication of galvanic cell action, and shows that the corrosion rate in the AARR would not exceed acceptable levels.

For further information refer to ANL Reactor Development Program Progress Reports: ANL-7176 (February 1966), ANL-7219 (May 1966), and ANL-7345 (July 1966).

10. *Criterion 11 states that no substantial pressure will be imposed on the structural materials unless the system temperatures are well above the NDT temperatures of the materials. Paragraph 11.2 states that this criterion will be adhered to except during hydrostatic testing. The basis for this exception should be provided. (CH)*

Answer:² Criterion 11.2 has been revised and now reads as follows (see also Part 1, Section II): The primary system will not be pressurized when the materials in the walls are at a temperature less than the nil-ductility transition temperature plus 60°F.

11. *Paragraph 11.4 states that the NDT properties of the steels used to construct the components of the primary system, containment liner and rebar (if necessary) will be determined by test. This statement should be clarified, if practicable, to indicate if these tests are made only at the time of fabrication or if tests will be conducted periodically on exposed representative metal specimens situated about and within the reactor vessel during the life of the primary system components. (CH)*

Answer:² As part of the AARR reactor vessel surveillance program, tensile and Charpy V-notch tests will be made on all materials used for fabrication of the reactor vessel. These tests will be performed at the time of fabrication and will include material from all nozzles, shell forgings or plate, top and bottom heads, weld metal and heat affected zones on welded joints. Similar tests will also be performed periodically on representative metal specimens which will have been exposed to

neutron irradiation within the vessel (see question 110, below, for surveillance program).

Components of the primary system other than the reactor vessel will have their NDT properties determined at the time of their fabrication (if required by choice of material). No further or periodic tests are planned, since no irradiation damage is expected.

The steel liner of the containment structure will have its NDT properties determined at the time of fabrication. No further or periodic tests are planned, since no irradiation damage is expected.

NDT properties of the reinforcing steel bars in the containment structure will not be determined as explained in questions 8 and 26(a).

12. Criterion 17:

A limit should be set on the energy potential of experiments placed in the reactor so as to assure containment in the event that this energy were also released to the RCB during a MCA. (CH)

Answer:³ See Part 1, Section II, Criterion 5.2.

13. Criterion 19:

The design of containment is to a leak rate of 0.1% per day or less. Some additional information should be presented to assure that this design value can be met. (CH)

Answer:⁴ The guaranteed leak rate criterion has been revised to 1%, by weight, of the contained mass per day at 7.5 psig. It is felt that this value can be achieved with a very high degree of confidence. The design objective is to have a leak rate no greater than 0.1% per day.

14. Criterion 20:

The penetration sealing materials should be selected to withstand the effect of MCA temperatures in the RCB without significant change in leaktightness. (CH)

Answer:^{2,3} The penetration sealing materials shall be selected with the clearly-defined objective of maintaining gastight integrity under severe MCA conditions. The selection of the seal material for each application will be determined during Title-II, with consideration of the following:

1. Environment
Define the liquid or gas environment of the seal element
2. Temperature
 - a. Normal temperature exposure
 - b. Maximum temperature exposure
 - c. Duration of maximum temperature exposure

4. Service Life

- a. Life expectancy under normal environmental conditions
- b. Life expectancy under worst environmental conditions

See also Part 1, Section II, Criterion 17.3.

15. Criteria 24-25:

It presently appears that provisions for full core handling requires definition of whether confinement or containment will be used. (CH)

Answer:² The design and arrangement of the fuel handling pool facilities has now been revised such that all refueling and core disassembly operations, as well as spent and defective core storage, are conducted within the Reactor Containment Building. After decay and after core disassembly into individual spent fuel elements, the individual elements are removed from the Reactor Building through the underwater hatch to the main canal in the Active Materials Handling Building, and loaded underwater into suitable shipping casks. The casks are then removed from the canal, decontaminated and shipped off-site by truck.

16. *The nearest boundary of the site is less than 2 km. What assurance is there that future residential encroachment will not occur? It may be appropriate to include maximum credible accident dosages at the site boundary along with possible future population densities. (CH)*

Answer:^{2,3} The estimated off-site population projection up to a radius of 2 km is 965 persons in 1980. All of these inhabitants are limited to the southwest and northwest quadrants as indicated in the response to question 86 below.

Maximum Credible Accident dosages at the site boundary are given in Part 1, Section VII-C-1. See also Part 1, Section III-B-2.

17. *Do the operating parameters represent the increase in inlet and outlet operating temperatures as presented by the AE (there appears to be a discrepancy with p. IV-A-2)? (RDT)*

Answer:² The operating parameters (inlet and outlet temperatures) presented by the AE represented the results of a "bare-bones" plant design study. That study was an attempt to reduce the facility cost and was not intended to be related to the PSAR as issued in April, 1966. The study was later abandoned. Therefore, no actual discrepancy existed between the PSAR and the AE.

Meanwhile, however, a Revised (September 28, 1966) Section IV has been written and submitted to the AEC. The new section contains a summary of the operating parameters in Table IV-A-1.

18. *Discuss the capability of the emergency system to keep the core covered by water following rupture of an emergency cooling system outlet line. (CH)*

Answer:² The question was directed toward the original emergency shutdown and cooling system and no longer applies to the revised emergency cooling provisions described in the Revised (September 28, 1966) Section IV-G. The original emergency system consisted of two independent emergency loops with separate inlet and outlet lines in the reactor vessel. The system was located in the Reactor Pool and devised to provide cooling even in the event of a rupture in the emergency cooling system outlet lines. However, since the inquiry, the primary system has been revised and the original emergency shutdown and cooling system has been replaced by a system utilizing the primary coolant loop. The new system utilizes the same components as the primary coolant loop, with the exception that motive power for the main circulation pumps is provided by pony motors mounted atop the main motors. Under emergency operation these pony motors, supplied with Class-I (uninterrupted) power, operate the pumps at reduced speed to remove decay heat from the reactor core.

19. (a) *What effect will the central wire spacers between fuel plates have on the results of the thermal analysis presented in Appendix B? (CH)*

Answer:⁴ This question is no longer applicable. The HFIR core, to be used in AARR, does not employ spacers between fuel plates.

- (b) *There is no clear discussion of the design criteria of the fuel, the fuel fabrication process, or the use of the different assemblies - A₁, A₂, A₃, A₄. (RDT)*

Answer:⁴ This question is no longer applicable. The HFIR core is now to be used identically in AARR, and ORNL-HFIR specifications will apply. The assembly designations mentioned have no significance in the HFIR core.

- (c) *What reason is there to believe that the manufacturing tolerance on channel spacing can be held to ± 1 mil, as estimated in Appendix B, using the brazing techniques to combine fuel plates to make fuel elements? (CH)*

Answer:⁴ This question is no longer applicable. See Part 1, Section IV-B-1 for a description of the HFIR core as fabricated and operated at design power at ORNL.

- (d) *What provisions are made for removal of failed fuel elements? (RDT)*

Answer:³ See Part 1, Section IV-C-3.

20. *What criterion will be used to determine whether or not beryllium reflector irradiation damage is excessive? What is the estimated period of time required at AARR conditions to result in excessive irradiation damage to the inner and outer reflectors? (CH)*

Answer:² The reflector for AARR with the HFIR core will now consist of three regions called the removable, semi-permanent and permanent reflectors.

Based on a review of work in the area of beryllium irradiation damage by R. L. Tromp of Battelle Northwest Laboratory and others, the criterion used for damage to the beryllium was taken as an accumulation of 10 cc He/cc Be, which appears to be conservative in relation to temperature, bowing and structural strength. Using this criterion, the absolute flux at the region interfaces, and the (n, 2n) and (n, α) cross sections corresponding to the energy groups used, the relative lifetimes of the various reflector pieces were calculated. These data are shown below.

<u>Region</u>	<u>Description</u>	<u>Estimated Operational Lifetime</u> <u>(Full Power Years)</u>
A	Removable Be	~ 0.6
B	Semi-permanent Be	~ 2.2
C	Permanent Be	~ 6.2

Until tests on reflector pieces irradiated in AARR show that a different criterion should be used, AARR reflector pieces will be replaced routinely when or before the above estimated lifetimes have been reached.

21. *Why does the difference in beryllium coolant hole size result in possible significant erosion in the outer reflector? (CH)*

Answer:² At the time of initial writing of the PSAR, work by B&W in support of the ATR^a indicated that the corrosion rate of beryllium was velocity dependent. ORNL data, however, indicate that, at ORNL test conditions, "within the velocity range of 13 to 81 fps, the average corrosion rate of Be was independent of velocity . . ." ^b Recent beryllium corrosion tests conducted at ORNL in support of AARR and at AARR (240 MW stainless steel core) conditions (44 fps through the beryllium) indicated a corrosion rate of ~ 2.8 mpy. ^c In view of the ORNL experience, the dependence of beryllium corrosion rate on velocity is no longer considered significant. Hence, the statement in the PSAR referred to in the question should be removed, and it does not appear in Part 1.

(a) IDO-24460
 (b) ORNL-TM-1030
 (c) Private Communication, J. Griess, ORNL

22. *In discussing the experimental facilities no mention is made of the exposure that the beam tubes receive. What considerations are given to replacing the tubes? (RDT)*

Answer:³ See Part 1, Section IV-E-1.

23. *In the first paragraph of Item 1, Incoming Power, a statement is made that Argonne has never suffered a loss of electrical service from the public utility. It is believed that this is no longer true due to the recent loss of power on April 23, 1966. This power loss should be taken into consideration when writing the final safety report. (CH)*

Answer:² On April 23, 1966 Argonne National Laboratory suffered its only total loss of utility electrical power. This failure occurred at 5:51 AM on a Saturday morning when the minimum number of maintenance personnel were on duty. It thereby provided a realistic test of the emergency electrical equipment and ANL preventive maintenance procedures.

Twenty-five emergency power generators are installed on the site and of these, sixteen have diesel-engine prime movers. All of the diesel-engines started and picked up their loads. One of these diesel-engines had to be shut down shortly after starting because of difficulty with its turbo-charger. All other emergency generators delivered the required power to their connected loads until utility power was restored one hour and forty-five minutes after power was lost.

24. *Are the injection pumps to be used in the emergency cooling loops? (CH)*

Answer:² The proper nomenclature for the "injection pumps" is the "pres-surization pumps." These pumps were not to be used for, or in, the former emergency cooling system.

25. *The second sentence in Paragraph 2 (Page IV-39) states as follows: "The generator engine or steam turbine governor is set to pick up the load automatically at each switchgear unit." This sentence should be rewritten for clarification. (CH)*

Answer:² Part 1, Section IV has incorporated the proposed change.

In the event of loss of normal power to either switchgear unit EA-1 or EA-2, the associated feeder circuit breaker trips and the bus tie circuit breaker closes. Simultaneously, the associated generator is started automatically and comes up to synchronous speed and rated voltage. Power is supplied to both switchgear sections from the energized substation, and the operating generator is available to supply power if needed. The generator would continue running as long as the operating personnel felt it was needed.

If utility power is lost to both switchgear units EA-1 and EA-2, the tie breaker does not close, or if closed it opens. Both prime movers start when the voltage drops to a predetermined level on their respective switchgear units. As each generator reaches rated speed and voltage, the generator breakers close automatically connecting each generator to its respective switchgear. The tie breaker remains open to eliminate the possibility of both units failing from a single fault. The tie breaker would be closed if one of the two machines failed to start.

The majority of the Class-II loads will be reconnected manually to the generator power supply. Certain loads such as the Lab and Office Building radioactive exhaust fans will be restarted automatically. As specific information on the operating requirements of the equipment becomes available, the most suitable method of reconnecting to the electrical supply will be used.

The circuit breakers supplying power to the Class-I busses will be closed automatically when the generator is up to rated speed and voltage.

26. (a) *We note that the Reactor Containment Building will be a reinforced concrete structure with a steel liner. Although the liner is not a primary strength member, it inevitably will be stressed when the containment building is pressurized. The material should be selected to provide adequate impact properties just as though it were a strength member, since a rip in the liner would lead to unacceptably large leakage.*

For the reinforced concrete structure, the situation is less certain. Reinforcing steel is generally produced to specification requirements, which in a plate material would be highly suspect. It has been stated that "brittle fracture is not a problem in reinforced concrete construction".

Although it is possible that brittle fracture is, in fact, no problem in reinforced concrete structures, evidence bearing on the question should be furnished. (RDT)

Answer:² At normal design conditions it is doubtful that a crack would propagate in the liner since the concrete, which is reinforced to function as the structural resistance to internal pressure, would constrain any radial deformation of the liner. It is possible, though, that a crack could develop at a location, say at a structural discontinuity (shell-to-base slab, for example) and propagate with only rotational deformation of the structural concrete.

We feel, however, that a much more important consideration exists, and that concerns the examination of the failure mode of this type of containment structure. At increasing pressures above the design conditions, the normally small cracks in the concrete will tend to open up. One of the functions of the modern deformed reinforcing steel bar is to distribute these cracks relatively uniformly along the length of the bar in a highly stressed region. Nevertheless, local straining of the liner is to be expected and stresses in the liner may, in fact, approach those

in the reinforcing steel. For this reason our design philosophy is to treat the liner as though it were a structural member. The result of this philosophy is to specify a liner material which meets the intent of Section III of the ASME Boiler and Pressure Vessel Code as related to Class B vessels (containment vessels). Specifically, we will require a liner material which will meet the NDT + 30°F requirement.

Regarding the reinforced concrete, it is clear that we are dealing with an entirely different structure than a conventional single-wall welded steel pressure vessel. Consider two methods used to prevent propagation of cracks in a brittle (or brittle-like) material, namely (1) drilling a hole at the end of a discovered crack, and (2) the use of riveted seams in an otherwise all-welded shell or vessel structure. The main function of these techniques is to stop the propagation of the crack by requiring that a new crack be initiated if the original failure is to continue. Since it requires far more energy to initiate the new crack than to propagate the original one, the initial crack is stopped unless a significant flaw existed at the exact point at which the hole was drilled or in the plate on the "other side" of the riveted seam.

The behavior of the multiple bars in a reinforced concrete slab or shell structure following a single bar failure (by cracking) is analogous to that of the riveted seam in the welded steel plate structure. In fact, since adjacent reinforcing bars are not in contact with each other, except through the concrete, and the two plates are firmly clamped by the rivets, the effectiveness of arresting of the crack propagation is more certain for the case of the reinforced concrete. In addition, there are hundreds, and even thousands of bars involved, and each bar may be considered as a "riveted seam."

The concern shown for possible brittle fracture in metallic vessels has had its roots in spectacular and disastrous failures. Intuitively, we know that failure of reinforced concrete due to the brittle fracture of the reinforcing bars is not a problem. There are thousands of exposed reinforced concrete structures constructed in cold climates which have performed satisfactorily. Additionally, examination of reported failures has not shown brittle fracture of the steel to be a cause in a single case.* (We are informed of one failure - believed to be unique by the engineer reporting the failure - in which the failure is stated to be due to brittle fracture of the bars. However, the structure concerned was a beam and column frame, containing only a few reinforcing bars, and thus the failure of just one bar could have caused a very significant increase in stress levels for the adjoining bars. See also the response to question 11).

We have also discussed this problem with personnel at the Institute of Gas Technology (IGT) at the Illinois Institute of Technology. We did this because IGT had recently completed a series of tests on a model of a prestressed concrete cryogenic vessel used for the storage of liquid nitrogen. Although both the prestressing cable and the ordinary reinforcing steel bars had NDT temperatures (as related to Charpy V-notch tests) near or even higher than normal room temperatures, the steel

* Jacob Feld, "Failures of Concrete Structures", Proceedings Am. Concrete Institute, Vol. 54, p. 449, December 1957.

performed completely satisfactorily at temperatures down to about -270°F . (The welded liner, however, was recognized as a potential problem, and was constructed of austenitic stainless steel.)

There is one potential source of concern in the design of a reinforced shell (or slab) which must be recognized, and that is the question of overlapping splices in many bars in one area. Good design practice dictates that splices should be staggered and strictly comply with code values for lap distances. The AARR containment structure will conform to good design practice.

(b) Item K; $1.2 \times 10^{-6} \text{ ft}^3$ should be $1.2 \times 10^6 \text{ ft}^3$.

Answer:⁴ The correction is appropriate. This is now covered in Part 1, Section IV-J.

27. (a) *What is the estimated leak rate of the atmosphere via the stack at design pressure with the stack isolation valves closed?*

(b) *What is the estimated leak rate with the valves open?*
(CH)

Answer:² a) With the stack isolation valves closed the leak rate is estimated to be a maximum of 0.85 lb/day and a minimum of 0.36 lb/day.

b) The estimated leak rate with a 7.5 psi overpressure and stack isolation valves open is 443 lb/sec.

Obviously, it would be impossible to maintain any overpressure in the building with those valves open. Consequently, the estimated leak rate has no practical significance.

28. *The allowable instantaneous stack radioactive effluent release limit should be provided. (CH)*

Answer:³ See Part 1, Section VI-E-2.

29. *The allowable continuous stack radioactive effluent release limit should be provided. (CH)*

Answer:³ See Part 1, Section VI-E-2.

30. *The limit on radioactive liquid waste discharge to the Lab Waste Treatment Plant should be provided. (CH)*

Answer:³ See Part 1, Section IV-M.

31. *Are the reactor pool and the various other canal pools cleaned by the same purification loop as indicated? (Figure VI-11) (CH)*

Answer:² The water in all the pool and canal compartments is continuously purified in a sweep flow demineralizer-filter unit. If a defective core is stored or disassembled in the defective core pool, the sweep-flow stream from this compartment is shut off in order to avoid contamination of the other compartments. The defective core pool has a separate cleanup loop for this event.

If needed because of contamination after transport of a spent core, a special high capacity demineralizer-filter unit can be substituted for the normal sweepflow purification circuit of the compartment. This unit will be used sequentially following the path of the core travel through the different compartments. Should further decontamination of the stainless steel clad pool walls become necessary, the water can be pumped from one compartment at a time to and from the pool and canal water storage tank.

32. *It is recommended that the title of Figure IV-19 as shown on the drawing and also on page vii of the List of Figures include the Electrical Equipment Building. (CH)*

Answer:³ See Part 1, Section IV, Figure IV-49.

33. *What is the estimated temperature coefficient of reactivity? (CH)*

Answer:³ See Part 1, Appendix A, Section A-VI.

34. (a) *The design core fuel reaches melting temperature with only 90 cents reactivity insertion. The engineered safeguards that are provided to cope with this very sensitive parameter should be explicitly described in association with transient analysis. (CH)*

Answer:⁴ With the change to the HFIR core, the 90 cent reactivity insertion is no longer applicable. However, for a discussion of safeguards provided to cope with reactivity insertions, see Part 1, Section VI-B-7.

(b) *Discuss the effect of the spacer between fuel plates on the hot spot in the skewed hot channel. (CH)*

Answer:⁴ This question is no longer applicable. The HFIR core, to be used in AARR, does not employ spacers between fuel plates.

35. *Have any studies been made of the transient response of the reactor for step and ramp reactivity inputs at initial power less than 1 Mw? What influence does an initial power of less than 1 Mw have on the peak power and the time required to attain peak power? (CH)*

Answer:³ See Part 1, Appendix C, Section C-IV-C.

36. *If the best current estimate of the net Doppler coefficient of reactivity for the fuel is zero, it is not clear why negative feedback was assumed from the Doppler effect in the bulk of the transient analysis. What effect does this assumption have on the transient analysis with respect to the attainment of burnout and the rapidity of reaching burnout? (CH)*

Answer:⁴ With the relatively lightly loaded HFIR core, it is estimated that the net Doppler broadening effects of the uranium isotopes provide a negligible reactivity-feedback contribution in reactor excursions, in comparison with the negative reactivity feedback effects of thermal expansion of the fuel plates and fuel zone coolant.

In the case of the earlier reference core, with ~60 kg U²³⁵, the reactivity coefficient of Doppler broadening of fuel isotopes might be positive. Argonne's best estimate is that the net Doppler coefficient of reactivity for that fuel is small. The analysis of specific accidents was predicated on a zero Doppler coefficient.

37. *Please clarify the statement, "If the core is restrained from expanding longitudinally, then added negative feedback would be produced." (CH)*

Answer:³ See Part 1, Appendix C, Section C-II.

38. *What is the maximum reactivity insertion within a short time period (33 msec or less) that can be credibly envisioned for AARR? (Fig. IV-B-5) (CH)*

Answer:³ See Part 1, Section VI-B-1.

39. *It is mentioned in various places in the report that fuel ruptures would be detected by the fission-product monitors; however, the report provides no information on these monitors - e.g., number, type, and location. Provide information on the fission-product monitors to be used in AARR. (CH)*

Answer:² The detection of fuel element failure is based on the measurement of fission-product activity in the primary coolant. The method used involves detecting the gamma rays from fission-product iodine present in the cooling water as extracted in an anion-exchange column.

A single channel is provided for the detection of small quantities of fission products in the primary coolant. The primary coolant is sampled immediately after it leaves the reactor and the sample is held up for approximately two minutes to allow decay of N-16 activity. It is next passed through a filter to remove certain activated particulates, and then through a cation-exchange column where the cations, consisting of corrosion products and certain fission products, are removed. The remaining active elements (anions) are collected in the anion-exchange column where they are monitored by a gamma-sensitive scintillation detector. By pulse height discrimination various low-energy activities

are biased out, confining the measured output to fission product iodine activity alone.

Two back-up means for detection of fuel element failures are provided. One consists of a channel that monitors the effluent gas from the primary coolant degasifier. A large release of fission product gases into the primary coolant would be detected by this instrument as a sudden increase in gross gamma activity. This channel has a lower sensitivity than the primary coolant sampler channel. In addition, three redundant channels that monitor gross gamma activity in the reactor vessel outlet stream are provided. These are sensitive only to the large increase in activity accompanying a gross cladding failure. They respond quickly, however, and are used for automatic reactor scram.

40. (a) *Is there any apparent reason for the difference between measured and calculated values of maximum worth of ITC void? (CH)*

Answer:⁴ In earlier analyses, the calculated maximum positive reactivity effect of partial voidings of water in the Internal Thermal Column was smaller than the experimentally determined reactivity gain. As stated in the original version of the PSAR (issue of March 31, 1966, Appendix A, page A-26), it is believed that this was principally due to the use of one-dimensional analysis, in that the positive reactivity contribution of an increased worth of the axial reflectors was not included.

Since the issuance of the earlier version of the PSAR, two-dimensional diffusion theory calculations have been performed to study this effect. The calculated reactivity gain, corresponding to a 50% void in the 100%-H₂O ITC of a Criticality Facility loading (stainless steel-uranium core), is 1.50%, in excellent agreement with the measured value of 1.49% for a 49% void. It should be noted that the ITC of this particular Criticality Facility loading was slightly smaller than the ITC of the reference system. Nevertheless, it is apparent that inclusion of the effect of an increased worth of the axial reflectors adequately reconciles the difference between theory and measurement, and it is expected that two-dimensional calculations also would provide good data of ITC void coefficients for the reference ITC and the HFIR core.

It is important to note that, both in the earlier version and in Part 1 reference is made to the measured reactivity effect of voiding. The experimental value of a maximum reactivity gain of 4-1/2 dollars was used in the accident analyses, not the smaller, calculated value.

(b) *Consideration should also be given to the consequences of an accident in which voids in the core channels are originally present and are then suddenly lost. (CH)*

Answer:⁴ Such accidents have not been considered in the current PSAR. Mechanisms by which such accidents could occur will be studied during Title-II design; if a credible mechanism can be found, suitable analysis will be included in the Final Safety Analysis Report.

41. (a) *Discuss the effects of dropping the reactor vessel head from the crane into the canal structure. Has the design included consideration of such an accident? (CH)*

Answer:² The removal of the complete flat head of the pressure vessel is not anticipated as a normal operation, and, therefore, the overhead polar crane is not sized for this operation. The head weighs about 25 tons, whereas the maximum capacity of the crane is 15 tons. The inner head, or breech plug, which is removed with each core change, will be removed from the pressure vessel, moved laterally, and set down within a short distance of the pressure vessel. It is not required to be taken out of the Reactor Pool.

During the Title-II design stage it may be deemed expedient to size the polar crane to enable it to pick up the head of the pressure vessel. If this is realized, the travel of the polar crane bridge will be mechanically limited such as to preclude the crane hook from physically being over the Spent Core Storage Pool or the defective core storage pool.

It is not considered possible that the stainless-steel-lined pool structure, which approximates 8 ft of concrete in wall thickness below the level of the top of the pressure vessel and 6 ft in the pool floor can be breached by dropping the reactor vessel head.

- (b) *The report rules out fuel handling accidents in the storage and transport canals on the basis that such accidents are not deemed credible. However, in view of the proposed absence of containment in these areas, we believe that the consequences of the following accidents should be analyzed:*

- (1) *A fresh core is dropped by the core handling mechanism in such a manner as to result in the removal of all control rods from the core. (CH)*

Answer:⁴ With the HFIR core, the two concentric fuel elements are transferred separately, with control rods remaining in the reactor vessel. The question is therefore inapplicable to the current design.

- (2) *A spent core is dropped by the handling machine to the bottom of the pool and the core comes to rest on its side. (CH)*

Answer:³ See Part 1, Section VI-C-2.

- (3) *With the maximum expected inventory of spent cores present in the storage pool, an uncontrolled water leak develops in the pool and results in the uncovering of the spent cores. (CH)*

Answer:³ See Part 1, Section VI-C-3.

(Note: In the current design, all fuel handling operations, including storage during decay, prior to shipment of decayed spent elements, are conducted within the Reactor Containment Building.)

42. *If there is available data on fission product retention during transport through deep pools of water it should be included. (CH)*

Answer:⁴ Some credible accidents in AARR involve the transport of fission products through water, and possible retention therein. Such retention would undoubtedly reduce the consequences of accidental release, particularly with regard to iodine doses to the thyroid. However, our Maximum Credible Accident analysis (see Part 1, Section VII) demonstrates that AARR can be constructed and safely operated at the proposed site even with no credit taken for any fission product retention in water pools. Therefore, no data on this effect are included in the report.

43. *Could storage and transport canal water be lost as a result of an uncontrolled leak from the reactor pool - i.e., does reactor pool water communicate with storage and transport canal water in any manner? (CH)*

Answer:² The Reactor Pool water communicates with the other compartments through separation gates. These gates are designed to maintain separation with the full static water load on one side only. All other communication between various compartments is made via surface lines. No bottom penetrations are provided.

Water from the transport canal can only be lost as a result of uncontrolled leakage from the Reactor Pool if the separation gate is left open. During operation of the reactor the gate between the Transfer Canal and Reactor Pool is normally closed. It is opened only during fuel transfer operations.

Uncovering of the stored cores in the Spent Core Storage Pool is highly improbable even if the Transfer Canal or the Reactor Pool are breached at their lowest level. The Spent Core Storage Pool is located below the Transfer Canal and separated from the Reactor Pool by a fixed concrete wall. The spent cores are stored in a pit below the Transfer Canal and there is no direct horizontal communication between the pit and the Reactor Pool. Cooling of the spent cores can be continued even if both the Transfer Canal and Reactor Pool are drained.

44. *The statements concerning the retention of containment and control capability under tornado conditions are presented as assumptions. Are these statements intended as assumptions, or are they expected consequences of the AARR design? (Page VI-G-2) (CH)*

Answer:^{2,3} The referenced statements represent the maximum consequences which could result from a direct tornado strike at the AARR facility, taking into account the tornado-protective features included in the plant design. The containment building and control room are designed to withstand tornadoes, and the cooling tower basin is entirely below grade and designed to minimize water loss in a tornado. Therefore the statements made with regard to these structures remaining undamaged are the results of design, and are not assumptions. On the other hand, the

other structures mentioned are not being designed for tornado resistance, so that the damage described for these structures represents an assumed maximum. The analysis then goes on to describe how the plant can be safely shut down and decay heat removed even though this assumed maximum damage has occurred.

See also a revised analysis of the consequences of an earthquake or tornado strike on the AARR facility, presented in Part 1, Section VI-F-2.

45. *What portions of the core are included in regions 1, 2, and 3? (CH)*

Answer:⁴ This question is no longer applicable. Regions 1, 2 and 3 have no significance in the HFIR core.

46. *The basis for the conclusion that a localized hydrogen detonation would not damage the RCB walls is not clear. Please elaborate upon the last sentence at the top of Page VI-H-6. (CH)*

Answer:³ The question of hydrogen detonation has been reanalyzed. See Part 1, Section VIII-D.

47. *There appears to be a significant discrepancy between the assumptions used in the determination of the RCB design overpressure required to assure containment in the event of a MCA and the analysis results presented in Appendix F. The assumption used in the design overpressure calculation (with the reactor operating at 100 Mw and no scram, all primary system water is released to the RCB at 220°F within 1 minute after the occurrence of a major rupture), results in an instantaneous rise in RCB pressure due to the flashing of a small fraction of the primary system inventory. However, the results given in Appendix F indicate that a much larger fraction of primary water would be expected to flash to steam under the postulated accident conditions. On the basis of Appendix F information, it would appear that the required design overpressure could be of the order of several atmospheres rather than 9 psi as stated in the report. Additional information should be provided to resolve this apparent discrepancy. (CH)*

Answer:^{3,4} The information presented in Appendix F of the original version of the PSAR (issue of March 31, 1966) is no longer applicable, since the primary system design has been changed materially and the primary coolant temperature level has been reduced. In the current design, the primary water temperature is 120°F at the reactor vessel inlet, and 156°F at the reactor vessel outlet. The average temperature of the water leaving the reactor core (at 100 MW) is 164°F. The only water in the entire system normally at a temperature higher than the atmospheric boiling point is a very small amount in the hottest channels of the core near the core exit; the maximum water temperature anticipated in this small region is 226°F. Under these circumstances, it has been assumed in the revised MCA analysis presented in Part 1, (see Section VII-B-1 and Section VII-B-2) that there is no significant flashing of primary coolant at a rupture, during the rupture accident.

48. *If heat losses to the surroundings are of significant importance in the overpressure calculations, it would appear that the direction (sign) of the temperature gradient between the containment air temperature and the constant surroundings temperature is nonconservative. What is the basis for the selection of the temperatures given in Table VI-J-1? (CH)*

Answer:³ See Part 1, Section VII-B-2c.

49. *What is the basis for the selected heat transfer coefficients given in Table VI-J-1? What is Reference 1? (CH)*

Answer:³ See Part 1, Section VII-B-2c.

50. *The calculation should be provided for showing that the range of a missile through reactor vessel water would be very short. (CH)*

Answer:³ See Part 1, Section VI-G-1a.

51. *With regard to the calculated direct gamma-ray dose rates, it is stated in the report that the design of the RCB roof will be such as to minimize the radiation dose due to "skyshine", since this effect was neglected in the calculations. A more practical approach might be to include the effects of "skyshine" in the buildup factor as was done in the HFBR FSAR (BNL-7661, Vol. II, pages 14-36). (CH)*

Answer:⁴ This is not so much a question as it is a suggestion of a method of analysis. This method will be reconsidered in the course of preparation of the Final Safety Analysis Report. It is intended that the design of the dome of the Reactor Containment Building will be such that "skyshine" will contribute only a small additional radiation dose. A calculation has been made of the MCA whole-body dose rate, due to skyshine, to personnel in the Laboratory and Office Building, which verifies that this is small in comparison with the direct dose from the Reactor Containment Building as per its Title-I design.

52. (a) *Pu²³⁹ and Pu²⁴⁰ doses resulting from an accident release should be considered and included in the same manner as the mixed fission product effects. (CH)*

Answer:³ See Part 1, Section VII-C-4.

- (b) *Are Sr⁸⁹ and Sr⁹⁰ the worst exposure cases calculated? (CH)*

Answer:^{3,4} In addition to the bone-dose contributions from inhalation of Sr⁸⁹ and Sr⁹⁰, inhalation-doses from other bone-seekers (e.g. Ce¹⁴⁴ and Ce^{144m}) and from halogens are included. The AISITE II report*

* R. A. Blaine and E. L. Bramblett, "AISITE II, A Digital Computer Program for Investigation of Reactor Siting," NAA-SR-9982, (Oct., 1964).

tabulates the isotopes which contribute to the bone doses; the AISITE II program was used to calculate these doses. The results show that the isotopes which produce the worst exposures are the iodines. See Part 1, Section VII-C-1.

(c) *The fission product inventory released through rabbit tubes following the onset of the accident may be a significant incremental addition to the source term for atmospheric dispersion. (CH)*

Answer:^{3,4} The design of the rabbit systems is such that no fission products originating in the reactor core can escape directly from the Containment Building via a rabbit system. See Part 1, Section IV-E-4.

53. *In discussing reactivity compensating mechanisms and reactivity margins no mention is made of the reactivity tied up in the burnable poison and its effect during burnup. (RDT)*

Answer:^{3,4} In the HFIR-type fuel zone, there will be a total of 2.8 gm of B¹⁰, controlling 0.05 Δk in the fresh reactor. This is discussed in Part 1, Appendix A, Section A-IV. Oak Ridge National Laboratory has calculated that in the HFIR, operating at 100 MW, the reactivity controlled by boron will drop from 0.05 Δk to 0.037 Δk after 2 days of operation. The calculated reactivity controlled by boron is 0.009 Δk after 14 days of full-power operation.*

54. *What is effect on peak fuel temperatures (Fig. VI-C-2 and 4 shows average fuel temp.) during transients? (RDT)*

Answer:³ See Part 1, Section VI-B-1.

55. *The subject of reactor reactivity is not treated in a clear, concise manner. There is no tabulation of reactivity values or coefficients. These values are widely scattered throughout the report. It is quite difficult to obtain an overall picture, or to verify such items as shutdown margins. For example, the 12 control rods are said to be worth 15% delta K/K (average value about 1-1/4%) and that 6% delta K/K would be controlled by burnable poisons. If the control rod of maximum worth is withdrawn, the total available negative reactivity would be less than 20% delta K/K. Although the cold, clean excess reactivity is not stated, it would be greater than the 17% delta K/K available in a hot unpoisoned core (Page A-21). Therefore, it does not seem possible to verify the statement on Page VI-D-3 that the actual value of the shutdown margin is estimated to be at least 3% with only 11 of the 12 control rods (fully) inserted. It seems that the shutdown margin may be near the 1% criterion (AARR 8.2).*

Because there doesn't seem to be sufficient information to verify the "worst case" condition, and since the use of chemical shim is not being considered during normal operation, the 1% shutdown margin with one rod withdrawn should be supported by further analysis and/or detailed discussion. This type of information should be provided:

* F. T. Binford and E. N. Cramer, "The High-Flux Isotope Reactor - A Functional Description," ORNL-3572 (Volume 1) (May 1964 - revised March 1965), Table 7.3.5.

- A. Total excess reactivity in core in most reactive configurations, normal and maximum credible.
- B. A tabulation of the reactivity involved with fission product poisons, fuel depletion, the various reactivity coefficients, burnable and/or soluble poisons, and rod worths.
- C. Minimum shutdown margin requirement:
 - (1) normal, hot, most reactive core,
 - (2) normal, cold, most reactive core,
 - (3) most reactive credible core.

Statements regarding shutdown margins are not clear and in certain areas, where additional margins will be supplied, the reactivity worths are not identified. During transport of a spent core or a fresh core, it is important to understand the source worth of various areas through which a core must pass as well as moderator worth of the environment in comparison with the shutdown margin built into the core. (RDT)

Answer:^{3,4} In part, this question, for the stainless steel-UO₂ system, is not relevant for the present AARR, with HFIR-type fuel. Reactivity tabulations are supplied in Part 1, Appendix A, Section A-IV.

56. Will the procedures and capabilities outlined in the PSAR be adequate and appropriate for dealing with a core in which significant clad failure has occurred? (RDT)

Answer:³ See Part 1, Section IV-C-3.

57. The core handling concept as shown pictorially in the PSAR does not provide positive positioning capability in transit. A more direct mechanical system, such as lead screws mounted to the Reactor Pool Personnel and Tooling Bridge would contribute rigidity plus a platform from which reactivity monitoring and additional shutdown margin can be supported. Such a platform may also be of value in providing a reference base for equipment to enhance coolant circulation if required. Such a system is described in the Preliminary Outline Specification for AARR by Burns & Roe dated Feb. 25, 1966. The inconsistency between the concepts reported should be resolved.

Considerable experience exists in handling individual core components by the method outlined. However, handling the complete core with control rods in place is not so common. It must be recognized that in the handling concept proposed there are numerous opportunities for error, some of which have heavy safety overtones without benefit of primary containment backup. Positive mechanical controls (in addition to administrative controls) must be designed into the system to preclude these opportunities for error. The refueling concept should be reviewed for opportunities to eliminate "hand-tool" work in the preparation of the exposed core for removal. (RDT)

Answer:⁴ With the HFIR core, the two concentric fuel elements are transferred separately, with control rods remaining in the reactor vessel. The fuel handling system will duplicate that in use in the HFIR as far as is practical in view of differences in pool design, headroom, etc. However, the use of more positive positioning mechanisms will be considered during Title-II design.

58. (a) *The general description of the control rod drives covers only the rack and pinion type of mechanism; and does not include a description of the regulating rod drive mechanism (mentioned as page D-29). The regulating rod mechanism is currently planned as a ball-nut and lead screw design but discussion of the regulating rod worth, motion requirements, core position, or design layout was not found. This information should be presented. (RDT)*

(b) *It appears that the detail of the control rods is still under investigation. The influence of the control rod detail on the control mechanism is not discussed. (RDT)*

Answer:^{3,4} In the current design, the HFIR control rod (plate) and the HFIR control rod (plate) drive mechanism designs will be used identically in AARR. See Part 1, Section IV-B-2 and Section IV-B-4, for descriptions of these designs.

59. *The calculated 110 psi core pressure loss does not include the entrance and exit pressure losses associated with grid and/or hold-down plates and core end fittings. What effect will these additional pressure losses have on the thermal and hydraulic analysis? (CH)*

Answer:⁴ Operating experience at HFIR indicates that the actual pressure loss is about 110-115 psi for a flow rate of ~13,000 gpm through the fuel regions. In a core with a large frictional pressure loss as in the AARR-HFIR core, the effect of hold-down plates and core end fittings is small because the resistance of these items to flow is kept low by design. With the HFIR core these losses are included in the pressure losses indicated and will not affect the results of the analyses presented. To illustrate this, an estimate was made of the effect a 10 psi loss over and above the design pressure loss of fittings would have on the flow in an individual channel. It would result in a decrease of about 4 to 5% in flow.

60. *Our calculations using equation B-5 give a pressure drop of 90 psia as compared to the 110 psia value given in Table B-VII. What values were used for the constants in this equation? (CH)*

Answer:⁴ This question is academic in view of the HFIR operating data now available.

61. *Some discussion should be included to justify the approach taken to allow for heating effect on the friction loss.*
(CH)

Answer:⁴ This question is academic in view of the HFIR operating data now available.

62. *The various factors used to modify the equations used in determining the temperature drops should be described more fully. The values assigned to various components of these factors should be justified.* (CH)

Answer:⁴ This question is no longer applicable. The ORNL steady state heat transfer analysis of the HFIR core* is used as the basis for this report.

63. *The as-built hot channel spacing is stated to be 40 mils throughout most of the report; however, elsewhere this value is given as 50 mils. (Pages A-5, B-33). The thermal analysis assumes a uniform 10 mil reduction in hot channel spacing, from 50 to 40 mils. (Page B-12). If this value has not yet been set at 50 mils, then the more conservative reduction in spacing (40 to 30 mils) should be used in the thermal analysis. To what extent does the more conservative reduction in hot channel spacing affect the anticipated thermal and hydraulic performance of the system?*
(CH)

Answer:³ See Part 1, Section IV-B-1.

64. *Shouldn't the exponent for the (s/s_0) term in eq. (B-4) be 0.44 rather than 0.24? Was the 0.24 exponent term in this equation used in the thermal analysis? The lower value is somewhat non-conservative in that its use results in an overestimate of the heat transfer coefficient.* (CH)

Answer:⁴ See response to question 62 above.

65. *With reference to the channel spacing effect, it is not clear that the assumption of a uniform 10 mil decrease in hot channel spacing is conservative. In particular, what is the estimated magnitude of the effects of thermal expansion, hydraulic loading (deflection), and scale formation on hot channel spacing?*
(CH)

Answer:³ See Part 1, Section IV-B-1.

* F. T. Binford and E. N. Cramer, "The High Flux Isotope Reactor - A Functional Description," ORNL-3572 (Volume 1) (May 1964 - Revised March 1965), p. 7-14.

66. *The factors F_q and F_Q loc do not make allowances for uncertainty in measurement or calculation of power density. The F_w factor does not include allowance for difference in velocity distribution in the various channels and velocity variation within a single channel. (CH)*

Answer:⁴ See response to question 62 above.

67. *With regard to the expected 3% expansion due to fuel burnup, is this diametral or radial expansion? (CH)*

What are the uncertainties with regard to swelling? What is the limit of swelling that can be tolerated, especially since swelling data were not obtained with boron loading in the fuel? (RDT)

Answer:⁴ No actual measurements have been made on HFIR fuel plates to determine the amount of fuel swelling. However, operating experience at burnups up to 2100 MWd at 100 MW has demonstrated that fuel swelling has not been a problem. HFIR has experienced small total flow rate changes during core life, and these changes have been attributed to one or a combination of changes in oxide thickness, surface roughness and fuel swelling. Calculations show that if the flow change were due only to fuel swelling, the average swelling would amount to about 1%. If the flow change were due only to oxide film buildup, the corresponding buildup would be about 1/2 mil. In each core run thus far at HFIR, the flow rate has increased for the first 1,000 MWd; thereafter, depending upon the pre-treatment given the fuel elements prior to operation, the flow has either remained constant or has decreased as much as 4% by the end of the 20-day core life. A 10% decrease of flow can be tolerated.

Some information concerning fuel swelling in the HFIR core can also be derived from IDO-17157.* This report presents preliminary results of irradiation tests on HFIR-type fuel materials. For average burnup and temperature, the amount of swelling which can be postulated from these test results lies between 1/2 and 1%. At the hot spot in the core, the postulated swelling is less than 5%.

68. *The information which provides the basis for the exclusion of a flux peaking factor should be provided. (CH)*

Answer:⁴ See response to question 62 above. The ORNL analysis of the HFIR core includes a flux peaking factor.

69. *Was a factor required to account for channel spacing variation omitted in equation B-13 or was this factor included in the mass flow rate term? The identifying subscript was omitted. (CH)*

Answer:⁴ See responses to questions 62 and 65 above.

* V. A. Walker, M. J. Brader, and G. W. Gibson, "ATR Fuel Materials Development Irradiation Results - Part II," IDO-17157 (June 1966).

70. *Europium oxide is indicated to be the control material. Is this definite; if so how does this affect the R&D program? (RDT)*

Answer:^{3,4} The HFIR control plate design, which includes europium oxide as the absorber material in the "black" sections, will be used identically in AARR (see Part 1, Section IV-B-2). Since such plates have been produced and are in use in HFIR, no R&D work on control plates is deemed necessary.

71. *It is stated that the estimate of the time required before flow reversal may be safely accomplished is valid only if the maximum surface temperature is below 300°F during the reversing process. In this respect, what is the corresponding temperature expected to be in the case of AARR? (CH)*

Answer:³ See Part 1, Section VI-C-2.

72. *It is not clear how the information plotted in Figure B-4 was derived. Also, it is not clear, from the information presented on Page B-6, as to how the dashed lines are used to determine the power increase factors. (CH)*

Answer:⁴ See response to question 62 above.

73. *Are the values for the parameters h_f and h_{fos} given on Page C-16 correct or are the numerical values f_o reversed? (CH)*

Answer:⁴ This question is no longer applicable with the change to the HFIR core.

74. *With reference to the Mark I steady-state heat transfer tests (Page D-4), should tests be made to determine the effect of the spacers since they were not considered to be present in the thermal and hydraulic analysis performed in Appendix B? (CH) Are the limits of the uncertainty adequate for 100 MW operation? (RDT)*

Answer:^{3,4} With the change to the HFIR core, most of the experimental heat transfer program previously planned has been dropped. No spacers are employed in the HFIR core. The major emphasis in the remaining heat transfer work is on transient tests. See Part 1, Appendix D, Section D-II.

75. *As noted in paragraph D-IIIA, page D-37, the design of the top head and beam tube arrangements are potentially critical, and difficult to analyze, in the case of the beam tubes. It appears that the analysis might be impossible on the basis of present knowledge. Will some experimental verification be necessary? (RDT)*

Answer:² Argonne has contracted with Combustion Engineering, Inc. to perform the engineering work on the reactor vessel. The vendor is presently working on the preliminary design phase of the contract, which includes a determination of whether experimental stress analysis is required in any area of the vessel in order to design the vessel. Of special concern in this study, of course, is the shell mid-section containing the horizontal beam tube nozzles.

At this time the vendor's work has not progressed far enough to state definitely whether experimental stress analysis will be required. However, there will be experimental verification of the vendor's design which will be performed during the hydrostatic testing of the vessel. At that time the stress intensities in the vessel shell ligaments between the horizontal beam and through tube nozzles will be determined by means of strain gages. Measurements will be executed in duplicate in not less than three ligaments of highest (calculated) stress. In addition, vessel shell stress intensities around the circumference of that nozzle connection contained in the region of highest (calculated) ligament stress will also be determined by strain gage measurements.

76. *It is not clear why the average peripheral power density in the hot cells was taken to be $2.22 \times 1.3 \text{ Mwt/liter} = 2.9 \text{ Mwt/liter}$ in the ARGUS calculations rather than 3.9 Mwt/liter as given in Table B-VII. (CH)*

Answer:⁴ This question is no longer applicable. No ARGUS calculations have been made for the HFIR core.

77. *Isn't a hot channel coolant velocity of 41.67 ft/sec slightly higher than would be indicated by use of the equation on Page B-6? (CH)*

Answer:⁴ This question is no longer applicable. No ARGUS calculations have been made for the HFIR core.

78. *The calculational model used to determine depressurization times following rupture of a primary coolant line should, if possible, be checked against available experimental data, e.g., see Trans. ANS Vol. 8, No. 1, Pg. 298. (ANL Note: Volume 9 is apparently intended) (CH)*

Answer:² The experimental parameters given in the reference specify a system in which the coolant temperatures throughout the loop exceed the saturation temperature at atmospheric pressure. A primary system rupture under these conditions results in complex two phase flow phenomena, immediately at the rupture location and shortly thereafter throughout the system. This data therefore cannot be used to test the AARR analytical model which is appropriate where the hot and cold leg temperatures are only 165°F and 120°F, respectively, and only liquid passes through the rupture.

In the revised system using the HFIR core the surge tank has been eliminated. Therefore, Appendix F no longer applies. See Part 1, Section VI-D.

79. *With regard to decay heat, has any consideration been given to residual fission power generated by multiplied neutrons originating from the neutron reaction in the beryllium reflectors: (CH)*

Answer:³ See Part 1, Appendix C, Section C-II.

80. *The 0.7 Mev value used for the effective gamma-ray energy (E) in Eq. H-3 appears to be low, at least for the two hour dose rates. USNRDL-TR-187 indicates that the effective gamma-ray energy is greater than the assumed value until approximately eight hours have elapsed. A value of 1.1 or 1.0 Mev might be more appropriately used in calculating the shorter time period exposure rates. In this regard note that (E) occurs in the numerator of Eq. 8.7 of AECU-3066 (Meteorology and Atomic Energy) - the 0.7 Mev constant in the denominator is a normalizing factor. (CH)*

Answer:^{3,4} The AISITE-II program has been used to calculate the gamma dose rate from the radiation plume, for the revised tabulations in Part 1, Section VII-C-1. The AISITE-II calculations are based on the formulas of WASH-740.* In any event, the whole-body doses are very small in comparison with the inhalation doses to the thyroid from the halogens.

81. (a) *For distances (X) downwind of 7200 meters, do the 30 day dose results given in Table H-IX include the exposure received during the first two hours after the MCA? (CH)*

Answer:^{3,4} All doses listed in Part 1, Section VII are accumulated doses, unless otherwise indicated. In particular, the 30-day doses tabulated include the doses received during the first two hours after the postulated MCA.

- (b) *The first sentence in the paragraph at the bottom of Page H-12 is not consistent with the statements made in the second paragraph of the same page. Please explain. (CH)*

Answer:⁴ This question results from a typographical error; the word "after" was inadvertently omitted from the first sentence of the 4th paragraph on page H-12. The sentence should have read: "It is assumed that after the first 2 hours the plume travels downwind..."

*"Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants," WASH-740 (March 1957).

82. *Some consideration should be given to the need for a hydrogen detector in series with the primary system degasifier to assure that hydrogen concentrations in the primary system do not exceed a pre-determined level. (RDT)*

Answer:² There is some uncertainty as to whether this question was prompted by a concern about high levels of dissolved hydrogen in the reactor water or about explosive mixtures in the off-gas line, but in either case we contemplate no need for a hydrogen detector in the primary degasifier line.

Based on HFIR and ETR operating experience the level of dissolved hydrogen in AARR is expected to be about 5 ppm or less. There is no reason to expect rapid changes in this parameter during normal operation. Water samples will, of course, be taken to monitor the level of all dissolved gases during low power runs on approach to full power for the first time and in this manner any unexpectedly high values will be noted before they become a hazard.

If, on the other hand, the question indicates concern that explosive mixtures of hydrogen and oxygen might be present in the off-gas line it should be noted that we intend to dilute the off-gas stream as it leaves the vacuum pumps with sufficient air to prevent the formation of explosive mixtures. The quantity of air required will be based on the concentration of dissolved hydrogen in the reactor water and the flow rate through the degasifier. A flow alarm on the dilution air line will indicate any drop in flow rate below the minimum amount determined by this calculation.

83. *No mention has been made of emergency or evacuation plans in the event of a major accident. Considering that several other reactors are located within a 600 meter radius of the AARR, the need for a coordinated evacuation plan, which would not route on-site personnel near the plant experiencing emergency conditions, becomes evident. (RDT)*

Answer:² In the event of a significant fission product release into the Reactor Containment Building (RCB), it may be necessary to evacuate AARR personnel and other laboratory personnel. Although detailed evacuation plans have not been formulated as yet, the following represents the current thinking which will be appropriately implemented during the Title-II design phase and the drafting of the operating procedures, and which will be discussed in detail in the Final Safety Analysis Report.

Whenever the radiation level of the RCB effluent air exceeds a predetermined level, the RCB will be isolated, with isolation resulting in the sounding of an evacuation alarm. Personnel inside the RCB will have been instructed to proceed as rapidly as possible to one of the three personnel airlocks (two normal and one emergency), whichever is nearest, for egress from the RCB. Each airlock will have a capacity of about fifteen people. It is estimated that the maximum number of experimenters

in the RCB at any time will not exceed twenty, with an average of about ten. By allowing a 50% increase for supporting and operating personnel, the total number of RCB evacuees would probably not exceed thirty. These people could easily be accommodated at one time in the two airlocks on the experiment floor which are located approximately 180° apart, even though some of those thirty personnel would very likely be on the pool wall level where the third airlock is located. Visitor control will be effected to the extent that a speedy, orderly RCB evacuation will not be compromised.

Laboratory emergency procedures require that the Area Emergency Supervisor or his alternate(s) evaluate the situation quickly, direct the evacuation of personnel, and report the situation to the Laboratory Emergency Coordinator via the dialing of telephone number "13". The Area Emergency Supervisor and his alternate(s) will be appointed from the operating staff of the AARR, so that the presence of at least one individual knowledgeable in emergency procedures can be assured at any time.

The Laboratory Emergency Coordinator is responsible for directing and coordinating all emergency activities on the site. He decides whether the site or portions thereof should be evacuated and initiates the necessary orders for any required emergency procedures. Lab-wide communications are effected through telephone, radio, public address and alarm systems. If site-wide evacuation is necessary, the Emergency Coordinator will determine the evacuation route(s). Laboratory personnel will be evacuated in accordance with the Shelter Relocation Plan and/or Site Relocation Plan, which are implemented by the particular Area Emergency Supervisor and his staff. These plans and site map are attached hereto as they appear in the Laboratory Emergency Handbook. The AARR site and facility outline are shown adjacent to Building 330 (CP-5), approximately 200 m north of Bluff Road and 300 m west of Meridian Road.

It should be emphasized that certain evacuation and recovery details are formulated at the time of the incident, since the course of any accident cannot be predicted in advance. Great reliance is placed upon all Laboratory personnel charged with carrying out emergency procedures to evaluate the situation and make decisions based on all available information. In this sense, it is not possible now, nor will it ever be, to specify exact site evacuation routes, rendezvous areas, etc., in advance.

SITE RELOCATION PLAN

The major release of toxic or radioactive materials or other hazards may require the evacuation of the site or some portion of the Laboratory. Although it is most probable that only an area will need to be cleared, the Emergency Staff will keep abreast of all community development to facilitate the choice of a suitable location for a site evacuation.

In principle, the Relocation Plan will be implemented when necessary by an emergency announcement, preceded by the ALERT signal. The Area Emergency Supervisor will then provide the direction for the loading of the maximum number of passengers into the minimum number of vehicles, as outlined in the Shelter Relocation Plan (following), instructing them to move along the shortest route to the relocation point as quickly and as carefully as conditions will permit. The Area Emergency Supervisor will assure complete evacuation and will then join his group at the relocation point to assist the Emergency Staff with the task of registering, monitoring, decontaminating, or otherwise assisting the evacuees.

Emergency units will establish procedures designed to salvage all equipment and materials which would be of significant value to the continuation of emergency measures.

SHELTER RELOCATION PLAN

The purpose of this plan is to provide procedures for the relocation of all Laboratory and AEC-CH employees to designated fallout shelters in the event of an enemy attack. The plan proposes to move all employees with the least number of vehicles and with the least possible exposure.

When implementation of this plan is required, the ALERT signal (a continuous tone signal) will be sounded over the outdoor sirens and site-wide Public Address system. Upon hearing this signal, all personnel are to move as close as possible to the Public Address speakers to receive verbal instructions. (The ALERT signal will also precede announcements relating to an on-site disaster or a tornado.) All pertinent information will be announced at this time.

Upon instructions, all employees will assemble near the main entrance of the building in which they are located, staying indoors. All persons having cars parked in nearby parking areas are to form a line to the left of the corridor or lobby, facing the exit doors. All others will form a line to the right. The Area Emergency Supervisor and his staff will then assign four passengers to each driver, have them move out to the assigned car, load and proceed in the direction indicated on the attached map for shelters - or as may otherwise be announced. If there is an excess of drivers, groups of five will be assembled from these employees and dispatched in a single car per group. Whenever possible, personnel to direct traffic will be stationed at appropriate places along the route. In buildings or areas served by more than one parking area and entrance, the Area Emergency Supervisor will issue specific assembly instructions.

The movement of personnel from certain buildings to designated fallout shelters and the routes to be followed are indicated on the attached map. Areas are color-keyed to facilitate identification. Personnel in buildings close to the designated fallout shelters are asked to walk to the shelter. This action is indicated on the map by the letter "W".

Should the necessity for relocation occur during the lunch period, personnel in the cafeterias should utilize the fallout shelters designated for the cafeterias. Specifically, those in the Building 3 cafeteria will relocate by car to the ZGS Area; those in the Building 213 cafeteria will walk to Building 205.

Emergency personnel assigned to designated fallout shelters will assist relocated personnel by directing the parking of vehicles and by directing personnel through building entrances to the proper shelter areas. The Area Emergency Supervisors for Buildings 203, 205 and ZGS are directed to instruct their emergency personnel in these duties.

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SHELTER RELOCATION PLAN

84. *The purpose of the cleanup pump in series with the primary system purification unit and cooler is not clear. Also, a flow diagram should be provided showing all valves and other flow control and measuring devices. (RDT)*

Answer:² The cleanup pump is not operated during normal operation of the plant. Although it appears as being in series with the purification system, it is actually by-passed (isolated) during plant operation. The cleanup pump is used only during decontamination or cleanup of the primary system.

In order to aid in understanding the operation of the plant, a flow sheet showing all valves, flow control, and measuring devices in the primary system has been prepared and is incorporated as Figure IV-32 in Part 1, Section IV-G.

85. *Exception is taken to the postulated consequences of a metal-water reaction following a large primary system rupture. Specifically, on the assumption that evolved hydrogen should "pocket" and ignite, the resulting explosion pressure of 43 psi after 7 seconds may very well be damaging to personnel and/or equipment. (RDT)*

Answer:^{3,4} The question of hydrogen detonation has been reanalyzed. See Part 1, Section VIII-D.

86. *What is the projected 1980 population distribution as a function of distance and direction around the site? (CH)*

Answer:² Attached is a Population Summary Table and a map of the Laboratory with the surrounding area. It has been divided into quadrants with the center placed at the approximate location of AARR. Circles have been inscribed at distances of 2 km, 5 km, and 7 km. The segments of area thus obtained have been studied to assess present population and potential growth, based on the following assumptions:

1. The present pattern of land use will continue in the future.
2. Land use for public open space (roads, forest preserves, etc.) will neither increase nor decrease in the future.

The Summary Table makes reference to 5 km and 7 km as the Low Population Zone Radius and the Population Center Distance, respectively. These values constitute a change from those given in the early version of the PSAR, and are discussed in detail in Part 1, Section VII-C.

OFF-SITE POPULATION SUMMARY

<u>Segment¹</u> <u>Number</u>	<u>POPULATION</u>		
	<u>1960</u>	<u>1970²</u>	<u>1980²</u>
<u>0-2 km³ radius</u>			
A-1 (Northeast)	-	-	-
A-2 (Southeast)	-	-	-
A-3 (Southwest)	140	160	170
A-4 (Northwest)	<u>495</u>	<u>620</u>	<u>795</u>
Total	635	780	965
<u>2-5 km⁴ radius</u>			
B-1 (Northeast)	2,561	3,329	4,328
B-2 (Southeast)	1,409	1,691	2,029
B-3 (Southwest)	4,626	5,551	6,661
B-4 (Northwest)	<u>3,316</u>	<u>4,642</u>	<u>6,499</u>
Total	11,912	15,213	19,517
<u>5-7 km⁵ radius</u>			
C-1 (Northeast)	7,520	13,723	24,315
C-2 (Southeast)	2,019	3,029	4,543
C-3 (Southwest)	1,186	1,660	2,324
C-4 (Northwest)	<u>4,150</u>	<u>7,075</u>	<u>11,320</u>
Total	14,875	25,487	42,502

1. See attached map for location of segments.
2. Population trends based in part on information from Northeastern Illinois Metropolitan Area Planning Commissioner and local governmental agencies.
3. The exclusion area radius for AARR is 1,500-meters. All residents shown here are situated between 1,500-meter and 2-km radii.
4. The low population zone outer radius for AARR is 5 km.
5. The population center distance for AARR is 7 km.

ARGONNE NATIONAL LABORATORY
AND IMMEDIATE SURROUNDING AREA

87. *The treatment of site characteristics presented in ANL-6271 is inadequate to the needs of a safety evaluation for the AARR facility. Provide more comprehensive information on site characteristics with particular emphasis on seismology and severe weather conditions, including (a) a conservative estimate of the maximum ground acceleration which might occur at the site during the lifetime of the plant, (b) a discussion of the potential for differential ground movement, (c) frequency and severity of tornadoes and other cyclonic storms experienced in Illinois and the adjoining states, and (d) substantiation of the assumptions concerning wind velocity, pressure drop, and missiles that may be used as a basis for the consequences that would result should the AARR be struck by a tornado. (DRL)*

Answer:² An expanded treatment of the early version of the PSAR Section III - Site Characteristics - parts D and E, is given in Part 1, Section III. Pertinent information follows.

a. Seismic Ground Accelerations

A conservative estimate of the maximum ground accelerations that might be expected to occur at the site during the lifetime of the facility is 0.1 g horizontal and 0.066 g vertical.

A ground acceleration of 0.07 g^{1*} has been experienced in this area and was caused by a relatively small earthquake of intensity VI with an epi-center approximately 20 miles southwest of ANL on January 12, 1912.

This is the only acceleration of significant magnitude that has been obtained for this area and represents a small local shock felt over only 40,000 square miles. Undoubtedly the site has experienced a greater acceleration. Records indicate that a large earthquake occurred on May 26, 1909 with an intensity VIII and with an epi-center approximately 15 miles west of ANL. This earthquake was felt over an area of 500,000 square miles but, unfortunately, accelerations of these larger tremors were not obtained.

The accelerations estimated above are similar to those used as a basis for the seismic design of the Dresden Nuclear Power Station Unit #2 which is located approximately 25 miles southwest of ANL. It is not anticipated that accelerations of greater magnitude would occur at the AARR site. However, these accelerations, in conjunction with the local geology and seismic history, will be reviewed by a recognized seismic expert to determine their appropriateness to the design of the facility.

b. Potential for Differential Ground Movement

Due to the dense character of the soils that exist between bed-rock and the reactor building foundation mat, it is not expected that any significant differential ground movement (consolidation of the soils beneath the foundation mat) will take place during an earthquake. The soil borings taken at the site do not indicate

* Superscripts refer to references at the end of this answer.

any zones or areas of soft materials existing below the foundation mat. The borings do, however, indicate that the soils are composed of very tough silty clays and clayey silts with penetration resistances that average over 50 blows per foot (standard penetration - 140-lb hammer falling 30 in. on a 2-in. O.D. split spoon) for the region between bedrock and the bottom of the foundation mat.

Some differential ground movement over a considerable period of time (relative settlement of the various building foundations) can be expected and will be taken into account in the foundation design. This small settlement will not constitute a safety hazard.

c & d. Tornado Frequency and Severity, and Substantiation of Assumptions of Tornado Dynamics

Statistics compiled by the Weather Bureau, U.S. Department of Commerce, indicate that over a 43-year period the State of Illinois as a whole has averaged 7.00 tornadoes per year. The averages for the adjoining states during the same period of time are as follows:

Iowa	13.93
Missouri	10.37
Kentucky	1.77
Indiana	6.26
Wisconsin	4.35

The frequency of occurrence of tornadoes in the immediate area is much less than in the state as a whole. Based on 2° squares (approximately 15,400 sq. miles) and for a 40 year period, Weather Bureau statistics show that this frequency is 1.25 tornadoes per year. Reducing the area to 1° squares (approximately 3,850 square miles) and using the data collected from the period of 1953 to 1958 by the Weather Bureau, the frequency becomes 1.00 tornado per year for the area within a 35-mile radius of Argonne National Laboratory.

Unfortunately, the severity of tornadoes is reported as a function of the resulting property loss rather than in terms of actual size or length of contact with the earth. It is not known exactly what wind velocities may be obtained within a tornado system. Some tornadoes have been estimated to have wind velocities as high as 500 mph.² This appears to be excessive for the average tornado (if there is such a thing as an "average" tornado). A number of studies concerning the distribution and final disposition of tornado debris have led to the general conclusion that tornado wind velocities are in the range of 300 mph.³ Wind velocities of up to 280 mph have been measured by the U.S. Weather Bureau's experimental Doppler radar equipment.⁴ Computations based upon motion pictures taken of the Fargo, N.D. tornadoes of June 20, 1957 indicate that the cyclostrophic winds at the tornado funnel edge were in the neighborhood of 230 mph.⁵ Based upon the above observations and assumptions, wind velocities of 300 mph existing from grade to 50 ft above grade and wind velocities of 340 mph for heights over 50 ft above grade will be used to determine wind pressure exerted on vertical surfaces due to a tornado.

The existence of a pressure drop through the "eye" of a tornado is well known, but sufficient statistical data to determine the actual pressure drop caused by a tornado is lacking. Various readings of this pressure drop have been obtained.⁶ However, it is not known if the actual "eye" of the tornado was sensed in these measurements. Barographic measurement of one tornado center, passing 40 yd distant from the barograph, indicated a sudden drop in pressure of 0.65-in. Hg.⁷ Another reading taken between two barometers about seven city blocks apart during a tornado showed that there exists at least a 2.42-in. Hg pressure drop.⁸ The latter reading indicates a little more than an 8% pressure drop. This seems to be a conservative estimate and possibly is the best measurement that has been obtained to date.

With regard to missiles generated by tornado activity, calculations were performed on the AARR containment structure to determine whether it could withstand the destructive effects of various sizes of steel and wooden missiles. The results show that the velocities required by the missiles for breaching containment cannot be obtained by tornado activity.

In considering the effects of missiles on targets it is necessary to distinguish between penetration and perforation. Penetration implies entry of a missile into a target without passing through it. Perforation means passage of all of the missile through the target. Thus, for containment purposes, the minimum velocity that will cause perforation of the AARR containment shell is the primary quantity of interest.

The basic formulae for calculations are taken from the following reference: R. Recht and T. Ipson, "Ballistic Perforation Dynamics," J. Appl. Mech. V. 30, pp 385-390, 1963. These authors developed an analytical equation for the minimum perforation velocity of thin plates by blunt projectiles. This equation was experimentally verified to within 10%. Thus, calculations made using this equation have a sound experimental basis.

The results of this analysis are given in the following table.

Minimum Perforation Velocities (ft/sec)

T/d

T/L	1/2	1/3	1/4	1/5	1/6	1/8	1/10	1/20
1/2	1790	1480	1210	1170	965	838	734	506
1/3	1360	1080	921	817	740	632	563	392
1/4	1130	900	770	683	620	530	470	328
1/5	980	781	670	590	540	460	411	286
1/6	876	695	595	540	484	412	366	256
1/7	800	638	548	482	443	378	336	235
1/8	745	592	503	448	409	350	311	218
1/9	680	552	472	418	382	326	288	203
1/10	645	522	446	394	360	310	274	192
1/15	520	414	340	314	286	246	218	153
1/20	458	354	268	264	246	211	189	132
1/25	388	316	243	240	216	189	167	117
1/30	350	285	220	218	197	171	152	105
1/40	305	246	183	182	158	147	130	91.0
1/50	270	218	168	167	150	130	116	81.5
1/75	218	177	153	135	123	107	95.0	66.0
1/100	188	153	130	116	105	92.0	81.5	57.0

In the table, T is the equivalent thickness of the containment structure, L is the length of the missile and d is the diameter of the missile. The impact velocity is at least twice the perforation velocity given in the table.

The assumptions inherent in this analysis are as follows:

- (1) the missile does not undergo any severe deformation upon impact or during the perforation process
- (2) the plate material is relatively strong in shear, so that it does deform excessively, and is relatively thin
- (3) the range of missile impact velocities is considered to be within the ballistic limit (minimum perforation velocity)
- (4) the major physical process in the plate is shearing action during perforation by the missile
- (5) an equivalent steel plate thickness can be found for the concrete in the outer containment shell
- (6) the missile and plate are of the same material
- (7) the missile impacts transversely (perpendicularly) on the plate.

These assumptions lead to results which are conservative since all other physical processes and properties would require higher impact velocities to perforate a given plate thickness.

The weakest part of the AARR outer containment shell is the dome which consists of one foot of concrete backed up by a 3/8-in. steel liner. This was chosen as the most vulnerable portion of the shell. To obtain a steel plate equivalent of the concrete the standard rule that the penetration of a missile varies roughly as the square of

the material density was used. Hence, the one foot of concrete is equal to 1.130 in. of steel. Coupled with the 3/8-in. steel liner the outer containment shell provides an equivalent 1.51 in. of steel as protection against perforation from missiles generated outside of the shell. Thus, the size of missiles (in inches) for which the table applies are:

$$3.0 \leq d \leq 30 \quad ; \quad 3.0 \leq L \leq 150.$$

The lowest perforation velocity in the table is 57 ft/sec for a steel missile 2-1/2 ft in diameter and 12-1/2 ft long and weighing 30,000 lb. This means that the lowest impact velocity is 114 ft/sec or 80 mph. Study of existing reports²⁻⁷ indicated that no missiles of this size would be made to strike the roof at the required impact velocity. If a missile of this size strikes the side of the building the required impact velocity for perforation would be four times as much or 320 mph, since the side wall is twice as thick as the dome. Since this velocity is greater than the maximum tornado velocity no penetration of the side wall can occur.

The calculations were repeated for a wooden missile of the same size with a density of 22.5 lb/ft.³ The results showed that the wooden missile would have to attain an impact velocity of 660 mph to cause perforation of the dome. Since this is greater than the tornado wind velocity no perforation can occur. Steel and wooden missiles of smaller sizes require even higher impact velocities.

It is conceivable that during tornado activity a portion of the exhaust stack could be broken off and fall against the outer containment shell and cause damage. The question then arises as to whether this missile could perforate the shell.

The section of exhaust stack that breaks off is assumed to be 5 ft in diameter, 20 ft long, and 3/8-in. thick. For purposes of study the cylindrical shell has a solid equivalent of 7 in. diameter. This analysis gives a perforation velocity of 62.5 ft/sec. However, since the cylindrical shell has twice the shear area of a solid shell the perforation velocity is twice that of the solid equivalent, or 125 ft/sec. This means that the impact velocity must be at least 250 ft/sec or 170 mph. The vertical velocity it obtains from free fall acceleration due to gravity is 113 ft/sec or 76 mph for a fall of 105 ft. This means that the wind would have to add at least 100 mph of velocity to cause perforation. Since the wind velocity vector is horizontal to the ground its component along the normal to the roof will be very small so that there is no conceivable way in which the stack section can receive the necessary velocity. In addition, the steel cylinder will most likely strike at an oblique angle which requires additional impact velocity to cause perforation. Although some gouging of the concrete might occur no loss of containment would result due to impact of this missile.

Thus, the result is that there is no conceivable way in which missiles generated by a tornado can cause containment to be breached. It should be recalled that the results of the table hold for normal impact. If the steel missiles strike at angles away from the normal even higher impact velocities will be required. Other missiles of materials softer than steel will suffer considerable deformation and possible fracture.

References

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2. Tetsuya Fujita, Aerial Survey of the Palm Sunday Tornadoes of April 11, 1965, Satellite & Mesometeorology Research Project, Department of the Geophysical Sciences, the University of Chicago, SMRP Research Paper Number 49, January, 1966.
3. Task Committee on Wind Forces of the Committee on Loads and Stresses of the Structural Division, Wind Forces on Structures, American Society of Civil Engineers, Transactions, Paper No. 3269, Vol. 126, Part II, (1961), p. 1124.
4. L. V. Wolford, Tornado Occurrences in the United States, Technical Paper No. 20, U. S. Department of Commerce, Weather Bureau.
5. Tetsuya Fujita, A Detailed Analysis of the Fargo Tornadoes of June 20, 1957, Research Paper No. 42, U. S. Department of Commerce, Weather Bureau.
6. S. D. Flora, Tornadoes of the United States, (1953), p. 28.
7. J. A. Carr, A Preliminary Report on the Tornadoes of March 21-22, 1952, Monthly Weather Review, Vol. 80, No. 3 (1952), pp. 50-58.
8. Julius Baier, Low Pressure in St. Louis Tornado, Monthly Weather Review, Vol. 24, No. 9, (1896), p. 332.

88. *Additional engineering design bases and analyses for the critical features of the plant which could be affected by a tornado or earthquake should be provided, and should include consideration of the following:*

(a) Discuss the criteria to be used in the design of the various plant components and containment as they are affected by seismic forces.

(1) State how the seismic design criteria are applied in design of the various structures and components. Include a listing of the primary and auxiliary equipment that must continue to function during and after an earthquake in order to provide adequate protection to public health and safety.

(2) State the maximum allowable stresses in terms of a percentage of working or yield stress for these structures and components under seismic, functional, and accident loadings. This consideration should include the load criteria, whether the design is to be made for working or on ultimate design concepts, and if the latter the load factors, combination of loadings, allowable stresses, etc. (DRL)

Answer:² Equipment, structures and systems important to safety are those which are required to maintain the functions of three principal items, namely: (1) shutdown capability; (2) decay heat removal capability; and (3) containment integrity. All such equipment, structures and systems are identified below as Class I Items. The seismic design criteria will be implemented as follows:

Containment

- (1) Primary steady state stresses, when combined with the seismic stresses resulting from the response to the design earthquake ground accelerations (considering horizontal and vertical accelerations, simultaneously), shall not exceed the allowable working stresses established by the appropriate design codes. (Design standards shall be Sections III and VIII of the ASME Boiler and Pressure Vessel Code and ACI 318-63, Building Code Requirements for Reinforced Concrete).
- (2) Primary steady state stresses, when combined with the seismic stresses resulting from the response to twice the design earthquake ground accelerations (considering horizontal and vertical accelerations, simultaneously), are limited such that they shall not exceed the minimum value of yield strength for the materials involved.

Class-I Items other than Containment

- (1) Primary steady state stresses, when combined with the seismic stresses resulting from the response to the design earthquake ground accelerations shall remain within the elastic region of the materials involved.
- (2) Primary steady state stresses, when combined with the seismic stresses resulting from the response to twice the design earthquake ground accelerations shall be limited such that the function of the component, structure or system shall not be impaired, whether it be to provide safe and orderly shutdown of the plant, remove decay heat, or to maintain containment integrity.

Following is a listing of Class-I Items:

- (1) Reactor containment building and all penetrations.
- (2) Primary and secondary cooling systems, including all related vital instrumentation.
- (3) Control room.
- (4) Class-I power supply and the Class-II diesel-generators and their fuel storage and supply systems.
- (5) Cooling tower basin and sump.
- (6) Control rod drives, supports, and control room indication of rod insertion.
- (7) Reactor vessel, internals and support.
- (8) All experimental facilities, the failure of which could add reactivity to the system.
- (9) Reactor containment building isolation valves, operators, their associated radiation monitoring channels, etc.
- (10) Earthquake detector and associated scram circuitry.

The allowable stresses for combined loadings on the containment structure are stated in question 89 below.

For the seismic loading conditions, all Class I Items - other than containment - will be designed such that the combination of stresses from the design earthquake loads plus functional loads shall not exceed the yield stress of the material. In addition, these items shall be capable of withstanding the effects of an earthquake of twice the intensity of the design earthquake, which when combined with functional loadings will not result in loss of function of such items of equipment or structure in order to assure a safe shutdown of the plant.

(b) Describe the procedures to be followed in making a dynamic design analysis of the critical items required to assure a safe plant shutdown. With respect to earthquakes and effects therefrom, provide a plot of the response spectra to be used in the design, with modifications for various degrees of damping. (DRL)

Answer:² All Class I structures, components and piping systems will be examined by a recognized seismic expert to determine which items will require dynamic analysis. The general procedure to be used in performing the dynamic analyses for each item will be as follows:

1. Determine the natural period of vibration of the component, structure or piping system.
2. Determine the response acceleration of the component from the acceleration response spectrum curve, at the appropriate natural period, using the spectral curves for a maximum ground acceleration of 0.1g.*
3. Calculate the seismic stresses in the component due to its response to the 0.1g* earthquake, combine these with other primary stresses caused by functional loadings and compare with limits set forth in seismic design criteria.
4. Determine the response acceleration of the component from the acceleration response spectrum curve, at the appropriate natural period, using the spectral curves for a maximum ground acceleration of 0.2g.*
5. Calculate the seismic stresses in the component due to its response to the 0.2g* earthquake, combine these with other functional loads and check resulting deformations and required plastic energy absorption to ensure functioning of Class I components in order to maintain safe plant shutdown.

Argonne is presently negotiating a contract with a recognized seismic expert, who will examine all of the AARR seismic design criteria and will perform all necessary dynamic analyses. As an interim measure, we are attaching a copy of the response spectrum for acceleration which was used for the Dresden Unit Nos. 2 and 3 for Commonwealth Edison Company. These apply for a maximum ground acceleration of 0.1g and are used for 0.2g by doubling the indicated values. The Dresden seismic data has been used for some of the preliminary design effort on AARR; however, it is recognized that a new set of curves must be generated for the AARR site.

The table below gives tentative values for damping factors to be used in conjunction with the spectral curves, and which will be confirmed or modified by the seismic expert.

* This value of maximum ground acceleration will be either confirmed or modified by a recognized seismic expert.

PREPARED BY JOHN A. BLUME AND ASSOCIATES, ENGINEERS, SAN FRANCISCO, CALIFORNIA
REPRODUCED THROUGH THE COURTESY OF SARGENT & LUNDY, ENGINEERS, CHICAGO, ILLINOIS

DAMPING FACTORS

<u>Item</u>	<u>% of Critical Damping</u>
Containment Building (Reinforced Concrete)	2.0
Concrete Support Structure for Reactor Vessel and Pool	2.0
Steel Assemblies	
(a) Bolted (or riveted)	2.5
(b) Welded	1.0
Critical Piping Systems (Class I)	0.5
Miscellaneous Concrete Structures Above Grade	5.0

(c) *What margin of safety is considered for the normal maximum earthquake, and under what more extreme conditions will it be possible to shut down the reactor safely? Will the design encompass provision for a maximum earthquake of twice the intensity of the design earthquake, with provision for no loss of function so far as safe shutdown is concerned? (DRL)*

Answer:² All Class I equipment, piping and structures will be designed such that their response to the normal maximum earthquake, when combined with functional loads (plus accident loads, in the case of containment), will remain in the elastic region of the materials involved. In addition, for the containment building, the combined stresses due to the response from the normal maximum earthquake, functional and accident loads will not exceed the normal working stresses permitted for the materials involved.

Class I Items other than Containment

In terms of a discussion of margin of safety, therefore, it is clear that for all Class I items (except containment) the margin of safety is related to how much plastic deformation the structure or component can undergo and still be functional. Any one particular component will normally have loads and forces occurring simultaneously with the earthquake, and the design provides for elastic response under the combined action of all these loads. Considering that at least a small amount of plastic deformation is acceptable for earthquakes of intensities greater than the normal maximum earthquake, it can be stated that the margin of safety is at least one. In other words, each design will be analyzed to ensure that no loss of function occurs under the combined action of functional and twice normal maximum earthquake loads.

Containment

Additional conservatism is exhibited in the design of containment since under the combined action of functional, accident and normal maximum earthquake loads the normal working stresses of the materials are not exceeded. Under the combined action of functional, accident and twice the normal maximum earthquake loads the stresses do not exceed the yield strength of the materials. Therefore, the margin of safety is one based on yielding of the materials. The actual margin of safety is greater than one, if some plastic deformation is permitted.

(d) *It is noted that there will be at least one large crane in the installation and possibly more than this. What provisions are made to ensure that these cranes will not topple or otherwise cause damage which would lead to difficulties with safe shutdown of the plant? (DRL)*

Answer:² The bridge of the 15-ton Reactor Building crane cannot be unseated from the circumferential haunch or girder that supports the crane rail while undergoing an upward acceleration due to an earthquake. There is not sufficient side-clearance between the building wall and the crane end trucks, nor is there sufficient headroom between the underside of the dome knuckle and the top of the crane to permit passage of the crane end trucks past the circumferential support.

The trolley will be secured to the crane bridge in the event of an upward acceleration capable of lifting it off the rails. This will be done by a mechanical device that will engage the underside of the rail head or the underside of the top flange of the crane bridge girder when the trolley moves upward, off the rails. This device, similar to an "L" or hook, will be fastened to both sides of the trolley carriage and will hang below the member it is to engage upon uplift, with enough clearance to permit normal operation of the trolley without interference.

(e) *It is noted that there is to be a 250 ft high reinforced concrete stack. What techniques are to be used in the design of this stack for wind, earthquake, and other loadings? Provide an analysis of the possible toppling of the stack and its effects upon the plant. (DRL)*

Answer:² The exhaust stack has been changed from a free standing reinforced concrete stack to a guyed steel stack. Since this stack is not considered a Class I structure (it is not required for the safe and orderly shutdown of the reactor, nor is it required for the containment or controlled release of fission products that may result from a MCA), no special design techniques will be used in its design. The Uniform Building Code will be employed for the design of this stack with regard to wind and earthquake loadings. The hazard of breaching the containment by virtue of the stack falling upon the containment building has been greatly reduced by the relocation of the stack away from the building and by the change to an all-steel design. However, an analysis was made assuming that the top portion of the steel stack became a missile generated by a tornado and that it struck the containment building. This analysis showed that the containment would not be breached by such a missile. Details are given in response to question 87 above.

89. *What consideration has been given to the effects of potential missiles that might be generated by a tornado? (DRL)*

Answer:² See response to question 87 above.

90. *With respect to the containment, discuss the limits established for the primary and secondary stresses, in terms of percent of yield strength, for the reinforcing steel and containment liner for each combination of calculated loadings associated with (a) normal operating conditions, (b) the pressurized condition during the structural integrity proof test, (c) the leakage rate test pressure conditions, and (d) the postulated design accident condition. (DRL)*

Answer:² Combinations of stresses for the design of the liner plate will be in accordance with Article 4 - Design of Section III - Nuclear Vessels - of the ASME Boiler and Pressure Vessel Code for all loading conditions except the condition of loading that includes a MCA plus twice the normal earthquake load. The limits for primary stresses have been set so as to ensure that the yield points of the materials used in the containment structure will not be exceeded. Generally, the "working stress design" method is to be employed. Normal working stress limits will only be exceeded under the condition of a MCA plus twice the normal earthquake load. Allowable stresses for the liner plate material will be 1.10 times the maximum allowable stress values given in Section VIII - Unfired Pressure Vessels of the ASME Boiler and Pressure Vessel Code. The table of allowable stresses following is presented to illustrate the above discussion of stress limits.

91. *Discuss the design parameters and design methods that will be employed to assure that the containment liner will function as an adequately leaktight membrane under various possible pressure and temperature conditions that might cause overstressing at points of attachment at containment openings and at transition sections. What consideration will be given to possible loss of integrity due to corrosion? (DRL)*

Answer:² Design of the steel plate containment liner is based on achieving a leak rate of 0.1% of the contained mass per day for the design pressure and is based on the following loads:

- (a) Internal negative pressure of 0.5 psig
- (b) Internal liner peak temperature of 117°F

Condition (b) is considered to impart to the liner plate of the walls and dome the greatest stress due to heat load, and plate anchorage spacing will be determined so as to prevent buckling.

The reinforced concrete portion of the containment structure will be designed to limit the strains that the liner can experience to strains that will be within the elastic range under the loading condition that includes twice the normal earthquake load plus the load due to pressure and temperature rise resulting from a MCA. The normal earthquake load is considered to be that load caused by a combination of ground accelerations of 0.1g horizontal and 0.066g vertical imparted to the containment structure simultaneously via its substructure.

ALLOWABLE STRESS LIMITS FOR CONTAINMENT

Loading Condition	Liner Plate*			Reinforcing Steel **
	P_m	$P_m + P_b$	$P_m + P_b + Q$	
Normal Operating Conditions D.L. + L.L.	S_m	$1.5 S_m$	$3 S_m$	$f_t = .5 F_y$
Structural Pressure Proof Integrity Test D.L. + 1.15 P.L.	S_m	$1.5 S_m$	$3 S_m$	$f_t = .5 F_y$
Leak Rate Test Conditions D.L. + M.A.P.	S_m	$1.5 S_m$	$3 S_m$	$f_t = .5 F_y$
MCA Condition #1 D.L. + L.L. + P.L. + T.L. + E.L.	S_m	$1.5 S_m$	$3 S_m$	$f_t = .5 F_y$
MCA Condition #1 Plus Twice the Normal Earthquake D.L. + L.L. + P.L. + T.L. + 2 E.L.	F_y	F_y	$3 S_m$	$f_t = F_y$
MCA Condition #2 D.L. + L.L. + P.L. + T.L. + W.L.	S_m	$1.5 S_m$	$3 S_m$	$f_t = .5 F_y$

Where

P_m = Primary membrane stress

P_b = Primary bending stress

Q = Secondary stress

S_m = Design stress intensity

f_t = Allowable tensile stress in reinforcing steel

F_y = Specified minimum yield point of steel used

M.A.P. = Maximum accident load due to MCA = 4.4 psig ⁺

D.L. = Dead load

L.L. = Live load

P.L. = Design pressure load = 7.5 psig

T.L. = Temperature load in liner due to MCA = 117°F liner plate peak temperature ⁺

E.L. = Earthquake load = 0.1g Horiz. and 0.066g Vert.

W.L. = Wind load associated with 300 mph tornado winds

* S_m is allowable membrane stress as per Section VIII, ASME Boiler and Pressure Vessel Code times a factor of 1.10.

** $F_{ymin} = 40,000$ psi

⁺ This value will be substantiated in the Title II design.

The adequateness and appropriateness of these accelerations will be reviewed by a recognized seismic expert in conjunction with the design of the containment structure.

Additional reinforcement will be provided at points where the continuity of the containment structure is interrupted, i.e., air locks and equipment hatches. Particular attention will be given to the design of the liner anchorage at all transition points for proper function during and after erection, including leak testing at either time.

Damp-proofing of the exterior surfaces of the below grade containment structure walls that are in contact with the earth will be done to mitigate the possibility of the loss of integrity due to corrosion.

Granular backfill and a perimeter drainage system at the base of the containment structure foundation will further ensure that ground water flowing through the earth adjacent to the containment structure will not corrode the liner plate or its anchorages.

The exposed surface of the liner plate (inside the containment) will be coated to reduce the corrosion possible due to water condensation, water leaks, etc.

92. *Provide the design criteria, system description, and description of the proposed schedule and method of testing for the following: (a) all isolation valves, (b) all types of penetration seals, (c) the vacuum relief system, and (d) the containment liner. To what extent will the penetrations be designed to allow independent leak testing at design pressure? Will the vacuum relief valves be redundant as to both vacuum relief function and containment integrity? Discuss the considerations which have been given to the extremes of temperature, pressure and humidity associated with severe internal or external environmental conditions. (DRL)*

Answer:² ISOLATION VALVES

A. General Description

Isolation valves are to be used on pipes penetrating the reactor building shell. They are to be automatically closed upon receipt of a signal from one or more parameters to be determined in Title-II. The isolation valves must be so constructed as to permit fluid passage into or out of the building under controlled conditions and should not violate the gastight integrity of the structure.

B. Design Criteria

1. Fluid-tight across valve body and valve disc, and to withstand normal operating conditions, maximum internal pressure and temperature, and reduction of temperature as a function of time. These parameters, as well as the normal and abnormal environment, shall be determined for each system in Title-II.

2. Valves shall be satisfactory for application involving frequent operation and operation after long periods of inactivity.
3. Where possible, the valve seating and shaft seal material shall be replaceable without removal of the valve from line.
4. Valves which are located outdoors shall have all moving parts weather-protected and shall have provision for manual closure.
5. Valve operators shall be normally-closed, air to open, spring to close, where practicable, or air to close. With air to close, the air compressor shall be supplied by Class-I power.
6. The prescribed leakage rate shall be determined in Title-II.
7. End connections shall be determined in Title-II.
8. Ventilation system isolation valve closing time shall not exceed one second; other valve closing times shall be determined in Title-II.

C. Proposed Schedule and Method of Testing

1. Valves in ventilation and gaseous vent systems
 - a. Shop test by manufacturer.
 - b. Field test by integrated leak test of containment structure.
 - c. Retest each valve periodically to demonstrate leaktightness.
2. Valves in liquid systems
 - a. Shop test by manufacturer.
 - b. Field test by system.
 - c. Retest each valve periodically to demonstrate leaktightness.

ELECTRICAL PENETRATIONS

A. General Description

Electrical cables for low voltage systems shall be routed in such a manner that several multiconductor cables shall pass through the reactor building shell via a single pipe penetration. Bulkhead-type separable connectors shall be attached to the interior and exterior ends of the pipe penetration. Each penetration shall be designed for periodic tests after installation, without disconnecting the cables, to determine leaktightness.

High voltage cables shall penetrate the reactor building shell by means of three-conductor potheads or high voltage bushings. The type of penetration shall be determined in Title-II.

B. Design Criteria

Basic configuration of the penetrations for the low voltage electrical services shall be standard pipe, with welded flanges and removable connector plates at each end. The removable connector plates shall have standard connector arrangements and be bolted to the end flanges. Each assembly shall be designed for periodic leaktightness tests.

Design of the high voltage penetrations shall be performed in Title-II.

C. Proposed Schedule and Method of Testing

Each penetration shall undergo the following tests:

1. Shop test by manufacturer to determine original leakage rate at design pressure.
2. Test of initial installation by integrated leak rate test of entire containment structure at maximum accident pressure, and periodic retest by the same method but at lower pressure.
3. Periodic retests to demonstrate leaktightness of individual penetrations at design pressure.

ACCESS PENETRATIONS

A. General Description

The air locks and the equipment hatches are the openings into the Reactor Building for passage of personnel and equipment into and out of the building. The air locks must permit entrance to the building under controlled conditions and without violating the gastight integrity of the structure.

The penetrations are as follows:

1. two personnel air locks
2. two equipment hatches
3. one emergency personnel air lock

The personnel air lock is employed for normal entrance and exit of personnel. The emergency air lock is employed for the exit of personnel under emergency conditions. The emergency personnel air lock does not permit entrance to the building under normal conditions. Each air lock includes two doors, both of which open toward the inside of the containment building. The equipment hatches are large bolted openings in the building wall used to install or remove large items of equipment. They are not air locks and are to be used only when the plant is shut down and secured, under conditions where it will be permissible for the building to be opened to the atmosphere.

The personnel and emergency air locks are designed for manual operation.

B. Design Criteria

1. Personnel air locks

- a. Each air lock includes two doors designed with interlocks which permit only one door to be opened at a time, thus maintaining the gastight integrity of the building.
- b. Each door shall be designed to withstand an internal gas pressure of 7.5 psig and gas temperature of 235°F.
- c. Equalizing valves shall be provided to eliminate differential pressure across the doors so they can be opened. The equalizing rate should be as fast as possible without causing personnel discomfort. Air locks shall not be vented to the area immediately surrounding the air lock exit.
- d. The doors must be interlocked to prevent both being open simultaneously.
- e. Visual indication shall be provided for position of interlock, equalizing valve, and door.
- f. An interior lighting system, with an emergency power supply, must be available.
- g. There shall be an emergency communication system.
- h. Provision will be made to override door interlock mechanism to allow both doors to be left open under proper administrative authority during plant shutdown.
- i. Air lock shall be manually operated.
- j. Provision shall be made to permit periodic retesting to demonstrate leaktightness.

2. Equipment Hatches

- a. Large bolted opening of sufficient size to accommodate the largest equipment to be transferred only during plant shutdown.
- b. Provision shall be made to permit retesting to demonstrate leaktightness after each hatch reclosure.

C. Proposed Schedule and Method of Testing

1. Personnel air locks
 - a. Shop test by manufacturer to determine original leakage rate at design pressure.
 - b. Field test by integrated leak test of containment to demonstrate leaktightness.
 - c. Retest periodically to demonstrate leaktightness at design pressure.
2. Equipment Hatches
 - a. Field test at design pressure to demonstrate leaktightness.
 - b. Retest after each closure to demonstrate leaktightness at design pressure.

VACUUM RELIEF SYSTEM

A. General Description

The containment structure will be designed and constructed to be capable of withstanding an internal pressure of 7.5 psig and internal negative pressure of 0.5 psig. The vacuum relief system, which consists of two valves in parallel, will prevent negative pressure within containment due to atmospheric pressure of 0.5 psig. The vacuum relief system, which consists of two valves in parallel, will prevent negative pressure within containment due to atmospheric pressure changes to go below 0.5 psig. The valves are spring-loaded, normally-closed, designed to satisfy the fail-safe criterion. In addition, internal pressure shall act to close the valve. Any valve position other than normal (closed) will be annunciated in the control room.

B. Design Criteria

1. Gastight across valve body and pressure relief device to withstand maximum internal pressure of 7.5 psig and gas temperature of 235°F.
2. Pressure relief device shall be spring-loaded, normally-closed.
3. Valve capacity shall be determined in Title-II.
4. Either valve shall have sufficient capacity to pass required flow rate.
5. Pressure relief device shall be set to start to function at a pressure differential not to exceed 0.5 psi.

C. Proposed Schedule and Method of Testing

1. Shop test by manufacturer.
2. Field test by pressurizing upstream side of pressure relief device.
3. Retesting same as field testing.

CONTAINMENT LINER

A. General

The Reactor Containment Building is a steel-lined reinforced concrete structure which is gastight and designed to withstand and contain internal pressure. The reinforced concrete structure is a cylinder with an ellipsoidal dome and a bottom which is flat except for the primary pump pit and a sump. The lower portion of the cylinder has an inside diameter of 119 ft. The ellipsoidal dome has a major inside radius of 60 ft and a minor inside radius of 30 ft. The entire interior surface of the exterior walls, dome, and foundation mat of the Reactor Containment Building is lined with a steel liner to prevent leakage to the atmosphere.

B. Design Criteria

1. Leak rate of 0.1% per day of the mass within the containment volume at 7.5 psig.
2. Liner material shall meet the requirements of Section III, Subsection B, Article N-1211, of the ASME Boiler and Pressure Vessel Code.
3. The test temperature for the Charpy V-specimens shall be +20°F.
4. Welding procedures and welder qualifications will be in conformance with Section IX of the ASME Boiler and Pressure Vessel Code.
5. The liner design is based on the following loads:
 - a. Internal negative pressure of 0.5 psig.
 - b. Internal liner temperature of 117°F.

C. Proposed Schedule and Method of Testing

1. Steel plate containment liner with no reinforced concrete.

a. Initial test

The purpose of this test is to locate leaks in the containment liner joints. The test will be consonant with the intent or spirit of Section III of the ASME Boiler and Pressure Vessel Code for Class B vessels. Defective welds will be cut out, rewelded and retested.

b. Integrated leak rate test at reduced pressure

After all leaks are eliminated by the above procedure this test shall be performed. All penetrations are closed, either by their normal closure or by test cap. The internal air pressure shall be determined in Title-II.

2. Steel plate containment liner with reinforced concrete

a. Strength test

With all penetrations closed, either by their normal closure or by test cap, the containment structure is pressurized to 8.6 psig to provide proof of its strength.

b. Integrated leak rate test at maximum accident pressure.

Following the proof-of-strength test this test shall be performed. The maximum allowable and design leakage rates are 1% per day and 0.1% respectively, of the mass within the containment volume at 7.5 psig. The duration of the test shall be not less than 24 hours.

GENERAL DESIGN CONSIDERATIONS

The relative humidity following a loss-of-coolant accident would soon reach a value of 100% because of evaporation of the warm water expelled from the primary system into the containment atmosphere. In addition, the internal pressure may approach the design pressure of the containment structure with a possible gas temperature of 235°F for a period of time following the accident. Thus, all penetrations, isolation valves, containment liner and vacuum relief system must be capable of operation not only at 100% relative humidity, but also at high temperature for some period of time.

93. *List those components (storage containers, operating equipment, instrumentation, etc.) which must continue to function at the containment design pressure, temperatures and humidity conditions. Indicate how this requirement is reflected in their respective design criteria and specifications. (DRL)*

Answer:⁴ There are no components which must continue to function at the containment design pressure except the containment itself, its penetrations and isolation valves. The design pressure will be higher than the maximum accident pressure by at least a factor of 1.1 (see Part 1, Section II, Criterion 17.2). With the current design, this factor is about 1.7.

For containment atmosphere conditions which have been calculated to exist following a MCA (see Part 1, Section VII), the components listed below must continue to function for the duration of the MCA. Design criteria are given in answer to other AEC questions and the appropriate question number is given adjacent to the listed component in each case:

- (a) Reactor Containment Building (RCB) - questions 90-92
- (b) RCB Isolation Valves - question 92
- (c) RCB Electrical Penetrations - question 92
- (d) RCB Access Penetrations - question 92
- (e) RCB Vacuum Relief System - question 92
- (f) RCB Penetration Seals - question 14.

General safety and design criteria for the above components are given in Part 1, Section II, Criteria 1.1, 1.4 through 1.7, 17.1 through 17.5, 19.1 through 19.7, 20.1 and 22.1 through 22.9. In addition, the design of all vital instrumentation, including that required to ensure RCB isolation, will be governed by Criteria 15.1 through 15.7.

94. *Discuss the manner in which the energy, liberated in the containment overpressure analysis, was distributed between the air and water vapor in the containment atmosphere. Also, indicate the quantity of water that was considered as a part of the available heat sink during the analysis. (DRL)*

Answer:³ See Part 1, Appendix F, Section F-II and Section VII-B-2.

95. *Consider the effect on the pressure decay curves if the hydrogen generated does not burn as released, but accumulates and then burns at the time that results in the highest pressure. (DRL)*

Answer:³ See Part 1, Section VII-B-2d. See also Part 1, Section VIII-D for a discussion on hydrogen detonation.

96. *Provide a failure analysis of the containment isolation system. (DRL)*

Answer:² As defined in Part 1, Section II, Criterion 22, the isolation system consists of the isolation valves, controls and operators that are provided in lines that open into either the primary coolant system or the containment volume and which also open or can be opened to the atmosphere outside the containment building. The systems that fall in this category are:

- 1. Reactor building ventilation system
- 2. Off-gas system
- 3. Tank vent system
- 4. Water storage system

In all these systems two isolation valves are used in series. These will fail preferentially to the closed position if valve motive or control power fails. The possibility for failure in the open position exists, but the probability of both valves in a particular line failing to close when needed is acceptably low. An analysis of specific failures in the containment isolation system follows.

Reactor building ventilation system

This system includes two penetrations through the containment wall, one for the supply duct, and one for the exhaust duct. Blowers are provided outside containment in the supply line, and both inside and outside containment in the exhaust line. All isolation valves are open during normal reactor operation.

Failure 1 - An isolation valve fails closed in the supply line:

- a. An alarm will be actuated in the control room.
- b. External exhaust and supply blowers will stop. The internal exhaust system flow will continue to run, recirculating air through the filters.

Failure 2 - An exhaust line isolation valve fails closed:

- a. An alarm will be actuated in the control room.
- b. The exhaust system bypass valve inside the building will open in response to a mechanical interlock between it and the exhaust isolation valve. This facilitates recirculation through the filters inside the building.
- c. The external exhaust and supply blowers will stop.

Failure 3 - Both inlet and exhaust valves fail closed:

This is no different in effect from a legitimate tripping of the building isolation system. Inlet and outlet flows are stopped, while recirculation continues through the internal clean-up system.

Off-gas system

This system combines the effluent gases from the primary system degasifier and the defective core pool degasser. The outlet line leads through a single containment penetration to the suction side of the containment building main exhaust blowers. The isolation valves are open during normal reactor operation.

Failure - An isolation valve fails closed:

- a. An alarm will be actuated in the control room.
- b. The flow of noncondensable gases will stop and pressure in the degasifier tank will begin to slowly increase. Pressure will increase very slightly until the partial pressure of the noncondensibles is sufficient to keep them in solution in the bulk water. The partial pressure of the water vapor is the principal determinant of degasifier pressure. It is determined by the temperatures of the off-gas condenser, and will not increase.

Tank vent system

The tank vent system combines the vent lines from several tanks and exits from the containment shell through a single penetration. The isolation valves are open during normal reactor operation.

Failure - An isolation valve fails closed:

- a. An alarm will be actuated in the control room.
- b. Ample time is available for repair since under normal conditions the levels of all the tanks involved change only very slowly.

Water storage system

This system provides a means for transfer and temporary storage of water normally contained in the various pools and the canal. The isolation valves are normally closed, and are opened only when the transfer pump is running.

Failure 1 - An isolation valve fails to open:

This is the safe position for these valves. Failure to open is only an operating inconvenience.

Failure 2 - One or more isolation valves fail to close during normal operation:

- a. Valve positions are indicated in the control room. The operator would normally notice the failure to close and take corrective steps.
- b. If the operator does not notice the failure to close, an alarm will be actuated if the isolation valve is still open when the transfer pump is stopped.

97. *Provide the criteria and their bases for containment isolation valve actuating times. (DRL)*

Answer:³ See Part 1, Section VII-C-2 and Section VI-E-2.

98. *What will be the criteria on the length of pipe lines between the containment shell and external isolation valves? What criteria apply to the connection of small lines to pipe lines between the containment shell and the external isolation valves? (DRL)*

Answer:² The length of pipe lines between the containment shell and external isolation valves shall be limited by the following:

1. The minimum distance required for bolting of ASA flanges which connect the penetration and valve.
2. The maximum distance which provides 6-in. clearance between any part of standard valve hardware and the outside surface of the containment wall.

Gas-containing connections made between the containment shell and external isolation valves shall meet the following requirements:

1. Small lines shall maintain containment integrity by employing two valves in series.
2. All lines penetrating containment shall meet the same design criteria as the containment structure for a length up to the isolation valves in those lines, as a minimum.

99. *What considerations have been given to missiles originating from:*

(a) *Penetrations in the head of the pressure vessel such as control drives, experiment loops, and reactor instrumentation. (DRL)*

Answer:³ See Part 1, Section VI-G-2a.

(b) *Regions external to the pressure vessel such as overhead cranes, disrupted experimental facilities near the reactor, and regions outside of containment. (DRL)*

Answer:^{3,4} See Part 1, Section VI-G-2b. A discussion of the effects of tornado-generated missiles on the Reactor Containment Building is given in question 87 above.

100. *Provide the following information on the reactor:*

(a) *The variations of the power coefficient with power and burnup. (DRL)*

Answer:^{3,4} Argonne has not yet calculated the transient-power coefficient of reactivity for the AARR with the HFIR-type fuel as a function of time of reactor operation. The power coefficient is determined primarily by the corresponding reduction in the total mass of coolant in the fuel zones, due to thermal expansion of the fuel plates and the increase in average temperature of the coolant.

The following question and answer for the HFIR are quoted from ORNL-CF-65-11-29.* They apply also to the AARR.

"(Question No.) 30. Quantitatively, how are the reactivity coefficients for the core expected to vary during the operating cycle?

"Answer The fuel region moderator density coefficients are affected by control rod position because the control rods, to a large extent, control the return leakage of neutrons from the reflector. When the rods are inserted, neutron leakage to the reflector has little chance of returning. With the rods out the return leakage is

* T. E. Cole to A. M. Weinberg, "High Flux Isotope Reactor - Safety Review Questions and Answers," Memorandum ORNL-CF-65-11-29 (November 12, 1965), p. 97

greater. Therefore, the moderator density coefficients become less negative as the rods are withdrawn. Calculations indicate that the coefficients decrease by a factor of two from the clean core symmetrical rod position to the fully withdrawn rod position."

The net temperature coefficients of reactivity for the clean reactor without targets in the ITC are summarized in Part 1, Appendix A, Section A-VI. With targets in the ITC, the temperature coefficients in the various zones are affected slightly. To determine approximately the equilibrium-power coefficient of reactivity, the temperature coefficient of reactivity at the given temperature should be multiplied by $0.49^{\circ}\text{F}/\text{MW}$, since operation at 100 MW heats the water entering the fuel zone by an average of 49°F . This includes the reactivity effect of the increased concentrations of xenon, which occur over a very much longer period of time and have a negligible effect on short-term variations in power.

Similarly, the power coefficient at operating conditions may be obtained by applying the factor of $0.49^{\circ}\text{F}/\text{MW}$ to the tabulated temperature coefficient at 155°F .

For reactivity insertions where the time scale for the accident is measured in seconds or less, it is the temperature coefficient in the fuel zone that is significant, since this is where most of the heat is absorbed. If the reactivity insertion is very rapid, the thermal expansion of the fuel plates becomes quite important. For further discussion, see Part 1, Section VIII-B.

(b) *The component values that make up the total power coefficient. (DRL)*

Answer:^{3,4} The component values of the power coefficient will be calculated in detail for presentation in the Final Safety Analysis Report for the AARR. The dominant contributor to the transient-power coefficient is the reduction in the mass of coolant in the fuel zone, because of thermal expansion of fuel plates and heating of the coolant in the fuel zone. The reactivity coefficients of voids in the inner and outer fuel elements were determined experimentally to be $-0.188\%/%$ coolant void, and $-0.384\%/%$ coolant void, respectively. As the fuel cycle progresses, it is indicated that these values become somewhat smaller in magnitude, but remain negative (see Part 1, Appendix A).

(c) *The effect that the $(n,2n)$ and (γ,n) reactions in beryllium have on the transient analysis. (DRL)*

Answer:^{3,4} The $(n,2n)$ reactions are included in the over-all reactivity computations, and the effect of such reactions on the mean prompt neutron lifetime is considered. It is assumed that the time scale for the $(n,2n)$ reactions is comparable to the time scale for the production of "prompt" neutrons. This assumption is conservative, in that the actual reactor may be considered to be a coupled system, with some of the neutron production occurring in the beryllium where the neutron lifetime is longer.

The (γ, n) reactions in the beryllium have a negligible effect on the analysis of positive reactivity transients. See also Part 1, Appendix C, Section C-II.

101. *Considering the variations of core design mentioned in the PSAR, on what basis is it concluded that the most conservative nuclear parameters have been assumed as input to the accident analyses? (DRL)*

Answer:^{3,4} The reactivity uncertainties due to allowed and conceivable variations in fuel loadings and burnable-poison loadings are summarized in Part 1, Appendix A, Section A-IV. These uncertainties in the net reactivity status of the reactor have a negligible effect on the reactivity-feedback coefficients used in the accident analyses.

102. *What analyses have been made to determine the effects of axial thermal expansion and bowing of the fuel plates on reactivity effects and potential for coolant channel blockage? (DRL)*

Answer:^{3,4} In Part 1, Section VIII-B-2, there is a tabulation of the reactivity effects of thermal expansion of fuel plates in a rapid transient. Two models of thermal expansion are discussed, one of which assumes that one-third of the total volumetric expansion of fuel occurs as an unrestrained longitudinal expansion. It is assumed that the thermal expansions of the fuel plates in the directions of thickness and width are relieved by non-preferential bowings of the fuel plate which uniformly eject water from the fuel zones.

See Part 1, Section IV-B-1 for a discussion of the potential for coolant channel flow blockage.

103. *Provide the calculated reactivity of the core in the following situations: (DRL)*

- (a) *Cold, unirradiated, beryllium and water reflected, no poison inserts.*
- (b) *Cold, unirradiated, beryllium and water reflected, poison inserts in place.*
- (c) *Cold, unirradiated, infinite water reflection, no poison inserts.*
- (d) *Cold, unirradiated, infinite water reflection, poison inserts in place.*
- (e) *Cold, spent, xenon free core with reflector and poison situations of (a) through (d). Account for effects of uneven fuel burnup on the worths of the reflectors and poison inserts at the end of core life.*

Answer:^{3,4} Questions 103 (a) and (b) are answered in Part 1, Appendix A, Section A-III.

Questions Nos. 103 (c) and (d) are no longer relevant because only one fuel element of the HFIR-type core will be moved at any given time, and not the entire core. However, as noted in Part 1, Section VI-C-1, even in the absence of control plates the reactor would be subcritical with full water internal and external reflectors (no beryllium). Thus, in particular, each separate fuel element would be grossly subcritical under these conditions. At the end of core life, the reactor would be essentially critical at operating conditions, and with equilibrium xenon and samarium, when all control rods are withdrawn. At this point, the estimated total reactivity loss due to temperature, xenon, and samarium is 5-1/2%. Argonne has not yet calculated the variation of control-plate worth during the core cycle, but it is apparent that the reactor is adequately controlled by the system of 5 control plates throughout core life (see Part 1, Appendix A, Section A-III).

104. *What is the history of reactivity losses from (a) cold to hot operating temperature, (b) fuel burnups, (c) no xenon to equilibrium xenon, and (d) equilibrium xenon to peak xenon? (DRL)*

Answer:^{3,4} Part 1, Appendix A, Section A-IV lists the estimated reactivity change corresponding to an increase in temperature from room temperature to operating level. It also includes the "equilibrium" reactivity effect of xenon (Xe^{135}) plus samarium (Sm^{149}) at 100 MW. Calculations reported in ORNL-3572* somewhat underestimate the reactivity life of the core. Argonne will perform independent computations of the reactivity history of fuel burnup; these calculations have not yet been performed for the now modified AARR, with the HFIR-type fuel.

Curve B in Fig. A-4 of Appendix A, Part 1, reproduced from ORNL-3572, is a plot of k_{eff} at 4 days after shutdown from equilibrium operation at 100 MW; k_{eff} is plotted as a function of the total time of operation at 100 MW. The motivation for Curve B is that during the first seven days of full-power operation the maximum value of k_{eff} is calculated to occur ~4 days after a shutdown from full power, assuming that xenon (Xe^{135}) has attained pseudo-equilibrium levels.* [Note: In Fig. A-4, the boron burnable poison is assumed to control 0.04 Δk . It is realized that this is not consistent with the value of 0.05 Δk shown in Part 1, Appendix A, Table A-2. See Part 1, Appendix A, Section A-VII, for further discussion.]

105. *Considering the thermal and hydraulic analysis, discuss the following:*

(a) *The design bases of the trip levels for safety system actions. What are the design margins at the trip levels or if more conservative, at the over shoot conditions associated with the transients which initiate the safety system action? (DRL)*

* F. T. Binford and E. N. Cramer, "The High Flux Isotope Reactor - A Functional Description," ORNL-3572 (Volume 1) (May 1964 - revised March 1965), p. 7-5.

Answer:^{3,4} Based upon the incipient boiling power level calculated by ORNL for the HFIR core (139 MW at end of cycle - see Part 1, Section IV-B-1, and Section II, Criterion 4.4) the trip level for reactor scram at high neutron flux/primary coolant flow ratio has been tentatively set at 1.3. With this setting, effectively the reactor will scram at 130 MW with full coolant flow. The trip level for rate of rise of neutron flux has been tentatively set equivalent to a power level increase rate of 20 MW/sec. See Part 1, Appendix C, Section C-IV for a discussion of the effectiveness of the reactor safety systems in preventing core damage, using these trip settings.

(b) *The hydrodynamic stability margins and the ratio of critical to maximum heat flux of the hot channel for the safety system trip levels, a complete loss of flow accident, and the sequential loss of flow accident. (DRL)*

Answer:³ See Part 1, Section VI-D-1

(c) *The burnout design margins for normal transients and those accidents for which reverse flow can occur in the reactor vessel. As a part of the discussion, provide the ratio of critical to maximum heat flux as a function of time. (DRL)*

Answer:³ See Part 1, Section VI-B and Section VI-D-1e.

106. *Discuss the basis for applying single channel data to parallel channel systems. (DRL)*

Answer:^{3,4} The HFIR method of steady state thermal analyses used the inception of boiling at the surface of the fuel plate to set a burnout power level. This type of analysis does not cause concern over parallel channel effects, and therefore this question is not considered applicable for the HFIR core. See also Part 1, Appendix C, Section C-II.

107. *Discuss the potential for and the consequences of failure in the inner or outer beryllium reflectors resulting from either heat generation, loss of cooling, or a combination of both. (DRL)*

Answer:^{3,4} The tentatively planned settings for setback and scram on high flux/flow ratio (1.0 for setback; 1.3 for scram - see Part 1, Section VI-D-1; see also the reply to question 105 (a), above) will effectively limit the heat generation rates in the beryllium reflectors. Adequate excess coolant is available for removing heat at these generation rates. See Part 1, Section VI-D-1d for a discussion of loss of reflector cooling.

108. *What data is available to establish that stainless steel clad europium oxide control rods will operate as required in the AARR? Include consideration of dimensional stability with irradiation, flow, and high heat generation rates in the control rods. (DRL)*

Answer:^{3,4} AARR will now use aluminum-clad europium oxide control plates identical to those now in use in HFIR (see Part 1, Section IV-B-2). Since operating conditions in AARR will be identical to those in HFIR, the successful operating history of the plates in HFIR will serve as evidence that satisfactory operation in AARR can be expected without further development and test.

109. *What preliminary hydraulic and thermal stress analyses have been performed on the structures inside the pressure vessel? Include but do not necessarily limit your discussion to control rod channels, the quick opening head closure, and flow dividing structures. (DRL)*

Answer:⁴ ORNL reports* that mechanical and thermal stresses in the HFIR core support structures were analyzed and were found to be well within the elastic limits. These structures have now been used in HFIR at design power level. The AARR will utilize most of these designs intact. Structures differing from HFIR will be analyzed during Title-II.

110. *Describe the material surveillance programs planned to be followed during operation of AARR, and discuss the extent to which design features are modified to accommodate these programs. (DRL)*

Answer:² Although the possibility of significant radiation damage to the vessel at any point appears remote, a surveillance program will nevertheless be instituted. The program will consist of an accelerated irradiation of specimens in AARR, at approximately one order of magnitude greater flux than expected at any point in the vessel wall. It is expected that the vessel wall adjacent to the neutron window (see Revised Section IV-D dated September 28, 1966, page IV-D-15 for discussion of neutron window and expected neutron fluxes) will be exposed to the highest fast neutron flux.

Irradiation surveillance samples will be prepared from at least several types of material: (1) from plugs removed from the vessel for the beam tube nozzles, (2) from beam tube nozzles, and (3) from vessel rings accompanying the fabrication of vessel shell sections. These specimens will be irradiated in the water region between the beryllium reflector and vessel wall. Other specimens representing the wall material will be irradiated at a lower rate, preferably in the region between the neutron window and the vessel wall.

It is considered possible that the neutron flux in the region of the beam nozzles might exceed that adjacent to the window due to streaming through the tube. This possibility is currently under investigation. If the study reveals a more severe problem in this region than at the neutron window, the surveillance program will be altered to account for the higher exposure; that is, the fast neutron flux at the nozzle region will be used as the basis for determining the irradiation rate, and consequently, the life of the vessel.

* F. T. Binford and E. N. Cramer, "The High Flux Isotope Reactor - A Functional Description," ORNL-3572 (Volume 1) (May 1964 - Revised March 1965), page 5-5.

111. With respect to the fuel handling systems:

(a) Discuss and illustrate as necessary the systems that will be used to carry out the design criteria established for the Fuel Handling System. Indicate the mechanical features and the controls and instrumentation that will be used to assure the safe handling of new and spent cores. (DRL)

Answer:³ See Part 1, Section IV-C.

(b) What fuel handling procedures are to be used if the spent core contains failed fuel plates or assemblies? (DRL)

Answer:³ See Part 1, Section IV-C-3.

(c) Provide an analysis for accidental dropping of a spent core including consideration of (a) physical damage to the core, (b) breaking of control blade locks, (c) impairing heat transfer capability by laying the core on its side, and (d) consequences of fuel melting outside of containment, including an evaluation of potential doses in the control room. (DRL)

Answer:^{3,4} See Part 1, Section VI-C-2. Note that with the HFIR core now to be used in AARR, control plates are not removed with the fuel. Also, all fuel handling except ultimate shipment is carried out within containment; therefore fuel melting outside containment is not considered credible.

(d) Evaluate the consequences of dropping a single fuel assembly in its worst position during core dismantling. Consider both impact damage and impaired heat transfer. (DRL)

Answer:^{3,4} See Part 1, Section VI-C-2. Note that in the present design, no dismantling of fuel is carried out at the AARR site except for the individual handling of the two fuel elements constituting the core during loading and unloading operations.

(e) Discuss the consequences of (1) a total loss of canal cooling capability over an extended period of time, and (2) a total loss of canal water. (DRL)

Answer:³ See Part 1, Section VI-C-2 and Section VI-C-3.

(f) Provide a safety evaluation for the Reactivity Measurement Facility (RMF). Will this facility be restricted to operation with clean cores? If not, consider potential accidents with spent cores in the safety evaluation. (DRL)

Answer:² No Reactivity Measurement Facility (RMF) is provided on-site in the AARR Title-I design. Any physics measurements required on new cores prior to their use in AARR will be made in another facility on the ANL site, or perhaps at a subcontractor's site. In the former case, a separate safety analysis on the ANL facility will be submitted, according to current regulations, at the time authorization is sought to perform the required measurements. In the latter case, the subcontractor will have the responsibility for obtaining the necessary safety approvals to perform the necessary measurements in his facility. At present there are no plans for making physics measurements on spent cores or on cores which have received significant exposure in the reactor.

(g) *With regard to the RMF and the transfer and storage of spent cores, provide justification for not making the Fuel Handling Building a part of containment. (DRL)*

Answer:² As stated in the responses to questions 15 and 111 (f) above, plans for a RMF within the AARR complex have now been eliminated, and all transfer and storage of spent cores will be carried out within the Reactor Containment Building. The Active Material Handling Building (formerly the more extensive Fuel Handling Building) will now house a pool system whose functions are (a) the underwater loading of individual spent fuel elements into shipping casks, and (b) the loading and unloading of samples into the reactor hydraulic rabbit system. During fuel element cask loading operations, appropriate safeguards such as physical separation of elements, control of the number of elements permitted in specified canal regions, and monitoring will be administratively enforced to render incredible the possibility of a fuel criticality accident in these pools. The nature of samples to be exposed in AARR rabbit facilities (see response to question 118 (a), below) effectively precludes the possibility of a criticality accident by sample accumulation and interaction. This restricts credible accidents in the Active Material Handling Building to release of contamination from the surfaces of spent fuel elements in loading, from the surfaces of exposed rabbit samples, and from an occasional damaged rabbit sample. Such releases can be very adequately handled by the confinement system provided, wherein reasonable precautions are taken to seal building openings against gross leakage, and an adequate ventilation system is provided to maintain the building at a negative pressure with respect to the surrounding atmosphere at all times and to discharge the exhausted building air to the 250-ft exhaust stack.

112. *With respect to the waste disposal systems:*

(a) *What criteria will be applied in the design and construction of the radioactive liquid waste storage tanks and the gaseous waste holdup tanks, such as codes, design margins, testing, leakage-rate specifications and maximum allowable radioactive inventory? Provide an evaluation of the consequences of failures in each of the waste systems. (DRL)*

Answer:² The suspect waste retention tanks are used to retain and monitor suspect wastes in order to dispose of them properly. The discharge from these tanks normally has no activity or else is of low level. These tanks shall be of all welded, lithcote (LC-73) lined, steel construction. Tank conditions are atmospheric pressure at 150°F. Design conditions shall be 15 psig at 175°F. The tanks shall conform to the ASME Unfired Pressure Vessel Code. The tank lining shall be tested by the Tinker and Rasor method using a Type 1-M detector to detect holidays which will be eliminated.

The active waste storage tanks are used to retain all known active liquid wastes. The discharge from these tanks is normally of low level but could be of intermediate level. All tank effluent is pumped to tank trucks which bring it to the Reclamation Building for treatment and disposal. These tanks shall be horizontal of either all welded, lithcote (LC-73) lined, steel construction or stainless-steel-clad carbon steel construction. Tank conditions are atmospheric pressure at 175°F. Design conditions shall be 15 psig and 200°F. The tanks shall conform to Class C, Section III of the ASME Boiler and Pressure Vessel Code. If lithcote lining is used it shall be tested by the Tinker and Rasor method using a Type 1-M detector to detect holidays which will be eliminated.

Gaseous waste holdup tanks have been eliminated from the facility, and consequently no design criteria for such tanks are necessary.

Both the suspect waste retention tanks and the active waste storage tanks are located in underground concrete vaults. The floors and walls of the vaults up to the capacity of the tanks are lined with decontaminable coatings. In the event of tank rupture or leakage it is expected the vaults will hold the liquids until they can be removed and the areas decontaminated.

A tentative maximum allowable radioactive inventory that may be stored in the radioactive liquid waste storage tanks is about 0.5 mc per liter and is based on the present capacity of Reclamation Building evaporators.

(b) *Indicate the design features and criteria which will assure that the systems cannot become involved in reactor incidents. (DRL)*

Answer:² The reactor is within containment while the waste systems are outside of containment. Normally-closed isolation valves separate the waste storage system from the containment building. Those valves would be open only when hot wastes are pumped to the storage tanks.

113. *With respect to the reactor pool:*

(a) *Describe each below surface penetration in the pool with a statement as to the purpose of each penetration. (DRL)*

Answer:² With the exception of penetrations in the Reactor Pool for the primary coolant pipes, reactor vessel, and the experimental facilities, all pools are free of subsurface penetrations.

(b) *Discuss the pool water makeup system design criteria and their bases. (DRL)*

Answer:² Makeup water for all pools and canals within the containment is made to the containment pool level tank of the sweep flow purification system. Makeup, obtained from the demineralized water system, is normally furnished at a rate of 40 gpm. This rate is the maximum capacity of the demineralizer units. However, rates as high as 60 gpm can be furnished for a short period if both demineralized water transfer pumps are operated, both containment sweep flow circulating pumps are on, and water is obtained from the 30,000 gallon demineralized water storage tank.

The makeup capacity is considered more than sufficient to compensate for normal leakage from pumps and equipment, surface losses, and evaporation. Makeup of excessive losses from the Reactor Pool and Transfer Canal within the containment is not considered necessary.

(c) *Provide an analysis of potential consequences for a total loss of pool water, including effects on experiment facilities. (DRL)*

Answer:³ See Part 1, Section VI-C-3.

114. *With respect to electrical power systems:*

(a) *Provide preliminary schematics in sufficient detail to show that a single fault or the failure of a single component or section of the electrical distribution system will not result in the loss of vital loads. (DRL)*

Answer:² Vital equipment in AARR may be divided into two categories: (1) emergency equipment installed as backup for normal operating apparatus, and (2) normal operating equipment installed in a redundant arrangement.

For the most critical processes, both systems are used. An example of the first category is the d-c emergency air compressor, which will be needed if the normal power supply fails. An example of redundancy is the reactor safety system instrumentation where two-out-of-three scram logic is incorporated. Failure of power to two channels will cause a scram but will not jeopardize the facility. Backup is used on the primary water system and secondary water system where pony motors are mounted on top of each of the main pump motors.

The attached figure is a simplified single line diagram which shows the method of power distribution within the facility. If a fault occurs in an inverter or on a bus marked A through I, there will be a power interruption to vital equipment supplied from the faulted component. However, this does not create a hazardous condition.

If a fault should occur in an inverter supplying power to an instrument bus, the power will be interrupted to one instrumentation bus, and the reactor will be scrammed. The second inverter continues to supply power to the other instrumentation bus and all controls and instruments connected thereto will continue to operate. Provisions will be made to prevent a tie circuit breaker from closing in on a faulted bus.

In the case of the d-c bus which supplies power to the pony motors and other vital motors, a fault on the bus will cause an interruption of power to the motors, but will not bring about an unsafe condition. The flow in the primary and secondary systems will remain essentially unchanged because the pony motors furnish power for a small fraction of the total flow when normal power is available to the main pump motors. However, when a fault occurs on the d-c bus, the reactor will be shut down because the redundancy of the backup system would be lost.

In the case where utility power is lost to the main pumps, the reactor is scrammed and the flow decreases to the emergency flow rate in a short period of time. A single pony motor is capable of maintaining that emergency flow rate with power from Class-I and Class-II supplies.

It is therefore concluded that a single fault on either the utility system or the emergency systems will not bring about an unsafe condition at the facility.

(b) *Provide a listing of emergency loads and verify that emergency generators and station batteries will have sufficient capacity to assume these loads for as long as may be required. What criteria will be applied in design of those systems which switch, connect or shed loads under conditions when power is required from the emergency generators? What is the sequencing and timing of the events which must occur? (DRL)*

Answer:² The emergency loads are as follows:

<u>CLASS-I LOADS</u>		
<u>250-VOLT D-C BUS DA2*</u>		
	H.P.	Amps.
1. Emergency Pressurizing Pump	15	55
2. Primary Pump Emergency Pony Motor	10	38
3. Emergency Air Compressor	3/4	3.7
4. Secondary Emergency Shutdown Pump	5	20
5. Poison Injection Pump	5	20
6. Control Rod Drives (6 @ 1/2 H.P.)	3	15.6
Total	38-3/4	152.3

* Bus DA1 has the same H.P. loads as Bus DA2 less the 15 & 3/4 H.P. Motors.

Amperes Required for 1/2 Hr. = 152 (Exide Battery Data Sect. 58.00)

Type EMP Battery Rated 28 Amps/Pos Plate for 1/2 Hr.
to 1.75 Volts Per Cell $152/28 = 5.44$

Use Battery With 6 Pos Plates Type EMP 13 Battery

Amperes Per Cell = $6 \times 28 = 168$

Reserve Capacity $168 - 152 = 16$ or 10.5%

Use 120 Cells for 250 Volt Battery Bank

Battery Bank Rating on 8-Hr Rate

$8 \times 30 = 240$ Ampere Hours

From the single line diagram it can be seen that all Class-II Loads are supplied from MCC-EM-1 & EM-2 or Power Panels CA1 & CA2.

Supplied from Class-II Switchgear EA1:

<u>MCC-EM-1</u>	<u>Connected KVA</u>	<u>Demand KVA</u>	<u>Standby KVA</u>
Reactor Clean-up Pump P-03-1	20	5	15
Reactor Hot Waste Sump Pump #1 P-22-5	3		3
Gas Rabbit Blower #1 P-28-1	3	3	
Gas Rabbit Blower #2 P-28-2	3	3	
Reactor Hot Water Sump Pump P-22-6	3	1	2
Ltg. Transf. (Reactor Service Fl.)	15	10	5
Emerg. 480-V Ltg. Panel	30	20	10
Emerg. Power Transf.	30	20	10
Control Room Press. System Fan SF-3	1	1	
Battery Rooms Exhaust Fan EF-13	1	1	
Control Room A/C Supply Fan SF-2	5	5	
Control Room A/C Return Air Fan EF-8	1-1/2	1-1/2	
Relay Rack Room Exhaust Fan EF-9	1	1	
Diesel Gen. Supply Fan SF-10A	3	3	
Diesel Gen. Supply Fan SF-10B	3	3	
Motorized Bridge Crane	<u>20</u>	<u>5</u>	<u>15</u>
Total	142.5	82.5	60.0

Supplied from Class-II Switchgear EA2:

<u>MCC-EM-2</u>	<u>Connected KVA</u>	<u>Demand KVA</u>	<u>Standby KVA</u>
High Press. Water Syst. Pump P-18-1	10	10	
Radioactive Cell Exhaust Fan EF-21	1-1/2	1-1/2	
Hot Water Pump Heating	2	2	
Lab Sewage Pumps Duplex	4	2	2
Toilet Sewage Pumps Duplex	10	5	5
Hot Water Pump Reheater	3	3	
Emerg. Ltg. Panel (Labs)	15	10	5
Radioactive Lab. Exhaust Fan EF-18	5	5	
Reactor & AMH Bldg. Sump Pumps - Duplex	3	1-1/2	1-1/2
Lab and Office & Elect Bldg. Sump Pumps - Duplex	1-1/2	3/4	3/4
Radioactive Waste Facility Sump Pumps - Duplex	2	1	1
Emerg. Power Transf.	<u>30</u>	<u>20</u>	<u>10</u>
Total	87.0	61.75	25.25

<u>Class-II Power Panel CA1</u>	<u>Connected KVA</u>	<u>Demand KVA</u>	<u>Standby KVA</u>
Instrument Panel Transf. IP1	10	0	10
Instrument Batt. Chgr.	20	5	15
Batt. Chgr. DA1-1	50	3	47
DC Control Center DA1			
Sec. Emerg. Shutdown Pump P12-1	5	0	5
Pri. Emerg. Shutdown Pump P02-1	10	0	10
DC Valve Panel	2	1	1
Control Rod Drives	<u>10</u>	<u>0</u>	<u>10</u>
Total	107	9	98

<u>Class-II Power Panel CA2</u>	<u>Connected KVA</u>	<u>Demand KVA</u>	<u>Standby KVA</u>
Instrument Panel Transf. IP2	10	0	10
Instrument Batt. Chgr.	20	5	15
Batt. Chgr. DA2-1	50	3	47
DC Control Center DA2			
Switchgear Cont. Panel	2	1	1
Emerg. Press. Pump P05-3	15	15	
Pri. Emerg. Shutdown Pump P02-2	10	10	0
Sec. Emerg. Shutdown Pump P12-2	5	5	0
Poison Inj. Pump P06-2	5	5	0
Emerg. Air Comp.	0.75	0.75	0
Control Rod Drives	<u>10</u>	<u>6</u>	<u>4</u>
Total	127.75	50.75	77.0

CLASS-II LOADS
480-VOLT SYSTEM

<u>Summary</u>	<u>Connected Load, KVA</u>	<u>Demand Load, KVA</u>
MCC-EM-1	142.5	82.5
MCC-EM-2	87.0	61.75
Panel CA1	107	9
Panel CA2	<u>127.75</u>	<u>50.75</u>
Total	464.25	204.0

Emergency Load	204 KVA
Battery Recharge	<u>47 KVA</u>

Total Load 251 KVA

Use two 300-KW diesel-generators KVA Rating = 375 KVA, 480-Volt 3 Ph. 60 cycles.

Spare Capacity Per Generator

$$\frac{375 - 251}{375} = 33.0\%$$

Without jeopardizing the safety of the facility, the design criteria for the emergency generator switchgear shall utilize industry tested, off-the-shelf equipment to the maximum extent possible.

Upon loss of normal utility electrical power all circuit breakers on Class-II switchgear units EA1 and EA2, including the generator breakers, would open. Simultaneously the diesel-engines would receive a signal to start. When the generators reach rated speed and voltage, the generator breaker closes, automatically connecting the generator to its respective switchgear unit. The tie circuit breaker remains open. As soon as power is available at the switchgear bus, the circuit breakers supplying power to the Class-I equipment and the emergency motor control center are closed automatically. The tie circuit breaker remains open and the generators operate independently of each other. Each generator is sized to carry the total emergency load of the facility. No timing or sequencing is required in picking up emergency loads when both generators start, since the generators cannot be overloaded by the connected load.

In the event one diesel-generator does not start, there would be a time delay involved in picking up the loads connected to the faulty generator. Total time allowed for each diesel generator to start is approximately two minutes. At the end of this time an alarm is sounded in the control room. The reactor operator would then send a man to the emergency switchgear to close the tie circuit breaker and pick up the remaining loads. The estimated maximum total elapsed time for this sequence of events would be five minutes.

Equipment necessary to keep the plant in a safe condition is provided in a dual arrangement. Two half-size units, or in some cases two full-size units, are installed. One-half of each system is connected to generator G1 and the other half is connected to generator G2. During emergency operation, the equipment receiving power from one generator is adequate to maintain the facility in a safe condition. Hence, it may be desirable that the load connected to the faulted diesel-generator be connected to the running diesel, as noted above, but it is not a requirement for maintaining the plant in a safe shutdown condition.

(c) *Analyze the consequences of a total loss of electrical power, including emergency power, in the plant. How long can such a power loss be sustained without fuel damage? (DRL)*

Answer:³ See Part 1, Section VIII-E.

115. *Provide sufficient preliminary design information including block or schematic diagrams for nuclear and process instrumentation and controls and the control rod drive system to verify that the following criteria will be met:*

(a) *No single component or circuit fault should result in loss of automatic protection against any reactor accident. (DRL)*

Answer:² The attached drawings are pertinent to one or more of the answers to this question.

Each plant parameter that has a reactor scram function is monitored by three independent redundant instrument channels. These channels are connected to the scram magnet circuits via two-out-of-three coincidence logic. The electrical power for these instruments is obtained from two redundant independent continuous power (Class-I) a-c buses. Each instrument channel in a redundant group is connected to a separate branch circuit. The branch circuits for any redundant group are not all connected to the same bus. Several channels, each monitoring a different parameter, may be fed from the same branch circuit.

Thus a single component failure or circuit fault within an instrument channel can at most result in disabling that particular channel. The remaining two channels in the affected redundant group will still be capable of automatically shutting down the reactor.

The circuit logic through which the individual channel output signals are interconnected to control the safety-rod magnets will also employ redundancy, so that again a single component failure or circuit fault cannot result in loss of automatic protection. This redundancy will be in the form of three independent series circuits each of which will control the current in one of the three windings in each safety rod magnet. If any two of the three windings are de-energized, the particular safety rod is released. Consequently, a circuit fault that results in directly connecting the magnet power supply to one winding does not result in loss of the capability for scrambling the rod.

NUCLEAR INSTRUMENTATION
TYPICAL BLOCK DIAGRAMS

NOTES:

1. P1, P2 & P3 INDICATE SEPARATE SOURCES OF POWER.
2. (A) - CURRENT INDICATOR
3. K_{1A}, K_{1B} & K_{1C} ARE CHANNEL TRIP RELAY CONTACTS FROM A GROUP OF THREE REDUNDANT SAFETY INSTRUMENTATION CHANNELS; K_{2A}, K_{2B} & K_{2C} ARE FROM A SECOND GROUP, ETC.
4. F.A.S. IS A FAST ACTING SWITCH CONTROLLED BY "FAST SCRAM" VARIABLES
5. R - RESISTOR FOR TESTING

ROD SCRAM CIRCUITS
BASIC ARRANGEMENT

(b) *No single component failure or circuit fault can both initiate an accident and reduce the degree of protection. (DRL)*

Answer:² The safety system is designed to protect the reactor against various hazards, among which are failures of the control system that could initiate an accident. The ability to cope with this particular hazard is achieved by complete separation of the circuits and components of the safety system from those of the control system. Where a power source is common to both safety and control channels, its voltage will be measured automatically. If the voltage approaches a value that could result in improper but on-scale operation of the connected control or safety instrumentation, a trip signal will automatically be generated for those safety channels.

(c) *Sufficient initial and periodic testing of systems and components must be conducted to assure that the proper operation of that component or system can be anticipated when it is required (including voltage, temperature, and humidity tests over the maximum permissible limits). (DRL)*

Answer:² The program for testing the safety system can be divided into two distinct categories: initial pre-operational testing and periodic on-line testing.

With the exception of item (3) below, the initial testing procedure will be completed prior to achieving reactor criticality. This portion of the testing program can be divided into three separate parts:

- (1) safety system instrumentation specification requiring certified test results by the manufacturer (16.4),*
- (2) an in situ control room demonstration of capability for proper operation at elevated temperatures (16.5),
- (3) observation of representative temperatures in the vicinity of transducers located in non-air-conditioned environments during initial plant operation with appropriate corrective action if necessary (16.6).

The safety system instrumentation will be specified for continuous operation in any combination of environmental conditions falling within the following extremes:

	Minimum	Maximum
Ambient temperature	5°C	50°C
Relative humidity	-	90%
D.C. Power Source		
Voltage	105 Volts	140 Volts
A.C. Power Source		
Voltage	105 Volts	125 Volts
Frequency	57 HZ	63 HZ

*Numbers in parentheses refer to applicable Safety and Design Criteria in Part 1, Section II.

The maximum control room temperature during the in situ test demonstration is expected to be about 110°F (43.3°C). After the initial test procedures have been completed, they will be repeated only if significant system design changes are made or if major packages of new equipment are installed.

Periodic on-line trips and calibration tests will be carried out on each instrument channel monitoring a scram variable. Whenever a channel is made inoperative during a test, it will automatically generate a trip output signal. Thus, in a two-out-of-three configuration, tripping of either of the remaining two channels will initiate a reactor scram. For each channel monitoring a scram variable, the scheduled testing procedure will have a periodicity that will make the probability of the simultaneous existence of two random undetected faults negligibly small. The calculation of the interval between tests will be based on conservative estimates of the number of failures per channel per year, and the mean time to repair a failure. If a failure is detected during a test or by observation of the instrumentation during normal operation, the defective channel will be tripped manually. Either of the remaining two channels in a two-out-of-three configuration may scram the reactor if an unsafe condition should arise.

Rod-drop tests will be performed with each new core or rod set. Rod-drop tests will also be performed if the reactor has been shut down for an extended period of time.

(d) *The intentional removal of a component because of failure or testing should not decrease the degree of protection afforded. (DRL)*

Answer:² All of the instrument channels that are part of the reactor safety system and which initiate scram for an excessively high or low value of the measured variable will also be automatically tripped for an apparent extreme value at the opposite end of the range (16.2). As a result, the removal or disconnecting of most components, including power supplies, signal cables, various amplifiers, etc., will be automatically detected and the affected channel tripped. In addition, all plug-in modules will be provided with interlocks that will trip the channel if a module is removed and the circuit continuity interrupted. The fail-safe principle is followed throughout in the design of the reactor safety system ensuring that disconnection or removal of channel trip relays, scram relays or any wires in their holding circuits, will result in interrupting appropriate safety circuits. Online test and calibration operations will be effected by (1) adding a test signal to the plant-derived signal in the direction of channel trip, (2) separately testing "opposite-end-of-scale" trips without disturbing primary trips, or (3) tripping the channel as an integral part of the calibration signal injection operation.

116. *For those transients, which rely on the reactor safety system to protect the reactor from damage, discuss the required safety system response times and the basis for these requirements.*
(DRL)

Answer:³ See Part 1, Appendix C, Section C-IV-B.

117. *Analyze the probability and consequences of the central control room becoming uninhabitable or ineffective due to a fire (or other disaster) in the control room or cable spreading area.*
(DRL)

Answer:² The control room, relay rack room and cable spreading room are located below grade on the service floor of the Laboratory and Office Building, and are of reinforced concrete construction. Equipment permanently installed in these rooms will be of fire-resistant construction. Control devices will be in metal-enclosed cabinets and all wire insulation will be flame resistant. Smoke detectors will be installed in each of these areas. Fire alarm annunciators will be located in the control room, main corridor and at the ANL site fire station. Portable carbon dioxide fire extinguishers will be installed inside each of these rooms and in the adjacent corridors.

Whenever there is a core in the reactor vessel, the control room will be manned by reactor operators. Maintenance personnel will also be on duty in the area at all times. In addition the Laboratory Fire Department periodically patrols all buildings to ensure that combustible materials do not accumulate.

With the above-mentioned type of construction, installation of protective devices and surveillance of the area, the probability of a fire starting, getting out-of-hand and forcing abandonment of the control room is very low. Such an event would require several very unique conditions to occur simultaneously.

In any event there would be a time lapse from the detection of the fire by the smoke detectors to the abandonment of the control room. During this time the reactor would be scrammed.

The design of the control room, relay rack room and cable spreading room will be such that a tornado or earthquake will not affect their normal functions. Also, in the event toxic or noxious fumes are brought into the control room through the outside air intake, emergency corrective action could be taken, such as the use of the emergency alternate air intake or the use of air-packs by occupants. Any other credible disaster would require a certain length of time to develop and during this time emergency corrective action could be taken.

Forced abandonment of the control room after the reactor has been scrammed would not create an additional hazard to equipment or personnel. The continuity of cooling of the shutdown reactor does not depend on an operator being in attendance.

118. *With respect to the experiment programs contemplated for the facility, provide the following information:*

(a) *Materials to be irradiated, their state and types of containment. (DRL)*

Answer:² In general a goodly proportion of the isotopes in the periodic table are conceivable as target materials or can occur as impurities in encapsulating materials. However, the primary experimental emphasis for ITC irradiations will be on isotopes with mass 244 or higher. Typical heavy element samples may be milligram quantities of ^{246}Cm , ^{252}Cf , etc. The physical state of such samples would be a dry solid (metal, salt or oxide). The reflector rabbits and thimbles will be utilized for irradiations which are experimentally profitable at the available fluxes. Typical reflector samples will be generally in the milligram to multi-gram range. These samples will be in the solid state and will tend to come, in general, from the middle of the periodic table.

Aluminum (typically type 1100) will be the primary encapsulating material. The target materials will be contained by the welding shut of the outer capsule. Particularly hazardous or corrosive samples will be evaluated on an individual basis and may require double encapsulation. Zirconium is another possible capsule material but is less desirable for post-irradiation separations chemistry. All experimental samples will be checked, evaluated, and leak tested for containment integrity and heat transfer characteristics on an individual basis.

(b) *Types of coolant loops to be considered, i.e., re-entrant loops, cryogenics facilities, liquid metal or gas loops, etc. (DRL)*

At the present time there are no contemplated experiments requiring in-pile cryogenic facilities (which would be prohibitively expensive in any case) or liquid metal loops. However, if in the future such an experimental requirement would arise, the specific experiment and associated equipment will be thoroughly evaluated and reviewed from a hazard and safety standpoint prior to approval for insertion in the reactor.

The only contemplated experimental facilities which are of the "loop" type are the hydraulic and gas (helium) rabbit systems which terminate in both ITC and reflector thimbles. The hydraulic rabbits utilize the primary reactor cooling water system for insertion and removal of irradiation capsules and to provide cooling during irradiation. A pressure (and water) change lock is provided close to the reactor vessel penetration to minimize the amount of high pressure piping external to the vessel. A separate low pressure transfer water system transports the experimental sample capsule - through interlocked containment integrity valving - between the reactor vessel and the Rabbit Pool loading stations located in the Active Material Handling Building.

The gas rabbit systems are essentially re-entrant helium loops. The helium flow provides both movement of the sample and cooling during irradiation.

(c) Criteria for requiring pressure, temperature and other process instrumentation and safety action. Criteria for design of experiment safety circuits including consideration of bypasses. Criteria for location and administrative control of experiment safety system consoles. (DRL)

Answer:² The instrumentation and control system for an experiment which is to be performed in or near the reactor will be reviewed from a standpoint of safety by two separate groups: the Experimenter Services Group and the AARR Facility Review Committee* as noted in the Part 1, Section V. In addition the Laboratory's Reactor Safety Review Committee will also review the experiment where required by standard Laboratory procedure.

The function of the Experimenter Services Group also includes assistance to experimenters in the design and operation of their equipment; planning, scheduling, and supervision of the installation of the equipment in the reactor; and maintaining safety surveillance over operation of the equipment. Thus, the group will review each experiment with respect to its operability as well as its safety. The safety and operability reviews will include but are not limited to the following items related to instrumentation and control:

1. Safety parameters	6. Power sources
2. Reactivity effects	7. Emergency procedures
3. Process instrumentation	8. Electrical controls
4. Nuclear instrumentation	9. Effects on reactor instrumentation
5. Operating procedures	10. Fail-safe design

With respect to the criteria for:

- 1) requiring the monitoring of certain experimental variables critical to reactor safety,
- 2) requiring reactor control and/or safety system action in response to those critical experimental parameters,
- 3) design of experiment safety circuits including bypasses, and
- 4) location and administrative control of experiment safety system consoles,

it can generally be stated that the reactor design and safety criteria delineated in Part 1, Section II will apply. This especially includes, but is not limited to, the following criteria: 1.3, 13.1, 15.1, 15.3-15.7, 15.9, 16.1, 16.3, 16.4.

"Experiment safety system consoles," as such, are not presently contemplated, although there will undoubtedly be local instrument panels serving particular experiment installations. Experiment instrumentation that is connected to the reactor safety system will not be readily

*The review includes but is not limited to instrumentation and control systems.

accessible to the experimenters for adjustment, and will be under the jurisdiction of the reactor operating organization.

(d) *Maximum reactivity of all experiments. (DRL)*

Answer:^{3,4} Introduction of aluminum target tubes plus a full loading of targets into the ITC is calculated to result in a net reactivity gain of 0.6%. This is the net result of: (1) the positive reactivity effect of the displacement of water by the aluminum target tubes and target diluent; and (2) the negative reactivity effect of the additional neutron poisoning by the target samples, assumed to be equivalent to a total of 1/2 gm boron in the ITC. If the target samples are more absorbing, the net reactivity effect will be smaller.

The beam tubes, through tubes, and most of the vertical irradiation facilities in the beryllium are located in the permanent beryllium reflector. They are isolated from the fuel zone by the control-channel region (~3.2 cm thick), and the removable and semi-permanent beryllium (~8.8 cm thick). It is conceivable that additional vertical irradiation facilities of small diameter may be provided in the semi-permanent beryllium. Argonne's present estimate is that the total reactivity loss due to the totality of vertical irradiation facilities (target tubes and targets) will be less than 2%. The beam tubes and through tubes are considered to be an integral portion of the beryllium and its replacement by beam tube materials and void therefore is not included. Introduction of samples in individual tubes is governed by Criterion 5.2 (Part 1, Section II).

119. *Evaluate the most severe consequences resulting from experiment failures. Consider metal-water reactions, thermite reactions, missiles, shock waves, radioactivity releases, reactivity transients, etc. (DRL)*

Answer:^{3,4} See Part 1, Section II, Criterion 5.2, Section VI-B-5, Section VI-G-2, and Section VII-C-4. Metal-water and thermite reactions for sample facilities are covered along with the core; such reactions for the samples themselves cannot be discussed at this time, but will be considered in evaluating proposed experiments for performances in AARR, per Criterion 5.2. See also Part 1, Section V-D.

120. *Discuss how the failure of experiments resulting from the maximum credible accident (MCA) defined in the PSAR can increase the consequences of the MCA. (DRL)*

Answer:^{3,4} The Sections of Part 1, referenced in reply to question 119 above indicate that no failure of an experimental facility can increase the consequences of MCA beyond those described in Part 1, Section VII. The evaluation of samples and experiments per Criterion 5.2 will eliminate any proposed experiments which do not also fulfill this requirement.

121. *Discuss the cooling requirements and the reliability of the cooling system for the experiment facilities. (DRL)*

Answer:³ See Part 1, Section IV-E-1, Section IV-E-4, and Section IV-G-1h.

122. *Discuss the design criteria, and design, of the control and instrumentation systems used for the reactor experiment facilities. (DRL)*

Answer:² The process systems associated with the experiment facilities are primarily concerned with cooling the facilities. All safety system instrumentation components and circuits connected to these process lines will be treated the same as the reactor safety system. In particular, they will be under the jurisdiction of the reactor operating organization and will not be accessible to the AARR users. This restriction includes both the adjustment of trip settings, etc., and the making or modifying of connections to the common reactor scram logic circuits.

The control and instrumentation systems for the reactor experiment facilities that are planned as a part of the initial construction package (i.e., beam tubes, rabbits, etc.) as well as control and instrumentation for future reactor experiment facilities will be designed in accordance with the safety criteria established in Part 1, Section II. Major criteria applicable to these systems are the following: 1.3, 4.5, 13.1, 15.3-15.6, 15.9, 16.3-16.6.

In addition to the above criteria, Criteria 21.1-21.3 cover the commercial utility, diesel-generator, and battery power sources.

123. *Discuss the normal and emergency sources of power for the experiment facilities (DRL)*

Answer:² Experimenter power distribution centers consist of one-480-volt, three-phase power distribution panel, one-30 KVA, 480-208Y120 volt transformer, and one-4-wire 120/208-volt distribution panelboard. Two experimenter power distribution centers spaced 180 degrees apart are located within the Reactor Containment Building. Each experimenter will have access to 30 KVA of 480-volt service and 7.5 KVA of 120/208-volt power. All of the above electrical service will be Class-III, utility service bus.

Emergency power of either the Class-I or Class-II category is not provided for the experimenters. However, Class-II power is distributed to some 480-volt receptacles on the experiment floor of the Reactor Containment Building. Experimenters can therefore be furnished emergency power as needed.

124. *What specific criteria will be used to limit the amount of toxic materials in the reactor in order to prevent the materials from significantly contributing to the consequences of an MCA? (DRL)*

Answer:³ See Part 1, Section II, Criterion 5.2.

125. *What is the basis for concluding that a regional or local hydraulic instability does not exist when the overall reactor characteristics appear to be stable? (DRL)*

Answer:³ See Part 1, Appendix C, Section C-II.

126. *With respect to Fig. VI-B-5, provide additional results for initial conditions of 1 Mw and zero power to support the contention that the worst transient accidents will be initiated at design power level rather than any other initial set of conditions? (DRL)*

Answer:³ See Part 1, Appendix C, Section C-IV-C.

127. *Provide the results of a startup accident assuming the rods are withdrawn in a bank at their maximum rate and scram does not occur. (DRL)*

Answer:³ See Part 1, Section VI-B-2.

128. *What reactivity excursions, considering both magnitude and rate, will lead to the failure of the following fission product barriers (starting from the worst set of initial conditions in each of the cases): (a) the beam tubes and internal thermal column experimental loops, (b) primary coolant system (excluding the reactor vessel), (c) reactor vessel, and (d) reactor containment building. (DRL)*

Answer:^{3,4} See Part 1, Section VIII-B and Appendix E. It should be noted that none of the fission product barriers mentioned will fail upon occurrence of any reactivity excursion considered credible, with a significant safety factor beyond the credible limits.

129. *What is the basis for the statement "that a conservative upper limit for the transient fission release is 100 Mw-sec" (Pg. VI-H-3)? (DRL)*

Answer:^{3,4} With the HFIR core, the conservative upper limit for energy releases resulting from credible reactivity insertions is 20 MW-sec. See Part 1, Section VI-B-3 and Section VI-B-4.

130. *What is the quantitative relationship between the magnitude of a metal-water reaction and the pressure pulse generated as a result of this reaction (Pg. VI-H-2)? (DRL)*

Answer:³ See Part 1, Appendix E, Section E-III.

131. *What would be the effect of a 43 psi pressure wave, from a postulated hydrogen explosion, on surrounding system components (Pg. VI-H-6)? (DRL)*

Answer:³ The question of hydrogen detonation has been re-analyzed. See Part 1, Section VIII-D.

132. *In accidents in which significant core melting is postulated, evaluate the additional metal-water reaction that may be initiated by the melting of core structures and sections of the pressure vessel. (DRL)*

Answer:^{3,4} In the realistic worst accident projected for AARR (resulting from a ruptured main primary coolant outlet line from the reactor vessel, outside the Reactor Pool) it is postulated that no more than the hot spots in the fuel (~5% of the fuel) would melt or undergo metal-water reaction. (See Part 1, Section VI-D-2b.) Under these circumstances, no melting of core structure or pressure vessel sections, or metal-water reactions involving these parts, would be expected. [Nevertheless a complete core metal-water reaction has been assumed for the computational model - see Part 1, Section VII-A.]

Hypothetical cases of complete metal-water reaction, following the incredible case of primary system rupture with simultaneous failure to scram, are discussed in Part 1, Section VIII-C.

133. *Describe the time sequence requirements for the ventilation flow to be reduced to 25% of normal capacity, and then to zero flow. In the postulated MCA, how much of the inventory would escape via this route and what would be the consequences, including assumptions. What would be the setpoints for this isolation system? (DRL)*

Answer:^{3,4} In the current facility design, there is no provision for an automatic reduction of Reactor Containment Building air ventilation to 25% of normal capacity. The only automatic reduction of air flow will be to zero flow and complete isolation of the RCB from the stack. A description of the ventilation and isolation system as now conceived is given in Part 1, Section IV-L. A discussion of associated radiation monitoring instrumentation is provided in Part 1, Section IV-H. An analysis of the basis for the containment-isolation system setpoints is presented in Part 1, Section VI-E-2. A detailed, conservative analysis of the radiation-dose contributions from stack release of activity following the MCA, in the time interval before the RCB is isolated, is supplied in Part 1, Section VII-C-2.

134. *Describe the emergency evacuation plan which justifies the assumption in the PSAR that evacuation within 15 minutes is possible. (DRL)*

Answer:² We can find no statement in the PSAR which implies that evacuation within 15 minutes is possible, or in fact, required. It is true that Section VII of the PSAR contains references to 15-minute doses, but these were given for information purposes.

The general question of evacuation procedures is covered in question 83, above. Although no time estimates were given there, it is felt by the Laboratory personnel responsible for site evacuation that 15 minutes is an upper limit on the time to evacuate the entire site population to off-site relocation areas. This judgment is based on the fact that the Laboratory rush-hour exodus is over in 15 minutes, and that with proper emergency coordination, this time could be reduced considerably.

135. *Reanalyze the 2 hour dose calculations for the MCA without taking credit for the lag time due to arrive at the point of interest. In the discussion of the results, compare the worst 2 hour dose with various 2 hour periods during the accident history. (DRL)*

Answer:³ See Part 1, Section VII-C-1.

136. *Could an autocatalytic positive reactivity insertion occur from fuel movement in any accident, including movement due to mechanical displacement, melting, and vaporization? If so, analyze the consequences of such an effect. (DRL)*

Answer:⁴ As yet, Argonne has not independently studied in detail mechanisms for and consequences of reactivity feedback effects due to movement of fuel in an accident. Both from analyses of the HFIR, by Oak Ridge National Laboratory, and analyses of the modified AARR, by Argonne National Laboratory, it is known that axial expansions contribute positive reactivity feedback, and that certain movements of fuel into the ITC could result in large reactivity gains. To prevent fuel from moving into the ITC, a strong inner side plate of the inner fuel element has been included.

Questions similar to this have been answered by Oak Ridge National Laboratory, for the HFIR. The answers provided apply also to the AARR. For convenience, both the questions and the answers are quoted below.

"Question No. 5, ORNL-CF-65-11-29* Provide an analysis of the reactivity effects resulting from possible fuel movement (e.g., effects of fuel melting and transport during an accident or relative movement of the two elements). Analyze appropriate accidents which could involve significant reactivity insertions or insertion rates from this source.

"Answer The HFIR core is undermoderated within the confines of the side plates but has positive void coefficients in the island and control region and nearly a zero coefficient between the two elements. As a result of the undermoderation within the fuel elements, removal of fuel plates (plus the associated fuel and burnable poison) increases reactivity. Calculations indicate that uniform removal of fuel plates from the core results in a maximum reactivity increase of $0.025 \Delta k$; the corresponding fraction of fuel plate removal is about 30% (one experimental point at about 3% fuel plate removal checks the calculations satisfactorily). If a complete loss of flow and/or sudden loss of pressure accident were to occur in which fuel plate melting resulted, it is perhaps possible that there would be some increase in reactivity due to molten fuel plate material dripping out the bottom of the core. However, the safety rods which would have scrammed as a result of the initiatory accident would be capable of maintaining the reactor subcritical even if the full $0.025 \Delta k$ addition were achieved, which, of course, is very unlikely.

* T. E. Cole to A. M. Weinberg, "High Flux Isotope Reactor - Safety Review Questions and Answers," Memorandum ORNL-CF-65-11-29 (November 12, 1965), p. 28.

"The amount of reactivity associated with longitudinal displacement of one fuel element relative to the other has not as yet been determined experimentally. However, since the water regions adjacent to the fuel elements already are over moderated, or nearly so in the case of the water gap between the elements, it is not likely that longitudinal movement of one element relative to the other would increase reactivity. Experiments of this type will be conducted in the near future.

"Relative movement of the two elements is normally prevented by the fact that the outer element is held in position by the inner shroud and the inner element is held in position by the target tower (see ORNL-3572,* Figs. 5.1.2 and 5.1.4).

"Questions Nos. 2E and 2F, ORNL-CF-65-11-29, Supplement No. 2**
(Question No.) 2E. If a local meltdown in a section of the core occurs, how large must the hole get before one stops gaining reactivity? How much reactivity is involved? What is the increase in power level at the edge of the region devoid of fuel, as a function of hole size? How great is the tendency to spread as a consequence? To what extent may such a mechanism lead to autocatalytic effects and transients starting in this manner or from other sources? Are there other autocatalytic effects from fuel movement?

"Answer A single local cavity might very well grow in size due to power density peaking, which can be as much as a factor of two greater than the local average. The maximum reactivity addition would be about 0.03% $\Delta k/k$ per spot, and the corresponding optimum hole size would be about 6 cm in diameter. A uniform removal of fuel plate would add reactivity in the amount of 2.6% $\Delta k/k$, and the corresponding removed volume would be 30%. It does not seem reasonable that this latter condition could be achieved by the former mechanism because the increased power density around a local "flux trap" tends to make the hole grow beyond the optimum and thus results in a decrease in reactivity.

"(Question No.) 2F. What pressures might be generated from local fuel melting? How much fuel failure is required to generate pressures which could move significant amounts of fuel inward (and thus gain reactivity), or which could interfere with control rod action? Does this allow for pressures from a "steam explosion" or a "water hammer"? How much reactivity could be gained by inward motion of the fuel into the flux trap? How much pressure would it take to produce such motion? Could flow blockage lead to such effects?

* F. T. Binford and E. N. Cramer, "The High Flux Isotope Reactor - A Functional Description," ORNL-3572 (Volume 1) (May 1964 - revised March 1965).

** T. E. Cole to A. M. Weinberg, "High Flux Isotope Reactor - Safety Review Questions and Answers," Memorandum ORNL-CF-65-11-20, Supplement No. 2 (August 15, 1966), pp. 17-18.

"Answer It does not appear that localized fuel melting would result in a steam explosion, and thus the accelerating pressures resulting from water and plate expansion, if an excursion were to exist, would be quite small. However, if several percent of the core were to be molten, it is quite possible that a steam explosion would occur. Because of the high pressures required to deflect the fuel element side plate significant amounts (see Appendix E)*, it is not anticipated that the tendency for inward movement of fuel would result in a significant addition of reactivity. It is more likely that movement of the safety rods would be impaired. Of course a steam explosion large enough to move significant amounts of fuel inwards or large enough to damage the safety rods, which are located beyond both the outer element, outer side plate, and the even stronger cylindrical shim-regulating rod, would disassemble the fuel element longitudinally. Thus it appears the pressure surges will only tend to shut the reactor down."

137. *Discuss the probability and consequences of a molten core melting through compartments below the core and then through the containment floor. (DRL)*

Answer:^{3,4} This effect is not considered credible in AARR. For a discussion of a hypothetical circumstance wherein this might occur, see Part 1, Section VIII-C.

* Note that this refers to Appendix E of the Reference document, and not of the AARR PSAR.

B. Questions from Reference (5)*138. *Criterion 5:*

There is no mention in the answer to question 3 above, of detailed criteria limiting the experimental program scope so that postulated accidents with the program do not result in more severe effects than the maximum accident considered. Previous discussions have indicated that these criteria are under development. What is their current status? (RDT)

Answer: A detailed criterion (Criterion 5.2) is given in Part 1, Section II.

139. *Criterion 9:*

The requested discussion under question 5 above, of the capabilities of backup shutdown system and the criteria for this system has not been provided. This is especially important in connection with the question of rupturing of the primary outlet line, and the operation of the redesigned emergency cooling system.

There is still some question on the reactivity worth of the backup shutdown system. Our position remains that the negative reactivity worth of this system should exceed the maximum loaded reactivity of the core. (RDT)

Answer: Criteria for the backup shutdown system are given in Part 1, Section II (Criteria 9.1 through 9.3). The system is described in Part 1, Section IV-G-1g, and further discussion on this system is included under question 5 above. The backup shutdown system is identical to the one used on the HFIR reactor, except that the poison solution inventory and speed of injection have been scaled up roughly in proportion to the increased primary system volume in AARR over HFIR.

140. *Criterion 10:*

The revised emergency cooling system described under question 7 above, does not provide adequate redundancy in either water supply or coolant loops. Utilization of the primary coolant loop and associated components (with the exception that motive power for the main circulation pumps is provided by pony motors mounted atop the main motors) is not appropriate for this important safety function. In the event of a "maximum credible accident" initiated by a rupture of a main coolant pipe anywhere outside the reactor pool, most of the primary system would be drained into the service floor of the containment building. The present emergency cooling system would be ineffective in removing decay heat from the core to prevent melting.

* Ed. note: Questions relating to certain ANL responses in Section III-A above. These questions were generated prior to AEC receipt of References (3) and (4). Responses to these questions were given in Reference (6).

There is still some question as to the ability of the original system to keep the core covered by water following rupture of one of that system's outlet lines. However, the original system, with its two independent emergency loops and separate inlet and outlet lines in the reactor vessel, is considered highly preferable, and appears more likely to meet current licensing criteria on redundancy. Discuss the reasons for having revised this system. Also discuss the reasons for eliminating criterion 10.3, regarding minimizing the possibility of uncovering the core in the event of a major rupture anywhere in the primary coolant system. (RDT)

Answer: The emergency cooling system is the same as that utilized on the HFIR reactor for an identical fuel-control rod-control drive system operating at the identical power level.

The decision to proceed with AARR utilizing a HFIR core system was based on the assumption that all those items basic to the reactor could be used as designed, built and successfully operated. It was of course recognized that the difference in locations required containment for AARR while containment was not required for HFIR.

However, the statements in question 140 that the HFIR system "does not provide adequate redundancy . . . is not appropriate for this important safety function," etc., are very serious statements in view of the fact that the HFIR is a relatively recently approved reactor which has been operating satisfactorily at its design conditions (including 100 MWt power level) for some time. If we cannot proceed on the premise that the HFIR system, with the addition of containment, will be approved as suitable for the Argonne site, we must know this immediately since our design, schedule, and cost estimate are based on this assumption.

We find it difficult to understand the discussion in question 140 regarding the ineffectiveness of the present (HFIR-type) emergency cooling system in removing decay heat from the core to prevent melting following a Maximum Credible Accident. The Maximum Credible Accident analyzed both in the original version (March 31, 1966) of the PSAR and in Part 1 above, assumes complete core melting. However, Section VI-D-2b of Part 1 indicates the events which might be expected with primary system ruptures in various locations, and shows that in cases of rupture outside the Reactor Pool (the worst accident we can realistically postulate) it is unlikely that more than 5% of the core (representing hot spot regions) would melt.

Criterion 10.3 was omitted from Reference (2), but is included in Part 1, Section II. (It should be noted that Part 1 supersedes Reference (2) in case of conflict between the two documents.)

141. Criterion 17, question 12 above:

Limits have not yet been set on the energy potential of experiments placed in the reactor so as to assure containment in the event that this energy were also released to the RCB during a MCA. (RDT)

Answer: Criterion 5.2 of Section II, Part 1 answers this question.

142. Question 18 above: see discussion under question 140. (RDT)

Answer: See response to question 140 above.

143. Question 46 above:

The basis for the conclusion that a localized hydrogen detonation would not damage the RCB walls is still not clear. (RDT)

Answer: See Part 1, Section VIII-D.

144. Question 50 above:

Provide further information showing that the range of a missile through the reactor vessel water would be very short. (RDT)

Answer: See Part 1, Section VI-G-1a.

145. *Comment on the fuel handling considerations discussed in questions 56 and 57 above. (RDT)*

Answer: Fuel handling techniques used in AARR and HFIR will be identical in principle. The fuel itself and the spent fuel and defective fuel storage facilities will be identical. Duplicates of the working ends of the HFIR fuel handling tools will be used in AARR; some changes in tool length may be necessary because of differences in headroom between the AARR and HFIR buildings. These differences arise primarily from the differences in function of the two facilities. See Part 1, Section IV-C.

146. Question 82 above:

The statement "Water samples will of course be taken ... during low power runs on approach to full power for the first time ..." should be changed to "Water samples will of course be taken ... during low power runs on approach to full power for the first time for each core ..." (RDT)

Answer: We disagree with the rewording of the statement given in response to question 82 above. With the HFIR experience and the check on the first core, there should be no need to check for dissolved hydrogen with every subsequent core.

147. Criterion 1.4 originally required that dynamic analyses of Class I structures be performed. The revised criterion says only that the need for dynamic analysis of each Class I component will be determined. Discuss the basis on which this need will be determined, and the reasons for changing the criterion from its earlier form. (RDT)

Answer: The basis requested is listed in the referenced section, viz., "all structures, systems, and components important to safety (Class I) will be designed such that the plant will maintain its capability of shutting down, removing decay heat, and being safely contained, and that malfunctions will not tend to increase the core reactivity." Stated in other words, the design will be such that the MCA will not be caused by an earthquake, but if a MCA should occur in conjunction with an earthquake, containment will remain effective. The criterion was changed from its earlier form since it appears that dynamic analysis of every Class I component may not be required to fulfill the above criterion.

148. *Also on Criterion 1.4, it is now stated that the maximum stresses produced by specified loadings resulting from the MCA will be "appropriately related" to the yield stresses of the materials. It was originally stated that these loadings would not exceed yield stresses. Discuss the reasons for this change, and the meaning of "appropriately related." (RDT)*

Answer: Our understanding of the containment criteria now being imposed is that if the MCA occurs simultaneously with twice the design earthquake loadings, containment functions must not be lost. (An earthquake having twice the severity of the design earthquake is referred to as the maximum earthquake.) We are not aware of a requirement that such an unlikely event must not distort anything. The words "appropriately related" mean in the manner recommended by our seismic consultant, modified as necessary to satisfy the DRL seismic consultant.

It should be noted that the original (March 31, 1966) Criterion 1.4 gave g-loadings for the design earthquake, and not the maximum earthquake referred to above. It is still our intention to design the containment for no yielding under the design earthquake loading plus MCA loading (see question 90 above).

149. *Discuss the "suitable provision ... for pressure accumulation" in the original Criterion 4.8, and the reason for its elimination. (RDT)*

Answer: During the Title I design phase (after submittal of the original PSAR), it became apparent that no reasonable pressure accumulation device could satisfy Criterion 4.8 if a rupture of a large pipe occurred. The new Criterion 4.8 (Part 1, Section II) takes note of this fact. Furthermore, an analysis of the consequences of a step reduction in pressure from the operating level to atmospheric (Part 1, Section VI-D-2b) shows that control plate insertion would be accomplished in such an accident probably before any fuel melting would occur, and certainly before sufficient melting could occur to hinder the insertion.

150. *Why have the emergency entrance and exit been eliminated from the reactor control room (Criterion 13.3)? (RDT)*

Answer: At the time of writing original Criterion 13.3, the control room was located on the first floor (at grade) of the Laboratory and Office Building, rather than on the service floor.

Further, with special shielding and ventilation of the control room, we now believe that personnel in that room can remain there long enough to ascertain such conditions as wind direction and to choose a safe surface evacuation route, rather than relying on a single emergency exit from the building which could lead to greater danger under certain circumstances.

151. *Referring to Criterion 15.2 as given in Reference (2):*

Is the operational ability of the emergency cooling system in any way related to scram or interlock actions? (RDT)

Answer: Criterion 15.2 as presented in Reference (2), is a partial list of the plant conditions causing automatic scram, and has no connection with the emergency cooling system except that the emergency cooling system operates regardless of the cause of a scram. Both pony motors as well as their power supplies will be continuously monitored for operability. Events following loss of primary pump pony motors are discussed in Part 1, Section VI-D-1a(4). There will undoubtedly be protection against reactor startup if the emergency cooling system is not functioning as required. Whether or not this protection will take the form of interlocks is a subject currently under study as part of the Title-II design effort.

152. *Why has the requirement that "all critical loads (from the safety viewpoint) be shared between the two separate electrical feeder systems" been eliminated from Criterion 21.1 as given in the draft version of the PSAR? (RDT)*

Answer: Our use of the words "critical loads" was incorrect in this case, since only Class-III (utility) power was involved. True critical loads are supplied with Class-I (uninterrupted) power.

C. Questions from Reference (7).*

153. *There is some information in the literature that indicates tornadic winds may approach velocities of 500 mph. What design capability, above that required to maintain the functional integrity of the containment, has been incorporated in the design to assure that the containment building and other vital structures will not fail in such a way (e.g., collapse or generate missiles) that the systems needed to shutdown and maintain the reactor in a safe condition would be rendered inoperable in the event of a tornado of this intensity?*

Answer: The design of AARR is based upon the criterion that all of the systems and features required to shut the reactor down and maintain it in a safe condition are to remain operable during and after a tornado strike. Maintenance of the functional integrity of the Reactor Containment Building is not necessary to satisfy this criterion.

The entire primary coolant system, except for the reactor vessel, inlet strainer and connecting 24-in. piping, is located within the service floor of the Reactor Containment Building. The reactor control plate drive mechanisms, below the reactor vessel, and all associated electrical circuitry within the Containment Building necessary for reactor shutdown, are also located within this area. This service floor is entirely below grade, and is bounded by the 6-ft-thick (minimum) building foundation mat at the bottom, the 2-1/2-ft-thick cylindrical containment building wall (with 3/8-in.-thick steel liner) on the sides, and the 6-ft-thick experiment floor above. The experiment floor is supported by the outer cylindrical walls, by the reactor and pool shield support structure, and by columns, all of which are supported by the foundation mat.

All electrical penetrations through which vital circuitry passes between equipment in the Reactor Containment Building and the control room or Electrical Equipment Building are below grade, opening into the service floor of the Containment Building on one side and into the cable distribution room in the service floor of the Laboratory and Office Building on the other. The reactor control room is also located on the Laboratory and Office Building service floor, adjacent to the cable distribution room.

This service floor is located entirely below grade, and that portion which houses the control and cable distribution rooms is protected by reinforced concrete walls and ceiling which are specifically designed to meet the requirements of Class-I structures for earthquake resistance.

All equipment associated with the Class-I and Class-II emergency power supplies, including batteries, motor-generator sets, diesel-generators, buses and cabling, are located on the service floor of the Electrical Equipment Building. This area is also entirely below grade, and is

*Ed. note: All questions in this Section were raised by DRL following their review of References (2)-(4). These questions were answered in Reference (8).

protected by reinforced concrete walls and ceiling which are specifically designed to constitute a Class-I structure for earthquake resistance.

It is apparent, then, that a tornado strike could not directly affect the reactor system control, or emergency power supply components located in the protected, below-grade areas of the plant described above, and therefore could not interrupt the ability of the reactor to be shut down, even if the integrity of the above-grade portion of the Reactor Containment Building, and/or the above-grade portions of the Laboratory and Office Building or the Electrical Equipment Building, were destroyed by the tornado. In each case the grade-level structure is obviously sufficiently strong to withstand collapse of the superstructure above without affecting the function of below-grade components.

A tornado could certainly cause great damage to incoming power lines, the outdoor substation, the cooling tower, the secondary pump house, etc., but none of these structures or components are vital to the capability to shut the reactor down and maintain it in a safe condition. This is demonstrated in the analysis presented in Part 1, Section VI-F-2.

It remains, then, to examine the possibilities for damage to those portions of the primary system, and other items which might constitute a source of hazard, which are not located in below-grade protected areas.

The upper part of the reactor vessel, inlet strainer and connecting 24-in. primary piping are located above the experiment floor in the Reactor Containment Building, but are enclosed within the reactor and storage pool complex.

The pool walls are ~7 ft thick surrounding the reactor vessel, 6 ft thick in other areas below the level of the top of the vessel, and 3-1/2 ft thick above that level. The walls of the pool complex extend upward ~17-1/2 ft beyond the top of the vessel and strainer. The inside dimensions of the pool complex, ignoring partitions between pools, are ~18-1/2 x ~59 ft. The pool complex is filled with water to within a few inches of the top during reactor operation.

Spent cores are stored in the pool complex, in a section separated from the Reactor Pool by a 2-ft-thick fixed concrete wall. Exterior walls of the Spent Core Storage Pool are 6 ft thick at the level of the stored cores. The cores are covered by removable sections of heavy stainless steel grating, 24 ft below the water surface, except during fuel transfer operations.

There are two basic mechanisms whereby one might consider damage to vital components within the pool complex to be possible as a result of a tornado; these are damage by (1) missiles generated by collapse of the above-grade portion of the Reactor Containment Building, and (2) missiles originating outside the building which are driven through the Containment Building wall by the tornadic wind. In the first of these cases, the very thick pool walls would certainly survive a collapse of the building shell, and would prevent any very large segments of building roof from entering the pool itself. Smaller fragments, which could enter the pool, would not have

sufficient velocity, and therefore energy, remaining after passing through the minimum 17-1/2 ft of water above the vital components to damage the vessel proper, strainer, 24-in. piping, or the heavy grating above the stored spent cores. The building radial crane, which could in itself constitute a missile, will not normally be used over the pool complex. In addition, the crane trolley will be captive to the bridge, so that it could not, of itself, become a missile.

The small lines on top of the reactor vessel which constitute a part of the rabbit systems could be damaged by missiles falling through the pool water. However, except during actual movement of a rabbit sample (an operation taking but a few seconds), these lines are not part of the primary system boundary. A rupture of the portion of the rabbit assembly just above the reactor vessel cover, containing the actuating slide valve, would constitute a breach of the primary system. The breach would be relatively small compared with ruptures discussed in Part 1, Section VI-D-2b, and moreover would be equivalent to the case of a rupture of the inlet line inside the Reactor Pool, for which no core melting is expected even for a full 24-in. pipe break. (See Part 1, Section VI-D-2b).

Our analysis under question 87, above, indicates that there is no way in which exterior missiles generated by a tornado can breach the Reactor Containment Building. We recognize that this analysis has been questioned (see question 155, below). However, it is certainly inconceivable that a missile could perforate the containment, enter the building, perforate the 3-1/2-ft (minimum) pool walls (or bounce off various objects so that it could enter the pool from the top), pass through the pool water, and still have sufficient energy remaining to cause significant damage to vital components.

Experimenter's equipment will be situated on the experiment floor, generally disposed around the points where the beam tubes penetrate the reactor biological shield wall. In most cases, this equipment will be surrounded by heavy shield blocks which in themselves will constitute protection from missiles. However, it is important to note that the reactor primary system boundary does not extend outward through the beam tube to the experimenter's equipment. The beam tube thimble, within the reactor vessel and projecting into the outer reflector, constitutes the primary system boundary, and this is backed up by a safety diaphragm located within the biological shield. Therefore, damage to experimenter's equipment outside the biological shield cannot affect capability to shut the reactor down and maintain it in a safe condition thereafter.

We conclude, then, that with the current design concept for AARR, all systems and features required to shut the reactor down and maintain it in a safe condition following the shutdown will remain operable during and after a direct tornado strike on the facility.

154. *Indicate the design criteria and loading combinations that will be used in the design of the crane and its supports so they will not become a source of missiles under conditions that may be imposed by earthquakes and tornadoes.*

Answer: The design criteria for the radial crane, its trolleys and supports will require these components to be capable of withstanding the same seismic accelerations as those projected to exist in the Reactor Containment Building structure at the crane elevation, under similar earthquake conditions. These accelerations will be established firmly as part of Title-II design. Suitable hold-down devices will be included if Title-II design analysis demonstrates the need.

155. *Re-examine the possibility of missile perforation in the light of data on missile perforation in reinforced concrete. In the event missile perforation can not be ruled out, discuss the vulnerability of other Class I components inside of the containment and the capability of maintaining the reactor in a safe and shutdown condition. (Part 1, Section IV-G-2b).*

Answer: The vulnerability of Class-I components other than the above-grade portion of the Reactor Containment Building to damage by external tornado-generated missiles is discussed in the answer to question 153 above. It is shown that after perforating the containment wall, a missile would also have to (1) perforate the 6-ft thick experiment floor, (2) perforate the wall of the pool complex, which ranges in thickness from 3-1/2 to 7-ft, or (3) make several changes of direction by rebounding and then pass through at least 17-ft of water, before damage could be done to Class-I components inside the containment. A further perforation of either the reactor vessel itself, within the pool structure, or of the pool and shield foundation below the experiment floor (to gain access to the control plate drive mechanisms), would be necessary before the damage inflicted could affect capability to shut the reactor down. It is not considered credible that any tornado-generated missile could possess sufficient energy to achieve such a sequence of events. Therefore, since neither the containment perforation in itself nor credible subsequent damage can alter capability to shut the reactor down and maintain a safe condition thereafter, the consideration of whether or not tornado-generated external missiles can perforate the containment is of secondary importance.

156. *Indicate the loading combinations that will be considered for the piping systems, the manner in which the applicable loads will be applied and the stress and/or deformation limits which will govern.*

Answer: In the design of the primary system piping, the March 1967 draft of the "American Standard Code for Pressure Piping," "Nuclear Power Piping," USA B31.7, will be satisfied. Primary piping design fabrication and testing will be in accordance with Class I requirements of the Code.

Evaluation of stresses, reactions and movements associated with pressure, temperature and external forces will all be carried out as outlined in this Code. The allowable stress intensity values will be in accordance

with Code Table A.1. Classification of stress intensities will be in accordance with Table 1.702.5.4, and stresses will be combined in accordance with Table 1.702.5.5. Deformation limits will be set in accordance with Paragraph 1.719 of the Code.

157. *Indicate the structural loads and load combinations selected as the design bases including dead load, pressure load, thermal loads (due to concrete and liner structure), wind load, tornado (wind loading and differential pressure) load, design and maximum earthquake loads (horizontal and vertical excitation), live load, buoyant water load, snowload, and vacuum load. Discuss the bases for the individual load selections, the manner in which each loading will be treated in combination with other applicable loadings, and the effect that each individual loading will have in generating moment, shear, deflection, longitudinal force and hoop tension throughout the reference structure. Also, indicate the stress (and strain, as applicable) criteria which will be met for all loading combinations.*

Answer: Table 157-1 gives a listing of the individual design loads and conditions being considered in the design of the AARR containment structure. Table 157-2 shows the manner in which these loads are combined to represent the various conditions to which the structure is postulated to be subjected. The information in these two tables constitutes the design loading criteria for the structure.

The exact manner in which the individual loads will be treated in the various postulated combinations is the subject of detailed Title-II design study, and cannot be discussed at this time. Moreover, it is neither customary nor necessary, in designing such a structure, to evaluate the separate contributions of individual loadings in generating moment, shear, etc. These effects are computed only for the various load combinations, and the design is adjusted as needed to meet the overall criteria.

The stress criterion which will be met for all loading combinations (cases), except Case III_A, is that stresses will be kept within the elastic range, i.e., within the normal working stresses for the materials involved, as specified in the applicable codes.

For Case III_A, the stress criterion will be such as to permit stresses to exceed yield, but strains will be limited to assure that the integrity of the containment is not violated.

158. *For seismic design criteria, indicate:*

- (a) *The damping factors that will be employed for design and maximum earthquakes for both the containment and other Class I systems.*
- (b) *The design spectra for both the design and maximum earthquake cases.*

TABLE 157-1

INDIVIDUAL LOADS CONSIDERED
IN AARR CONTAINMENT STRUCTURAL DESIGN

<u>Symbol</u>	<u>Explanation</u>
D. L. PERM.	<u>Dead Loads-Permanent.</u> Includes the weight of the structure and of the permanently installed mechanical equipment.
D. L. VAR.	<u>Dead Loads-Variable.</u> Includes the weight of the water in the pools and canals, and experiment floor long term loads such as those resulting from experimenters equipment and shielding. The load associated with each beam port is assumed to be 300 kips acting at a center of gravity 16 ft from the outside face of the reactor biological shield, on the beam tube centerline.
P. L. (MCA)	<u>Pressure Load.</u> This is the load resulting from incidence of the maximum accident pressure calculated result from a MCA, with an added factor of safety of 1.1. The pressure load corresponds to an internal pressure of 6.0 psig.*
T. L. (MCA)	<u>Temperature Load.</u> The maximum liner temperature varies throughout the structure from a maximum of 123°F on the dome liner to a maximum of 108.7°F on the service floor liner. These liner temperatures also vary with the pressure associated with a MCA. The combination of temperature and pressure that will produce the maximum stresses or strains will be selected for the design of the individual components of the containment.
W. L.	<u>Wind Load.</u> The load generated by a 90 mph wind at 30 ft elevation above grade. Velocity is considered to increase with increased height in accordance with the "1/7 Power Rule." Shape and drag factors, and application of the resulting wind pressure will be in accordance with "Wind Forces on Structures," ASCE Paper No. 3269.
E. L.	<u>Earthquake Load.</u> The load produced by a 0.1g horizontal acceleration and a 0.066g vertical acceleration, applied simultaneously. (These values are tentative and will be confirmed by the ANL seismic consultant.)

*The maximum accident pressure of 5.3 psig given in PART I, SECTION VII-B-2d has been increased to 5.45 psig because of a recent reduction in the Containment Building internal volume.

TABLE 157-1 (continued)

<u>Symbol</u>	<u>Explanation</u>												
S. L.	<u>Snow Load.</u> A load of 30 lb/ft ²												
T. P. L. (TEST)	<u>Test Pressure Load.</u> The load produced by the maximum test pressure of 6.8 psig. which is in accord with ASME Code Sections III and VIII.												
B. L.	<u>Barometric Load.</u> The load resulting from a maximum internal negative pressure of 0.5 psig, due to barometric changes.												
Crane and Balcony L. L.	<u>Crane and Balcony Live Load.</u> The load resulting from the maximum forces which could be produced in any direction by any combination of crane dead load and/or lifted load, accelerated in any direction within the capacity, speed, and braking limits of the radial crane. These are: <table> <tr> <td>Main hook capacity</td><td>25 ton</td></tr> <tr> <td>Auxiliary hook capacity</td><td>5 ton</td></tr> <tr> <td>Bridge speed</td><td>150 fpm at rim</td></tr> <tr> <td>Trolley speed</td><td>100 fpm</td></tr> <tr> <td>Main hoist speed</td><td>10 fpm</td></tr> <tr> <td>Auxiliary hoist speed</td><td>20 fpm</td></tr> </table>	Main hook capacity	25 ton	Auxiliary hook capacity	5 ton	Bridge speed	150 fpm at rim	Trolley speed	100 fpm	Main hoist speed	10 fpm	Auxiliary hoist speed	20 fpm
Main hook capacity	25 ton												
Auxiliary hook capacity	5 ton												
Bridge speed	150 fpm at rim												
Trolley speed	100 fpm												
Main hoist speed	10 fpm												
Auxiliary hoist speed	20 fpm												

The instrument balcony live load is 150 lb/ft².

TABLE 157-2

LOAD COMBINATIONS CONSIDERED
IN AARR CONTAINMENT STRUCTURE DESIGN (a)

	D.L. PERM.	D.L. VAR.	P.L. (MCA)	T.L. (MCA)	W.L.	E.L.	S.L.	T.P.L. (TEST)	B.L.	CRANE AND BALCONY L.L.
Case I	X	X	X						X	
Case I _A	X	X	X	X	X				X	X
Case II	X	X			X					X
Case II _A	X				X					
Case III	X	X	X	X		X			X	X
Case III _A	X	X	X	X		2EL			X	X
Case IV	X	X					X		X	X
Case V	X							X		

(a) See Table 157-1 for nomenclature. These loading combinations supersede those given in response to question 90, above.

- (c) *The manner in which the horizontal and vertical earthquake stress will be combined together and with the other applicable loadings.*
- (d) *The general analytical model including mass distribution, stiffness coefficients, modes of vibrations, and analytical procedures for arriving at a loading distribution on the structure.*
- (e) *Indicate the magnitude of lateral earth pressure under seismic loading and indicate how such loading will be factored into the containment design.*

Answer: At the present time, ANL's seismic consultant is developing seismic design criteria for the entire AARR facility. Since this work has not yet been completed, it is not possible to give specific answers to these questions. However, a general discussion can be presented describing the particular approach being employed.

The basic approach of the seismic consultant is related to seismic risk and probability of survival, and is predicated on the reasoning that a structural system that is vulnerable to seismic effects has a certain probability of survival which depends on its resistance, the level of seismic activity to which it is exposed, and the duration or term of exposure. The resistance of the system can be defined by a design shock spectrum which relates its allowable response (deflection, pseudo-velocity or acceleration) to period and internal damping. The procedure under development for AARR treats this design spectrum as the critical shock spectrum. This critical shock spectrum can be produced by earthquakes of different magnitudes at appropriate distances from the reactor site.

In a region with a relatively low level of activity, the existing information for that area may be insufficient to predict possible earthquakes and their effects. However, in such a case (as it is with the AARR site), it may be possible to make a prediction by comparisons with regions of greater seismic activity. For example, if it is assumed that the relative numbers of small and large shocks are independent of region, one can take the distribution of shocks for the entire world as a base and multiply it by an appropriate factor so that there is agreement with the observed numbers of small shocks in the region under investigation.

Briefly, the containment building will be treated as a relatively stiff cylinder supported by a flexible mat resting on an elastic medium. The lowest and most important frequencies of vibration due to a horizontal disturbance are believed to correspond mainly to deformation of the base and the supporting soil and only slightly to distortion of the cylinder. For a rigid cylinder elastically supported at the base, there are two modes of horizontal motion and one mode of vertical motion. The response to an earthquake is found by use of the spectrum coordinates corresponding to the periods and damping of the modes of vibration of the structure. Work is presently underway to determine all pertinent factors, i.e., damping coefficients, ground accelerations, elastic properties of soil and their effects of earthquake response.

As an example of the manners in which seismic stresses will be combined with other loads, consider the building base (or mat) slab. This structure will be designed to resist simultaneously all dead and live loads (vertically), and soil reactions combined with vertical load increase (or decrease, whichever is worst loading) due to vertical acceleration, plus the positive and negative soil pressures due to the dynamic couple acting because of rocking of the building.

159. *For the design pressure-temperature conditions, provide:*

(a) The general geometry to be utilized to determine structural stresses.

Answer: For design purposes, the containment structure will be separated into various different shell elements of "simple geometry," these being the ellipsoidal dome, the cylindrical wall, and the flat plate or foundation mat.

(b) Considerations given to discontinuity stresses. What assumptions with regard to structural stiffness form the basis for determination of these stresses?

Answer: Consideration will be given to discontinuity stresses developed at abrupt changes in section, such as at the juncture of the cylindrical wall and the foundation mat and at the juncture of the cylindrical wall and the ellipsoidal dome. It is assumed that the cylindrical wall and the ellipsoidal dome are sufficiently thin that the use of "thin shell" theory is justified. Consideration will also be given to discontinuity stresses at all large openings in the cylindrical shell such as the containment building entrance hatches, air locks, and large penetrations. The exact means for considering discontinuity stresses is being developed as part of the Title-II structural design effort.

(c) A justification for the use of lapped splicing for reinforcing steel.

Answer: The American Concrete Institute Building Code ACI 318-63 states in Section #805 that "lapped splices in tension shall not be used for bar sizes larger than #11." Bars carrying tensile loads that are larger than #11 will be either welded or joined together utilizing a mechanical butt-splicing method with filler metal and an enclosing steel sleeve, i.e., the "Cadweld" process. Bars carrying tensile loads that are #11 or smaller in size will be spliced in accordance with Section #805. These splices will be staggered in patterns that least impair the strength of the structure.

160. *Indicate the design methods to be used in the structural design for radial shear, lateral shear, longitudinal shear, and uplift shear (due to accident pressure). Discuss the extent*

to which liner participation will be relied upon to cope with these shear loadings and indicate the general reinforcing patterns that will be used to reinforce against such shears.

Answer: Generally, for the ellipsoidal dome and cylindrical shell, an analysis employing deformation compatibility will be used to determine the various shear forces existing for the various loading conditions. Conventional methods will be used during Title-II to combine the various shear stresses resulting. Conventional analysis will also be used for the Title-II design of the foundation mat.

The mat and wall liner will not be relied upon to resist any of the shear loadings, i.e., radial shear, lateral shear, longitudinal shear, and up-lift shear. However, the liner of the dome may be used to resist shear loads. This will be determined during the Title-II design.

Patterns of reinforcement to resist the above shears will be developed in the Title-II design.

161. *With respect to liner design, provide:*

(a) The stress limits and types of loadings that will be considered in control of liner buckling and strain concentration at discontinuities.

Answer: The major sources of loadings tending to cause buckling in the liner are shrinkage of the concrete building shell and increases in liner temperature. The liner anchorage spacing will be determined such that these compressive loads do not exceed a conservative estimate of the critical (elastic) buckling stress of the liner, with appropriate margin of safety. The effects of negative building pressure on this problem will be considered.

(b) The stress and strain limits that will be used for the liner, the bases for these limits, and the extent to which these limits are related to liner design leakage.

Answer: Stress and strain limits for the liner will be in accordance with Section III of the ASME Boiler and Pressure Vessel Code. As pointed out in question 91 above, the liner strains will remain in the elastic region, and it is felt that this provides a high degree of assurance of liner leaktightness integrity. Allowable stress limits for the liner for various loading conditions are discussed in response to question 157 above.

(c) The type, character, and magnitude of cyclic loads for which the containment liner will be designed.

Answer: Cyclic loads on the liner are of two types: gross and local. Gross, or overall, cyclic straining of the liner will be caused by fluctuations of atmospheric pressure and temperature. Local cyclic effects are those mainly at penetrations caused by heating and cooling of pipe lines. The magnitude of these cyclic loads is not known at this time.

The actual design of the liner will take these loads into account, especially as it might affect differential strains in adjoining liner panels. Evaluations of fatigue damage will follow that included in the ASME Code, Section III.

(d) The analytical procedures and techniques to be used in liner anchorage design including calculational procedures.

Answer: In general, the procedures to be used in liner anchorage design will be to determine conservative limits on plate buckling stresses, effects of differential strains in adjacent panels, and structural compatibility of liner thickness and anchorage stud flexibility. The effects of an anchorage failure on resulting load patterns and ability of adjacent anchors to redistribute loads will be analyzed.

(e) For anchorages, the cyclic stress failure mode and failure propagation characteristics. Discuss extent to which these factors influence leaktightness integrity. What design provisions will be incorporated to prevent such failures from jeopardizing leaktight integrity?

(f) Tolerances on liner plate out-of-roundness and liner plate fit-up and their bases.

(g) The analytical procedures for analysis of liner stresses around penetrations. Also provide the method of liner design to accommodate these stresses.

Answer: A proper reply to these three questions requires detailed design information being developed as part of the Title-II design effort. Indeed, some of the data required cannot be determined until the liner construction contract has been awarded and the contractor has begun his detailed shop design work. Therefore, no discussion can be presented at this time.

(h) The design approach that will be used where loadings must be transferred through the liner such as at crane brackets or machinery equipment mounts.

Answer: The general design approach to the various localized attachments to the liner, such as crane rail support brackets, is to design the attachment such that its failure would not impair the leaktightness integrity of the liner. One way to accomplish this would be to have each bracket designed to transmit its load through the liner directly into the concrete by using appropriate anchors at the location of the bracket. Strains in the liner at the bracket would be limited to values previously described, up to failure of the bracket itself.

162. For penetrations, provide:

(a) Design assurance that piping failures do not lead to violations of containment integrity.

Answer: In general, pipe loads will not be transmitted to the containment liner, but to anchor points located away from each side of the liner. The piping lines will communicate with the liner through thin membranes for sealing purposes.

(b) The upper limits on magnitude and frequency of vibratory loadings on penetrations and surrounding liner regions. Indicate how such loadings will be treated in the design of the liner.

Answer: The magnitude and frequency of vibratory loads will be determined in Title II. In general, pipe lines or other penetrations which might incur vibratory loads will be isolated from the liner with thin membrane seals as in (a) above.

(c) The general methods for reinforcing the concrete structure around penetrations (include typical examples).

Answer: In general, reinforcement around large penetrations will be based on consideration of total force blocks which are interrupted by the opening. Consideration will be given to the reinforcing action of the penetration liner, particularly as it affects strain compatibility of the opening and the shell membrane. Reinforcing bars will be distributed around the opening in a manner to accommodate concentrations of tensile conditions indicated by detailed analysis. The advisability of utilizing the liner and/or the penetration as reinforcement will also be studied. (For a low pressure system such as the AARR containment, the contribution of liner and penetration steel is very significant.)

163. *With regard to design of large openings in the concrete structure, provide:*

(a) The number and size of these openings.

Answer: The following table lists large openings in the containment building.

<u>Description</u>	<u>Number</u>	<u>Size</u>
Access Hatch	1	12' x 12'
Access Hatch	1	10' x 10'
Air Lock No. 1	1	8' Dia.
Air Lock No. 2	1	9' Dia.
Air Lock No. 3	1	3' Dia.
Canal Pressure Hatch	1	6' Dia.
Beam Tube Sleeves	6	3' Dia.
Secondary Pipe Lines	2	5' Dia.
Ventilation Ducts	2	4' Dia.

(b) The primary, secondary, and thermal loads that will be considered for these openings, including design combinations.

Answer: Loadings and their combinations will be those given in the tables presented in response to question 157 above.

(c) The analytical procedures that will be used for openings, including those procedures used in checking for stiffening effects.

Answer: Specific analytical procedures for large openings have not yet been determined. In general, it is planned to use information presented at the recent Prestressed Concrete Pressure Vessel Conference held in London, England (March 1967). The value of this information is that quite a bit of experimental work has been carried out to correlate various methods of analysis. It is recognized that the AARR containment building is not prestressed, but certain similarities do exist, e.g., the effects of large openings. It can be stated that the analytical methods will consider: (1) reinforcement, either as rebar or penetration liner steel; (2) strains imposed on the liner; (3) stress conditions in the concrete; and (4) effect on overall behavior of vessel.

(d) Criteria governing need for additional reinforcement around the openings to resist local effects.

Answer: Only general criteria can be given at this time. As a design objective, efforts will be made to design large openings such that strains in the vessel liner at the opening are compatible with penetration liner strains. Reinforcement needs will be determined on this basis and also to replace the amount of steel rebars which have been interrupted by the opening. Consideration will be given to the reinforcement and stiffening effect of the penetration liner in determining additional reinforcement in the form of bonded steel bars.

(e) The conservatism used in the design of the openings and adjacent transition regions against failure in shear.

Answer: The precise degree of conservatism against shear failure is not known at this time. However, the overriding criterion will always be that a high degree of conservatism will be applied to the shear design in order to assure that the tensile capabilities of the vessel structure can be fully developed. An advantage of pressure vessel design is that the value of total shear is known with a high degree of accuracy, and except for local effects such as earthquake loads, it is known that their distribution is uniform.

164. *With respect to concrete construction, provide:*

(a) The specifications, procedures and tests to be used in the design of the concrete mixes and in developing and verifying the design properties of the concrete.

(b) The concrete mixing, placing and curing procedures to be used.

Answer: The information requested in these two questions will be developed as part of the Title-II design effort.

(c) The general procedures for bonding between lifts.

Answer: Bonding between the various concrete pours or lifts will be accomplished in the conventional manner, i.e., by scarification and/or the inclusion of construction keys as required.

(d) The general manner in which concrete lifts will be placed or staggered.

Answer: It is not known at this time what the actual pouring sequence will be, but it is reasonable to assume that an alternate block scheme of pours will be employed for each lift.

165. *With respect to liner construction, provide:*

(a) The amount of seam weld radiography and other quality control to be provided. Also, provide an evaluation of the extent to which the selected liner radiography provides a high degree of assurance that flaws that may develop into positive leakage paths under design basis accident conditions do not, in fact, exist.

Answer: The amount of radiography and other quality control for the liner erection have not, at this time, been determined. However, it should be noted that radiography will be used for monitoring structural integrity and welder performance, not to provide assurance of leaktightness. Radiography is incapable of providing a "high degree of assurance" that leaks do not exist; in fact, it provides almost no assurance of leaktightness even when carried out to the extent of 100% of the welds. Current tentative plans for AARR call for radiography of perhaps 10% of the welds.

(b) A discussion of construction controls that will be employed to assure that "out-of-roundness" of liner plates does not exceed the design tolerances.

Answer: Complete construction controls have, at this time, not been determined. However, dimensional measurements will be instituted as part of the erection surveillance to assure that design tolerances are complied with as erection proceeds.

(c) The applicable ASME Code sections that will be adhered to in the construction.

Answer: Applicable parts of the ASME Boiler and Pressure Vessel Code, Sections III, VIII, and IX will be adhered to for the construction of the liner.

(d) The criteria to be used to govern plate and weld ductility.

Answer: Ductility requirements of the liner plate will be determined during the Title-II design, and a plate material having the properties required by the design will be selected. The ASME specifications of the plate material selected will govern the plate ductility.

(e) The construction inspection procedures that will be used for liner plate material properties, seam weld leakage, liner anchorage weldments and penetration welds.

Answer: These procedures will be developed during the Title-II design.

(f) The general sequence of construction of the liner with respect to concrete construction, with emphasis given to the construction procedures for placement of the liner in regions that will become inaccessible for testing and repair and to the zoning of and testing of weld channels to be accomplished prior to placement of backing concrete.

Answer: After the installation of the foundation mat, the liner covering the mat, i.e., the Containment Building service floor, will be installed. This will be welded to suitable anchorages that will act as leveling devices and backing strips. The Title-II design will determine the requirements, details, and application of leak-testing devices for these weld seams. The cylindrical wall liner is next installed to approximately 5 ft above grade except in the area of the transfer canal, where it will be installed to the full height of the canal. The concrete work is next completed to grade, including the experiment floor slab and the cylindrical walls ("backing concrete") outside the wall liner. The concrete walls of the pool complex and reactor biological shield are then completed to their full height. The remaining portions of the cylindrical liner and the ellipsoidal dome liner are next erected and tested. After completion of the liner work, the concrete cylindrical walls and the dome are poured. The Title-II design will determine the details required for the leak testing of areas becoming inaccessible upon completion of the concrete work. At this time, the plans do not include the use of weld channels.

(g) The manner by which liner leaks will be traced and repaired should excessive leakage be indicated during initial or subsequent leak rate tests.

Answer: Vacuum-box testing will determine areas of leakage in the welded seams. The weld areas found to be defective will be repaired by conventional means, and retested.

(h) Procedures that will be used to control seam weld porosity.

Answer: Radiography, used in accordance with the ASME Boiler and Pressure Vessel Code, will be employed to monitor seam-weld porosity. Multiple-pass welding will be used to minimize through porosity.

166. *With regard to on-site concrete construction quality control, provide the following:*

(a) The plans for quality control.

Answer: An on-site inspection staff will maintain 100% surveillance of all concrete batches and pours. Standard field procedures such as slump tests will be used. The inspector will also monitor the sampling and cylinder production for subsequent testing. The detailed plans are part of Title-III design and will be developed later.

(b) The codes of practice that will be followed with regard to the construction quality control. Indicate where and to what extent the proposed ACI-301 standard practices for construction will be equaled, exceeded, or not followed.

Answer: The exact codes to be followed for construction quality control will be set forth in the detailed construction specifications which are developed as part of the Title-II design effort. It is expected that applicable sections of ACI standards, such as Sections 214, 301, 306, 311, etc., will be used.

(c) The controls and limitations with regard to deleterious substances in the concrete mix.

Answer: The limitations on deleterious substances in the concrete mix will be specified during Title-II and are related to the required concrete properties. The controls used to assure conformance to these limits will be developed as part of Title-III. It is expected that the ASTM specifications listed in ACI-613, including the Appendix, will be applied.

(d) A list and description of the user inspection and testing that will be done for all materials utilized in the containment construction. Include the basis for and the results to be achieved by such testing and if testing of a particular construction material is not deemed necessary, provide justification for its omission.

Answer: All inspection and testing for this project are the responsibility of the user, Argonne National Laboratory. The actual tests to be conducted, the basis for the tests, and the results to be achieved cannot be determined until the design is completed so that the requirements of the design are known. Basically, all materials critical to the containment function will be inspected and/or tested as is appropriate.

167. *Describe the site inspection procedures that will be used including the following:*

(a) Minimum qualifications and experience of inspectors.

Answer: The inspection staff will report to the Argonne National Laboratory Project Manager. It will be composed of personnel from the

Architect-Engineer, the ANL Inspection Department, the ANL Plant Engineering Department, and the ANL Reactor Engineering Division design groups. There are no minimum qualifications or experience requirements specified, but rather people deemed capable will be used as required.

(b) The degree to which materials preparation and construction activities will be subject to inspector surveillance and the authority of the inspectors.

Answer: All material preparation and construction activities will be subject to 100% inspector surveillance. While 100% surveillance will be conducted on all field work, the same will not be true for vendors' shop work. Shop work will be inspected on the basis of estimated criticality and schedule. For example, as indicated in the response to Question 172.below, a full-time resident inspector will be assigned to follow fabrication of the reactor pressure vessel, which is a critically engineered component whose complete compliance to specifications is vital to plant safety and performance and whose timely delivery is highly important to the overall construction schedule.

For all Argonne-procured materials or for material procured by Argonne's contractors, the inspectors will have direct legal authority. For all materials procurement activities and contracts administered by AEC-CH, the ANL inspectors have been delegated inspection authority by the Manager of AEC-CH.

(c) The extent to which engineering personnel associated with the design will participate in on-site inspections.

Answer: The engineering personnel associated with the design will participate in on-site inspections of all critical operations, and on a random basis for non-critical areas.

(d) The extent to which the inspection organization will be independent of the construction organization.

Answer: With one exception, the inspection organization is completely independent of the construction organization. The inspection organization is made up as indicated in (a) above, and the construction will be by various contractors working on a lump-sum bid basis. The exception is in the area of the reactor internals and experimental facilities. These will be installed by a CPFF contractor under the direct control of the ANL project organization. While inspection and construction supervision will be done by different people, the responsibility for both will lie in the same organization.

(e) The records of inspection that will be kept.

Answer: Each inspector will file a daily report in addition to maintaining a daily log.

168.

With regard to containment pneumatic testing, provide:

(a) The extent to which the selected pneumatic test pressure will simulate design basis accident conditions by comparing the stresses under pneumatic test pressure with those in the structure under accident pressure plus earthquake (and other combinations, if governing) for the following structural elements: (1) circumferential reinforcing; (2) axial (longitudinal) reinforcing; (3) dome reinforcing; (4) base slab reinforcing; and (5) large openings.

In the event significant differences exist between stresses and strains the structural elements experience under test loading and those calculated to exist under design basis accident loading, provide a discussion in support of the selected test pressure. Include in this discussion the extent to which an increased test pressure or design modifications or both have been considered in an effort to obtain closer test verification of structural integrity.

Answer: The completed concrete containment structure will be pneumatically pressure tested to 6.8 psig, which is 1.25 times the design pressure. Specific evaluation of the extent to which this test pressure simulates accident conditions cannot be made until Title-II design is well along to completion. However, certain general statements can be made which are pertinent to this question. First, the test pressure has been selected as greater than the MCA pressure and in this regard, we are guided by test pressures required of all-steel containment vessels. It is very important to note that the maximum accident pressure is reached well in advance of significant liner temperature increases. The significance of this is that as far as internal pressure is concerned, the test will provide excellent verification of the structural integrity. What is more important is the time of year that the tests are made. Since the liner temperature rises only a few degrees at maximum pressure (to 81°F) the outside ambient temperature will be the governing factor as far as effects of ΔT across the structure are concerned. It may be that a re-evaluation of test conditions or details will be necessary once the time of testing is certain and the final design of the structure is known.

With regard to general comparison of test conditions to those existing under accident plus earthquake, it is expected that significant discrepancies will exist. For example, the response of the base slab to pressure and earthquake cannot be simulated by pressure testing. (Nor can the loading on the soil be simulated.) Peak stresses could be reached, but the imbalanced loading of the dynamic couple on the base support cannot be accounted for by pressure test alone, regardless of its level. The very nature of the dynamic effects of earthquake loadings which seek out and test all components in accordance with mass distribution and structural response leads us to conclude the following: no amount of increased uniform pressure loading or design modification can properly verify the expected response of the structure to the effects of accident pressure plus earthquake loads.

(b) The sequence of procedures for containment pneumatic testing. Provide the criteria for structural acceptance and the general stress and strain ranges.

Answer: The general plan for pressure testing is as follows:

- (1) The liner will be completed after casting the structural concrete wall to about grade level. At this time a low pressure (value not yet determined) pneumatic test will be made to check overall leaktightness of the liner membrane.
- (2) Upon completion of the concrete structure, the entire RCB will be pressure-tested (test pressure is tentatively set at 6.8 psig). Strains and deflections will be measured at critical in the structure.
- (3) An integrated leakage rate test will be conducted at design pressure after all penetrations have been installed.
- (4) It is tentatively planned to perform an integrated leakage rate test at a level less than design pressure to establish a basis for future leakage rate tests to be conducted.

(c) The instrumentation program to be employed to verify the design. Identify the structural and liner areas that will be instrumented, and the purpose, type, expected accuracy and redundancy of the instruments to be used. Discuss the protective measures to be taken to ensure performance over the required interval between placement and use.

Answer: The details of test instrumentation will be developed during Title-II design, and as part of construction.

169. *Discuss the provisions that will be followed to minimize reinforcing bar corrosion, and the measures taken for cathodic protection.*

Answer: The amount of corrosion conceivable in the containment structure reinforcing steel is practically nil and does not warrant special cathodic protection. This conclusion is based on observations of the satisfactory performance of below-grade reinforced concrete structures in this area, considerations of the ground water chemistry, and the absence of conditions conducive to electrolysis. It is also based further on the fact that the underground drain system will maintain dry conditions around the Containment Building.

170. *Describe the surveillance capabilities provided by the containment design which would facilitate periodic inspection of the steel liner, and periodic pressure-testing of the containment system. If such tests are to be performed at reduced pressure, provide an evaluation of the minimum level that would be required to verify continued structural integrity considering structural response and installed surveillance instrumentation requirements.*

Answer: Substantially all of the inner surface of the steel liner is accessible for visual examination. We are not presently aware of any causes for hidden corrosion to occur. With regard to periodic leakage tests, each penetration will have the capability for individual leakage testing. The overall leakage rate and continued structural integrity will be periodically verified to assure compliance with acceptance criteria to be established. It is not yet established at what level these tests will be performed. For leakage rate tests, results of the initial acceptance test and tests at reduced pressures will be used for comparison purposes if subsequent tests are run at reduced pressures. The structural integrity will be verified, possibly at reduced pressures, in which case an evaluation of a minimum value will be made considering final design, results of initial structural test and instrumentation capability. It is expected that a reasonably uniform load-deformation response will exist, and this will be factored into this evaluation.

171. *Describe the preliminary foundation design associated with the Reactor Containment Building and other structures containing Class I systems or components. Discuss the basis for, and the magnitudes of the loadings that will be used in the design of the foundation.*

Answer: The preliminary foundation design is based upon the following founding concept:

- (1) The Containment Building and the pools and canals exterior to the containment structure (in the Active Material Handling Building) are carried on a common foundation mat with its bottom located approximately 36 ft below grade.
- (2) Structures containing Class-I systems and components and which are located adjacent to the Containment Building are founded on either isolated spread footings or caissons.
- (3) Differential settlement between the differently loaded and founded structures is recognized and is being limited and allowed for in the Title-II design.

The dead weight of the structure has been divided into two parts:

- (1) Dead Loads-Permanent (weight of structure and permanently installed mechanical equipment), currently estimated at 47,900 kips.
- (2) Dead Loads-Variable (weight of water in pools and canals, and experimenter equipment and shielding loads), currently estimated at 5,200 kips. The total estimated dead load is therefore about 53,100 kips.

These loadings, plus the effect of their eccentricity to the center of gravity of the foundation, will be appropriately combined with the loadings generated in the Title-II design (for the effect of wind, earthquake, and pressure load) for the various cases required to achieve the final foundation design.

172. *Will ANL have a resident inspector in the (pressure vessel) manufacturer's shop who will be responsible for seeing that the inspections and tests required by the ASME Code plus supplementary criteria are carried out with satisfactory results?*

Answer: Yes. In addition, all important tests will be witnessed by appropriate Project staff personnel.

173. *How will the Inconel plates (for the pressure vessel) be inspected? What acceptance standards will be used?*

Answer: All Inconel plates will be ultrasonically examined in accordance with the requirements of ASME Code Case 1338-2, using longitudinal wave 1-1/8-in. diameter 2-1/4 Mc crystals. Deviations from this procedure may be requested by the mill, i.e., use of 1 Mc crystals. Acceptance standards will be as follows: (1) any defect which shows a total loss of back reflection that cannot be contained within a circle whose diameter is the greater of 3 in. or 1/2 of the plate thickness shall be unacceptable; (2) two or more defects smaller than described in (1), which cause a complete loss of back reflection, shall be unacceptable unless separated by a minimum distance equal to the greatest dimension of the larger defect, unless the defects are contained within the area described in (1).

174. *In the event that the (pressure vessel) plates were to have some unacceptable defects, how will the mill go about repairing these defects, and how will the repair work be validated?*

Answer: Defects in plates will be repaired in accordance with Paragraph N-321.2 in Section III, ASME Code, which also delineates repair validation procedures.

175. *How will the test coupons be treated to reflect the properties of the material of the as-fabricated vessel?*

Answer: For the top and bottom head forgings, one or more test coupons will be quenched and tempered with the heads in accordance with the requirements of ASME Code Case 1358, Paragraph 7, and will then be subjected to the same scheduled thermal treatments to which the vessel or vessel component will be later subjected. Samples of the Inconel plate consisting of edge strips from the shell plate and/or large nozzle cutouts will be tacked to the vessel and will receive the same thermal treatments as the vessel shell. (It is also planned to have representative plate and nozzle welded samples accompany the vessel during fabrication.) Tests will then be made after fabrication to establish as-built vessel properties.

176. *What dimensional tolerances have been specified for the plate thickness, the vessel diameter, and maximum offset for the longitudinal and circumferential weld joints?*

Answer: A nominal 3/16 in. on plate thickness is added to all shell sections to permit machining, if required, to maintain assurance of the specified minimum plate thicknesses and to maintain weld joint alignment to $\pm 1/16$ in.

177. *What fatigue data, pertaining to Inconel-600, are available?*

Answer: Fatigue data for some tests are summarized and references are given in Technical Bulletin T-7, Engineering Properties of Inconel Alloy 600, published by the International Nickel Company. In addition, results of work performed by the Naval Research Laboratory are given in their publication NRS-1.

178. *Pertaining to manufacturing practices:*

(a) How will the vessel manufacturer validate the proposed welding procedures for welding 3 in. Inconel plates? Will the same welding procedures be used for longitudinal as well as circumferential welds?

Answer: The welding procedures will be validated in accordance with ASME Code Case 1336, Paragraph 2, which references Section IX of the ASME Code and includes additional requirements. We do not know at this time whether the same procedures will be used for longitudinal and circumferential welds.

(b) How will the welds be inspected during welding and upon completion?

Answer: Nondestructive testing of welds and weld preparations will be performed in accordance with Section III of the ASME Code, and in accordance with a detailed welding procedure not yet prepared. In general, the following tests will be performed:

- (1) Magnetic particle or liquid penetrant testing of base material shall be performed at the following stages of fabrication:
 - (a) weld preparations prior to fit-up,
 - (b) surfaces to be overlaid with weld-deposited cladding,
 - (c) all surfaces of low alloy base material after quench and temper heat treatment,
 - (d) cut edges of plates, forgings, and fittings,
 - (e) during welding - for root layer, each layer of weld, after back chipping, each 1/2 in. of weld, after heat treatment and final surface.
- (2) Upon completion, all welds will be radiographed.
- (c) Will the nozzles be welded to the vessel wall using full penetration welds?*

Answer: Yes.

179. *Stress corrosion failures of 90 mm. diameter tubes of Inconel-600 welded to ferritic steel has been reported of the Ågesta Reactor Vessel. Is sufficient information available on the corrosion resistance of Inconel-600, in water, to assure the integrity of the vessel and its internals during the service life of the reactor plant?*

Answer: It is noted that the Ågesta Swedish reactor is a pressurized heavy water power reactor operating at primary coolant temperatures between 400 and 470°F. Although we normally would not consider Inconel to be subject to stress-corrosion cracking failure at low and moderate temperatures, the alloy may be subject to this problem at very high temperatures. Thus, the higher temperatures in the Ågesta vessel would be far more likely to cause stress-corrosion cracking than would the AARR operating temperatures.* In the AARR design, deliberate efforts have been made to avoid such a condition by completely overlaying the low alloy head forgings with Inconel and making strength welds between tubular penetrations and the head from Inconel to Inconel. Thus, at no point is the combination of low alloy head material and Inconel sleeves or other parts subjected to a common water environment.

180. *Will the pressure vessel be subjected to periodic hydrostatic tests during its service life, and if planned, at what pressure level?*

Answer: The reactor pressure vessel will be hydrostatically pressurized after each new core loading, i.e., approximately once every three to four weeks. The pressure level of this test is not yet determined, but will be in excess of the operating pressure.

181. *Reactor vessel standards presently under consideration would require 100% volumetric inspectability of the reactor vessel, either from the inside or outside. What inspection capability, especially in critical areas such as nozzle penetrations, flanges, and other structural discontinuities, will ANL specify as a design requirement for the reactor vessel? What capability will ANL have for this type of inspection of the reactor vessel during the vessel's service life?*

Answer: Except for the 47-in. i.d. lower extension shell (which passes through the pool floor), the entire outer surface of the AARR reactor is exposed to either pool water or building air atmosphere (for the bottom head). Thus, the capability for visual examination of virtually the entire outer surface of the vessel exists. The 47-in. i.d. lower section is inspectable internally by removing the lower head.

* Argonne personnel have had direct discussions of the reported failures with Ågesta personnel, and it appears that the failures were actually in the tube surfaces and not in proximity to any welds, dissimilar or not. The Ågesta people believe this to be a case of faulty tube material manufacture, since the exposure conditions were pH 9.5-10.5, chloride concentration <0.5 ppm, and stress <20% of yield, and since the manufacturer has been unwilling to divulge the results of metallurgical examinations of samples returned to him after the failures occurred.

182. *Provide the design criteria and the loading conditions that will be used in the design of the vessel support and vessel internal structure and support.*

Answer: The vessel supports will be designed for the dead loads in the vessel and internals, lateral and vertical seismic loads due to effects of lateral ground acceleration of 0.1g and vertical ground acceleration of 0.067g, piping reaction effects, and control plate scram reactions. The vessel internals and support will be designed for all of these loads and effects plus the effect of the maximum pressure drop across the core. In addition, the internals support structure will be designed for minimum deflections and deformation to maintain core and control plate alignment.

183. *How much field welding must be performed? Will these field welds present any difficulties from the point of view of preheat, stress relief, or inspection?*

Answer: Except for the possibility of a thin membrane seal, at the bottom flange of the vessel to the pool liner, and the piping connections to the vessel, there is no field welding on the vessel. No particular problems are anticipated from these welds.

184. *How will the manufacturer prepare and inspect the edges of the sections to be welded?*

Answer: In general, all weld preparations will be machined to dimensions shown on the shop drawings. If any powder burning is done the cut edges will be machined back to remove all heat-affected zones. All weld preparations will be dye-penetrant inspected.

185. *Has the vessel manufacturer worked out procedures for weld repair of plates, forgings, and the welds proper? How will a weld repair be stress relieved? How will it be inspected after stress relief? Will corrosion testing of a simulated repair job be performed?*

Answer: These are construction details yet to be resolved between the vessel manufacturer and ANL.

186. *Discuss the differences, if any, between the AARR and the HFIR safety instrumentation and control systems.*

Answer: Although the AARR safety instrumentation and reactor control systems are similar to their HFIR counterparts, they differ in the following principal respects:

- (1) AARR does not employ "Reactor Heat Power" either as a scram parameter, or for initiation of corrective action in the reactor control system. This is based on present information which indicates that the nuclear channels employed in the reactor safety and control systems will provide a sufficiently close measure of reactor power throughout the core cycle. The precision of the neutron flux measurement of reactor power will be increased by periodic recalibration of the flux control and safety channels.

- (2) Neutron flux instrument calibration adjustments to match thermal power will be accomplished manually in AARR. In HFIR this adjustment is made continuously and automatically. The maximum expected change in calibration, less than 10% per core cycle, can readily be compensated on a periodic basis. In addition, failure to recalibrate these channels during a core cycle would result in the reactor automatically running at reduced power and with lower power trip settings.
- (3) In AARR, the ionization chambers associated with the neutron flux safety channels will be separate from, and in different thimbles from, the chambers used with the automatic flux control system. The HFIR installation combines a control and a safety channel detector in a single combination unit.
- (4) In addition to the scram parameters used in HFIR, AARR also employs "Primary Coolant Pressure High" and "Seismic Disturbance," to initiate reactor scram.
- (5) The AARR relay scram circuits are presently arranged to initiate a scram only upon tripping two instrument channels measuring the same variable (see Figure IV-35 of Part 1, Section IV-H). The HFIR scram circuits are so arranged that the reactor is scrammed if any two channels that are connected in different scram circuits (i.e., different variables) are tripped. The AARR approach results in a lower probability of spurious scrams without reducing the protection of legitimate scrams.
- (6) The HFIR nuclear instrumentation utilizes wide-range logarithmic counting channels, having automatically repositioned flux sensors over the full range from source level to full power. The AARR system, as described in Section IV-H-1 of the 12/9/66 revision of the PSAR, utilizes log-count-rate channels in the lower power range, in conjunction with logarithmic current channels for the intermediate and power range. These channels employ fixed detectors. There is at least one decade overlap between the ranges covered by the counting and current channels.
- (7) At AARR, the shim-regulating cylinder as well as the shim-safety plates are automatically driven into the reactor following a scram signal, (see Part 1, Section II, Criterion 12.2). The HFIR control scheme provides for running in only the shim-safety drives following a scram, and then only if each rod individually has seated.

187. *Operating bypasses and channel bypasses were not addressed in the PSAR and supplements. Discuss the planned use of bypasses in AARR and the manner in which they will comply with sections 4.11 through 4.14 of the proposed IEEE standard for the reactor protective system.*

Answer: As of the present design there are no operating or channel bypasses employed in the reactor (nuclear) safety system instruments and circuits. Three instrumentation channels are provided for monitoring each scram parameter, arranged in two-out-of-three trip logic. Thus, any one of the three can be tripped for testing or calibration without the need for a bypass.

The only operating bypass in the reactor control system at this time is used for bypassing the period signals generated by the start-up channels when in the intermediate and power ranges. This bypass occurs automatically as described in Part 1, Section II, Criterion 15.8. If provision for additional operating modes is included, several operating bypasses, associated primarily with primary coolant pressure and flow, will be required. These will comply with Part 1, Section II, Criterion 15.9, and Sections 4.11 through 4.14 of the IEEE proposed standard. The actual components and circuits will be established in Title-II design.

188. *Does AARR have any trip settings which have to be changed for different modes of operation? If so, how do the applicable systems meet section 4.15 of the above mentioned proposed IEEE standard?*

Answer: Only one operating mode is included in the design at this time, and there are no scram trip settings that must be changed during normal operation. If additional operating modes that require different trip settings are introduced later, the provisions for changing the trip settings to suit the operating mode will satisfy Section 4.15 of the IEEE proposed standard. The details of the circuits necessary to ensure proper operation will be developed during Title-II design if and when the additional operating modes are adopted.

189. *Criterion 21.1 in the PSAR states that the two separate Class III power feeder systems will be isolated from each other to prevent common modes of failure. With the automatic switching feature described as a part of this system, it is not clear how this criterion is being met. On the basis of a fault analysis of the automatic bus transfer system and the associated buses, indicate how this criterion is being met.*

Answer: A revised one-line electrical diagram (Fig. IV-33 of the 12/9/66 PSAR revision), showing modifications which have been made to split the 13 KV supply bus at AARR, is shown on the following page. No automatic transfer is involved with the present arrangement.

Fault analyses are performed as part of Title-II design, and are not available at this time.

190. *The system description indicates that Class II power is fed into the reactor systems through the same two buses used to carry the Class III power. By means of a fault analysis of common portions of both Class II and Class III systems discuss the potential for simultaneous loss of Class II and Class III power.*

Answer: Fault analyses are part of Title-II design, and are therefore not available at this time. It should be noted that there conceivably could be cases of simultaneous loss of both Class-II and Class-III power, although the probability is extremely remote. Therefore, all items essential to plant shutdown and containment are supplied with Class-I power.

191. *Discuss the maximum rate and amount of reactivity that may be added if the hydraulic shock absorber action were to fail, and/or the drive plate were to break, and the shim-regulating cylinder were to fall by gravity? (Part 1, Section VI-B-2)*

Answer: Assuming that the regulating plate falls by gravity from an initial normal operating position of vertical symmetry with respect to the four shim-safety plates, the maximum rate of reactivity addition is calculated to be \$30/sec, and the maximum amount of reactivity added would be 6-1/2 dollars. These values are based upon Fig. 7.3.7 of ORNL-3572,* assuming that the worth of the shim-regulating plate is equal to the total worth of the four shim-safety plates.

In arriving at these data, it was assumed that no shim-safety-plate scram occurs during the incident and that the net reactivity addition is not limited by reactivity feedback effects.

In fact, a high-flux trip would scram the four shim-safety plates at an initial acceleration of 4g, and would reduce the net maximum reactivity insertion rate. The large negative reactivity coefficient of fuel-plate heating and ejection of coolant fluid from the fuel zone would also be effective.

192. *Describe the method of actuation of the mechanical syphon breaker as well as its subsequent operation for a rupture in: (a) the inlet coolant line below the pool floor; and (b) the outlet line below the pool floor. (Part 1, Section VI-D-2b)*

Answer: The syphon breaker mentioned in Part 1, as currently conceived, will consist of a small, open line which cross-connects the top of the reactor outlet line gooseneck with the reactor inlet line at a point above the elevation of the gooseneck. This is shown schematically in the figure on the following page. During normal reactor operation, this line will bypass a small fraction of the total primary system flow to the reactor, but sufficient reserve capacity is being designed into the

* F. T. Binford and E. N. Cramer, "The High Flux Isotope Reactor - A Functional Description," ORNL-3572 (Volume 1) (May 1964 - revised March 1965).

SCHEMATIC ARRANGEMENT OF REACTOR VESSEL
SYPHON BREAK LINE

system to handle this diverted flow without adversely affecting the cooling of the core and other reactor internals.

The following amplification of the material presented in Part 1, Section VI-D-2b is provided to illustrate the function of the syphon break line in preventing complete drainage of the reactor vessel by syphoning action.

(a) Rupture of the Inlet Coolant Line below the Pool Floor

Decompression of the system would be completed with the almost-instantaneous ejection of ~50 gallons of 120°F water from the system through the rupture. A momentary disturbance of coolant flow in the primary loop might occur during the depressurization period, and the flow would be markedly reduced because of the reduced primary pump efficiency at the lower NPSH which would then be available. (see Part 1, Fig. VI-D-3) However, following depressurization, flow would continue (or resume) through the reactor and syphon break line in parallel, in the normal direction of primary loop flow, because of continued pump operation and because of initial syphoning. (The latter would result from the difference in gravity head between the water in the reactor outlet line and in the inlet line above the point of rupture; this section of inlet line would now contain some air drawn in through the rupture by the continued flow and by displacement of water draining from the inlet pipe through the rupture by gravity.)

This condition would continue until the inlet line were completely devoid of water. Air would pass from the inlet pipe through the syphon break line to the top of the outlet line gooseneck, and the syphoning action would cease. The water in the outlet line downstream of the gooseneck would drain at least to the elevation of the inlet line rupture, and to a point below this depending upon the NPSH at which the primary pump efficiency becomes zero. The water drained would be replaced by air entering the gooseneck through the inlet line and syphon break line. The water in the reactor vessel would drain over the top of the outlet line gooseneck, being replaced by air entering through the inlet line, until the reactor vessel level had dropped to the elevation of the top of the gooseneck. At this point, the situation would become stagnant, with the reactor vessel still filled with water to a point well above the elevation of the top of the core.

(b) Rupture of the Outlet Coolant Line below the Pool Floor

Again, decompression of the system would be completed with the almost-instantaneous ejection of ~50 gallons of water here at the normal reactor outlet temperature of 156°F, from the system, through the rupture. Flow through the reactor vessel, and through the syphon break line would continue because of continued pumping action, but at a reduced rate because of lower primary pump NPSH available. This would continue until the water in the primary piping between the point of rupture and the primary pumps had been forced through the pumps and the pumps had run dry; an equal amount of water would be ejected from the outlet line coming from the reactor at the point of rupture.

It is probable that at this point the primary pump discharge check valves would close and maintain a water-filled reactor inlet line. This would prevent any syphoning action. The water lying in the reactor outlet line between the top of the gooseneck and the point of rupture would drain by gravity, being displaced by air entering at the rupture. This air would then pass through the syphon break line to the reactor inlet piping and would allow the water in the reactor vessel to drain over the top of the outlet line gooseneck, until the vessel level had dropped to the elevation of the top of the gooseneck. The situation would then become stagnant, with the reactor vessel still filled with water to a point well above the elevation of the top of the core.

It is possible that at the time at which the primary pumps are postulated (above) to run dry, the discharge check valves would not close, but that the pumps would continue to force an air-water mixture into the reactor inlet piping. If this were the case, the gravity head of the air-water mixture in the inlet line might drop below that of the water in the outlet line between the top of the gooseneck and the point of rupture. This would tend to initiate a syphoning action which potentially could drain the reactor vessel through the outlet line. However, the syphon break line would prevent this by allowing air to pass from the inlet to the outlet line, to displace the water in the outlet line downstream of the gooseneck and allow it to drain by gravity through the rupture. Thus, the syphoning action would be aborted and the situation would revert to that described above wherein the pump discharge check valves do close.

193. *Clarify the pipe rupture accident analyses in the following areas:*

(a) Indicate the experimental/analytical bases for concluding that flow will continue in the high heat flux environment of the AARR flow channels following a depressurization and flow reduction.

Answer: As pointed out in the final paragraph of the Introduction to Part 1, and again in the opening paragraph of the Safety Analysis and Evaluation (Section VI), this report, as modified to reflect incorporation of the HFIR system, depends heavily upon the final safety analyses performed on the HFIR, as reported in ORNL documents (References 5 through 8, Section 1, Part 1). This was because the decision to adopt the HFIR system was relatively recent, and the Laboratory had not yet had the opportunity to perform thorough independent analyses on all aspects of the plant. Moreover, in view of the successful operation of HFIR, the Laboratory did not (and does not) believe this to be necessary at the preliminary safety analysis stage.

In essence, the Laboratory has assumed (1) that the HFIR system is acceptable as constructed, approved, and operated, and (2) that AARR components and system conditions which are identical or nearly identical with HFIR would be similarly acceptable, at least for preliminary analysis purposes. Prior to the compilation of the AARR Final Safety Analysis Report,

independent analyses based upon the final designs will be made as necessary to assure that all current safety standards are adequately met.

The above remarks apply equally to both parts of this question, as well as generally to many of the other questions treated in Part 2, Section III of this report.

With specific reference to part (a) of the question, no exhaustive analyses or experiments have been performed at Argonne to serve as the basis for the tentative conclusions stated in Part 1 regarding continuation of flow following a major pipe rupture and system depressurization. In the case of major ruptures within the Reactor Pool (see Part 1, Section VI-D-2b), the system retains the benefit of the pressure caused by the head of water in the Reactor Pool. From analyses presented in the HFIR safety documents, as well as brief consideration of fuel surface temperatures, coolant temperatures, and reduction in primary flow which would result from pump operation at low NPSH values, the conclusion was reached that flow would continue in AARR in these cases, that the reactor would be shut down in the order of 1/2 sec after rupture, and that no core melting would occur.

In the case of ruptures outside the pool, similar considerations led to the conclusion that the reactor would be shut down in the same order of time after rupture, and that flow would continue at least during the period of shutting down - possibly as long as several seconds. It was further concluded that probably only the hot spot areas (~5% of the core) would melt under these circumstances. Some data from natural circulation tests run at Argonne (see Reference 16 of Section VI, Part 1) were used to support this conclusion.

It is possible that steam formation within the core near the coolant channel exits might cause flow reversal in some of the hot channels, but the conclusions in Part 1, Section VI-D-2b regarding the length of time required to heat these regions to melting temperatures are still valid. Indeed, the formation of steam voids in fuel channels would supplement the control plate action in achieving rapid reactor shutdown; this would tend to limit the extent of melting and might even prevent it entirely.

Nevertheless, for added safety assurance, a 100% core melt, plus a 100% metal-water reaction, were assumed as the basis for MCA dose and containment overpressure calculations.

(b) Discuss the "scramability" of the safety-control segments during depressurization-blowdown conditions.

Answer: (Note that the opening paragraphs under part (a) above apply to this response also.)

The blowdown period immediately following a pipe rupture in the AARR primary system (or in any reactor system) is a complex situation involving a multitude of parameters and one for which no rigorous analysis is presently feasible. The listing in Part 1 of important factors

during blowdown (Table VI-D-2) as well as the inclusion in the AEC reactor development effort of an extensive program (LOFT) to examine such effects experimentally, support this assertion.

For major pipe ruptures such as those discussed in Part 1, Section VI-D-2b, the time scale for rupture, depressurization and control element action is unlikely to exceed a few hundred milliseconds. During this period, the control system may experience very unusual loading conditions. It has first been assumed that no essential component of the control system has been damaged by the undefined phenomena which caused the rupture. During a rapid depressurization, the system could be subjected to complex hydraulic loading resulting from reflection and transmission of compression and decompression waves. Such effects have been predicted for the subcooled portion of the LOFT blowdown using the BURST code.* Since this code is not yet available for use on AARR, and since the design of major primary system components is at only a preliminary stage, a realistic estimate of the magnitude and duration of such loads has not been possible. A preliminary approach, however, has indicated that considerable potential exists for scrambling the control elements against these forces.

It has been assumed in the case of reactor inlet line rupture that (1) after rupture the pressure in the upper plenum would drop to 1 atm while the lower plenum would remain at its initial steady state level, and (2) a flow reversal and steady hydraulic forces corresponding to these pressure differences would instantly appear (the actual differential pressure acting on the control elements would be considerably reduced from this value due to the presence of flow control orifices). Although these forces are very conservatively evaluated, calculations indicate that the spring driven control elements would still travel about 6 in. before they would be halted by the reversed flow. Reactor shutdown cannot be assured on this basis but it should be noted that these hydraulic forces would be accompanied by high coolant velocities in the core, which would reduce the tendency toward melting during the blowdown period. It is also instructive to observe that the generation of the scram signal, magnet release, and element travel to this hypothetical stalling point would consume approximately 200 msec, a period long enough to allow a significant, or possibly even complete, dissipation of the postulated forces.

A major rupture of the reactor outlet line appears to involve less serious forces on the elements since the direction of coolant flow through the reactor tends to remain the same as during normal operation.

More detailed analyses of blowdown effects will be made for the Final Safety Analysis Report.

* "STEP Project Quarterly Technical Report," IDO-17186 (September 1966).

194. *Discuss the capability of the primary coolant loop components to withstand postulated severe reactor accidents (other than accidents initiated by a pipe rupture).*

Answer: The containment potential of the primary coolant system is a function of the bursting pressure and the incremental volume of the system. In the case of piping, the containment potential is a function of the wall thickness, diameter of the pipe, length of the pipe and the ultimate strain of the piping material. The geometry of the piping system and components must be well defined in order to analyze for containment potential. As stated in Appendix E of Part 1, the mechanical features of the primary coolant system are not yet sufficiently defined to determine accurately or completely the distribution of energy absorption throughout the system or the total energy absorption capacity of the system. Once the AARR primary system is sufficiently defined to include the heat exchanger tubes, strainer, piping, and other components, calculations will be made to ascertain the total containment potential.

195. *What are the design bases and performance requirements for the poison injection system?*

Answer: The poison injection system will be designed to inject a sufficient quantity of neutron poison in the form of a water solution of cadmium nitrate into the primary water to shut the reactor down from any normal operating condition and to maintain subcriticality even with all control plates in their position of minimum effectiveness. The system will be capable of accomplishing this against full operating pressure even in the event of failure of normal (Class-III) utility electrical power. Addition of poison solution will be manually initiated from the control room. Total addition time will be based on the time required for the primary coolant to make one complete circuit of the primary system piping at a flow rate equal to that provided by one primary pump operated by its pony motor. This rate will be more than sufficient to compensate for the gain of reactivity during cool-down of the reactor after shutdown from equilibrium operation.

The mechanical arrangement of the system is indicated in Fig. IV-32 of Part 1, and is basically the same as the HFIR system except that capacities have been increased.

196. *Generally sufficient information has been supplied on the consequences of a rupture of a main primary coolant pipe. However, further analyses should be supplied to indicate the degree of core damage that might occur owing to a failure in smaller pipes connected to the primary system. These analyses should also cover the capability provided to maintain core cooling for breaks at various locations.*

Answer: The discussions relating to pipe ruptures presented in Part 1, Section VI-D-2b are limited in scope (to the main coolant pipes) for a variety of reasons. Primarily, however, it is felt that a quantitative analysis of the consequences of ruptures of smaller pipes connected to the primary system cannot be accomplished until the Title-II design has

fixed the characteristics of auxiliary systems and the associated locations of connections to the primary coolant loop.

Comprehensive information needed to provide a basis for logical decisions in these matters has not as yet been derived. However, toward this end, the development of an analog computer program, which will simulate the major features of the AARR plant and associated control systems, is presently in progress. Also in progress are investigative studies of available computer codes which are designed to calculate the blowdown and subsequent behavior of pressurized water reactors, hopefully to match results with the LOFT experiments. Examples of such codes are the FLASH program developed by Westinghouse and modifications thereto, e.g., those made by Phillips Petroleum Company in conjunction with the LOFT program. Therefore, as the AARR design proceeds during Title-II it is expected that the computational tools necessary to conduct detailed system rupture studies will become available at Argonne and that the results of such studies will be included in the Final Safety Analysis Report.

197. *Criterion 5.2(b), Part 1, Section II, indicates that experimental facilities could not significantly worsen the doses of the design basis accident. However, it is not evident that potential experimental materials, such as plutonium, were included in the radioactive inventory used to estimate doses for the design basis accident. Clarify the manner in which the criterion will be met by giving an estimate of the dose contribution that would be caused by experimental materials.*

Answer: The radioactive inventory used to estimate doses for the design basis accident did not include potential experimental materials such as plutonium. All experiments and experimental materials will be reviewed to ensure that they are not likely either to increase unduly the probability of an accident or to increase the severity beyond tolerable limits. The review will include considerations of energy as well as radiation. Both physical location and form of the experimental material will be considered.

As an example, plutonium target plates or converter plates in a beam tube would in general probably be from 8 to 12 ft from the outside of the core. Since it is considered extremely unlikely that even complete core melting could affect such plates, more material would be allowed here than will be allowed in internal rabbit samples. If a rabbit target material is to be a powder, a higher dispersal factor would be applied than for a solid; for stable oxides the factor would be lower than for combustible metals. Daughter materials as well as original target materials will be considered.

Review of all irradiation and experiments will be a formal procedure, and written records will exist for all cases examined.

198. *Discuss the specific details of the mathematical model used to calculate doses associated with the release of radioactive material from the Reactor Containment Building (RCB) and the stack. Include all values used for the variables involved*

for short term releases, intermediate term releases, and long term releases.

Answer: The specific details of the mathematical model used to calculate the doses are supplied in Part 1, Section VII-B and in Reference 5 of Section VII. Reference 5, a report on the reactor-siting program AISITE II, supplies details of the models computed by AISITE II. The AISITE II program was not modified in performing the analysis for the AARR.

199. *For the fuel temperature and extent of core damage associated with the design basis accident, it is questionable that the release of "solid" fission products should be limited to 1% as assumed. This assumption should be re-examined in the light of recent experimental studies on fission product release.*

Answer: The assumption that no more than 1% of the solid fission products are released has been a standard assumption for water-moderated reactors, of which AARR is an example. If there are "recent experimental studies" which cast doubt on this assumption, specific references are requested to provide a basis for re-examination.

D. Questions from Reference (9)*

200. It has been proposed that basic vessel shall be fabricated from inconel, but that the top and bottom flat heads be fabricated from Ni-Cr-Mo low alloy steel (ASTM A-508, Cl 4), clad with weld deposited inconel cladding. The basic reason for proposing to use low alloy steel is presumably a matter of cost and somewhat higher allowable stresses. The natural choice would be A-508, Cl 2 (Mn-Mo, nickel modified, corresponding to A-302 plate). However, the head is better than 15" thick, and it is realized the mill will not guarantee the desired impact properties in this thickness. This led to the selection of A-508, Cl 4 material, as indicated above. Although this material has been Code approved under Section III, there has been sufficient speculation concerning its degradation due to temper embrittlement such that we cannot recommend its use for the vessel proper until this question has been fully investigated and resolved (i.e., for the cylindrical section of the vessel or other parts involving strength welding and multiple stress reliefs.)

In using the material in the form proposed for AARR, it appears that the welding is limited to cladding and deposit of inconel inlays to which the nozzles are subsequently welded. In this case, it is possible that no more than one stress relief would be required, which would help in avoiding excessive degradation of the material. Nevertheless, there would appear to be enough uncertainty in the use of this material that further study and consideration should be given to using the conventional material, A-508, Cl 2. It is known that this latter material has been used in thicknesses materially greater than 15", ostensibly meeting the required impact properties, although in some cases the holes in the head were rough machined prior to heat treatment in order to enhance the cooling rate (and impact properties) achieved. Therefore, whereas the mills may not be willing to guarantee the impact properties, we believe that there is a good chance that they are actually achievable.

If, upon additional investigation, this appears to be an unacceptable risk, further consideration should be given to the use of domed heads, either in A-508, Cl 2 or inconel. Finally, if it appears that A-508, Cl 4 material is the only reasonable alternative, the potentialities of temper embrittlement must be fully explored before actual fabrication of the heads is undertaken. Also, it is understood that a high heat input in the welding of this material (including the cladding operation) is conducive to the formation of an embrittled weld-heat-affected zone.

For this reason it must ordinarily be welded (or clad) by a manual process using very small electrodes, with rigid control on preheat and interpass temperatures (maintained within the range of, say, 200°F min. to 300°F max.)

The results of this investigation together with final recommendations should be referred to RDT as soon as possible.

*Ed. Note: These questions were raised by RDT after review of Reference (3). They were answered in Reference (10).

Answer: The information requested by RDT has been previously transmitted to AEC-CH in a letter, M. Levenson to A. E. Mravca, "RDT Comments Concerning the AARR Reactor Vessel Phase I Design," dated April 13, 1967. For your convenience, the pertinent material in that letter is reproduced below:

"We have reviewed our original material selection studies and concluded that the use of the A-508, Class 4 material is still a reasonable choice for the closure head forgings. With full consideration of the potential problem of temper embrittlement of the Class 4 forgings, this conclusion is based on the following reasons:

- "1. The two major causes of temper embrittlement are heat treatment during fabrication and welding. Strict controls will be enforced during the fabrication of the vessel to assure fulfillment of the heat treatment and welding process specifications. These detailed specifications will be designed to minimize possible degrading effects of welding on the Inconel overlay and heat treatment of the heads. Specifically, only one heat treatment is planned, which will help to reduce possible degradation of the material. Also, heat input during the cladding process will be minimized, and the preheat and interpass temperatures will be carefully controlled in order to avoid the problem of an embrittled weld heat affected zone.
- "2. Although it is possible that the impact properties of an A508, Class 2 forging for a flat heat might fulfill the design requirements, the cost and schedule risk involved is considerable since the mills will not guarantee this. Data which the vessel vendor has on a material similar to A508, Class 2 - ASTM A336 forgings - indicate Charpy V-notch values as low as 10 ft-lbs at +10°F just three (3) inches below the surface on thick forgings. Although the AARR closure heads are subjected primarily to bending, there are complex stress conditions existing in the area of the quick-opening closure hole and adjacent holes for vertical facilities. Therefore, we feel that it is important for the forgings to possess good impact properties throughout its thickness. To the best of our knowledge, the A508, Class 4 forging will, indeed, possess these desired properties.
- "3. The cost of providing a domed head of either A508, Class 2 or solid Inconel is quite high and the use of a domed head might increase the fabrication time for the vessel.

"The use of a test forging mockup to demonstrate the adequacy of the fabrication processes has been considered, but for the present is not planned. This decision is based primarily on cost and schedule. Delivery time for the A508, Class 4 forging material is about 6-8 months, and the present fabrication schedule cannot be delayed in any

way if the end date is to be met. However, the Inconel cladding weld procedure will be performed on 4-in. thick HY-85 plate, and the vendor will provide sufficient material to enable taking additional Charpy V-notch specimens. The bottom of the vee will be at the interface of the cladding weld and the base metal. The procedure will not be qualified unless these Charpy tests meet the ASME Section III requirements for the base material forgings."

It should be noted that we are attempting to locate an available piece of A508, Class 4 forging material to use for the Inconel clad welding procedure qualification tests, instead of the HY-85 material mentioned above. If we are unsuccessful in this attempt, the HY-85 material will be used as stated, in order to avoid a significant schedule delay.

201. Discuss the reasons for choosing cadmium nitrate over boron compounds. What will be the concentration of the stored poison, and how much cadmium nitrate will be necessary to meet the requirement of Criterion 9.2?

Answer: As stated under question 195 above, the AARR poison injection system is based on the HFIR system in all respects, including the soluble poison compound used, the piping arrangement, and the equipment used to introduce the poison solution into the primary coolant system. Both HFIR and HFBR use cadmium nitrate - $\text{Cd}(\text{NO}_3)_2 \cdot 4 \text{H}_2\text{O}$ - as the soluble poison compound. The major advantage of using this compound is that it is much more soluble in water than any of the boron compounds considered.

The poison storage tank will contain approximately 200 gallons of 44 wt % $\text{Cd}(\text{NO}_3)_2 \cdot 4 \text{H}_2\text{O}$ solution.

202. Criterion 1.4 (Part 1, Section II) states that the AARR facility is to be located in a region of seismic probability designated as Zone I. Is the seismic loading design of AARR different from that of HFIR, and if so, how might these differences affect the design of the control rod drive system?

Answer: Oak Ridge National Laboratory, site of the HFIR, is located in a region of seismic probability designated as Zone I, the same as ANL. However, to our knowledge, no seismic loading criteria were imposed on ORNL in their design of the HFIR control rod drive system. Therefore, differences may arise between the HFIR design and the design for AARR, in which consideration of seismic loadings is required.

An analysis of the HFIR control rod drive system design will be made as part of the AARR Title II design effort, to determine whether that design would still be adequate under the AARR seismic loading condition. Design modifications will be initiated if they are shown to be necessary by the results of this analysis. These modifications would probably take the form of strengthening and stiffening certain components of the system in order to withstand the required loadings. The principles of the safety system would not be affected by such changes.

203. *The design of the fuel handling system and storage facilities seems to be adequate. However, the PSAR lacks a description and drawings of the grappling technique for transporting the fuel elements to and from the core. This is considered to be an important part of the handling system design, and should be made available to the reviewers.*

Answer: The grappling tools for handling the inner and outer fuel elements are of similar design. (See ORNL-HFIR drawings M 10070 ϕ A 200 E, Rev. 2, Inner Annulus Fuel Element Handling Tool Assembly, and M 10070 ϕ A 100 E, Rev. 2, Outer Annulus Fuel Element Handling Tool Assembly.) The tools have three cams which are positively locked in a groove in the inner fuel element, and below the fuel inner liner on the outer fuel element. The tool is operated from the personnel bridge and suspended with a dynamometer attached.

204. *Comment on the desirability of checking all new fuel elements for reactivity as a standard procedure. The reactivity measurement is simple and provides a sensitive and last check on the combined fissile-poison loading. The method used by ORNL for this purpose might be adopted by ANL.*

Answer: ANL will perform a final reactivity measurement, but not necessarily in a separate facility. The reactivity measurement may be done as a part of the loading sequence.

205. *Elaborate on those situations where new elements will not be subjected to physics measurements (Part 1, Section IV-C-2 specifies that new elements may be subjected to physics measurements in another ANL facility to assure specifications compliance.)*

Answer: The PSAR statement that new fuel elements may be subjected to reactivity measurements in another ANL facility, was intended to indicate that ANL was investigating the need for such a facility. It was not intended to imply that such measurements would be made sometimes, but not at other times. As stated above under question 204, final reactivity measurements will be made before a core is operated at power in the reactor.

206. *The sluicing of highly radioactive ion exchange resins presents a number of safety problems which it would be desirable to avoid (passing a pulse of radiation down a pipe, plugged lines, disposal of sluice water, contamination of piping, rehandling of resins again at sluice tank, disposal, etc.). The alternative of using discardable demineralizer vessels with quick disconnect couplings (similar to those specified for beam tube shield plug cooling water supply) should be analyzed. Another alternative is to employ replaceable vessels with discardable cartridges or baskets of resin. Experience with sluicing systems has been very bad and experience with discardable demineralizers has been good. If this system is redesigned, operating these demineralizers in a pool of water for shielding purposes should be considered - waste demineralizers could be stored in the pool for long decay periods before handling. It is also possible to employ an exhausted demineralizer as an upstream filter for a fresh one.*

Answer: In AARR, ion exchange resins are regenerated chemically in place when exhausted. A sluicing system is provided to remove and replace resins only under circumstances where the resins have been damaged and are no longer capable of regeneration for further use, or where regeneration in place would result in a significantly greater waste disposal problem than would replacement of the resin.

Damage to resins can result from operator error in allowing resins to be exposed to excessive water temperatures, or to excessive concentrations of regenerant chemicals during the regeneration process. The provision of adequate alarms and detailed operational procedures is considered sufficient to prevent such damage under all but the most abnormal circumstances. Resin damage can also occur as a result of deterioration with time, but this is such a slow process that resin replacement for this reason alone is not expected to be necessary more than a few times during the entire life of the plant.

Resin damage can also result from exposure to very high radiation levels, as in the event of a massive fuel failure and gross contamination of primary water. In this case, however, resin replacement would probably be desirable for reasons other than resin damage. Regeneration of primary demineralizer resin produces about 6000 gal. of active liquid waste, and the activity level of this waste would be very much higher after a fuel failure than under normal conditions. Under these circumstances the handling of the contaminated solid resin, plus the relatively small amount of contaminated water required for sluicing, would probably be preferred. The volume of water used for sluicing would be far less than that of the chemical solutions required for regeneration, and the level of contamination in the sluicing water would be much lower because most of the activity would remain adsorbed on the solid resin particles. Moreover, this water would not be released, but would be retained in the recirculating sluicing system, and could be cleaned up at a later time.

The use of a well-engineered sluicing system has been evaluated in comparison with disposable resin cartridges or baskets, including consideration of the radioactive handling problems which would arise in each case. For the particular physical arrangements which exist in AARR, the sluicing system has been adjudged best and has been selected for development in Title II design.

207. Some pH cells employ a salt bridge which apparently can release chloride to the primary water. Since this could contribute to corrosion in a stainless steel and aluminum system, discuss the potential for chloride stress corrosion from this source. Are there any alternatives to the use of these pH cells?

Answer: Both the pH cells and the oxygen analyzers present a possible source of contamination of primary system water. Therefore, the effluent from these instruments is first passed through a small mixed bed demineralizer, and is then discharged to the primary degasifier tank. Since all water from the degasifier normally flows through the main primary purification system before entering the makeup storage tank for return to the high pressure primary loop, a double protection

against coolant contamination by the pH cells and oxygen analyzers is afforded.

208. *High off-gas activity will close the degasifier block valve and no gases will be stripped. It might be preferable to continue stripping, but divert the gas to a holdup tank or gas bottling system. With no provision for gas storage, the only way radioactive gas can leave the plant is through the stack. As a temporary measure, storing fission gas in the primary loop appears to be a good idea but reliance only on dilution in the stack and beyond may require long release times and delay opening of the reactor vessel for core or element replacement.*

Answer: The use of holdup tanks or a gas bottling system for retention of radioactive gas released from the primary coolant was considered in the earlier stages of AARR design, but it was concluded that the expense of the necessary compressors, pressure tank, and control equipment was not justified. The likelihood of release from the core of quantities of gas sufficient to present a problem upon discharge from the stack, after holdup in the primary system for a reasonable period, appeared extremely remote. Moreover, such a large release would be indicative of major core failure which would almost certainly require an extended shutdown of the reactor for analysis of the situation and planning the repair operations, before the primary system were actually opened, and this time would be available for decay of the gases held within the system.

209. *The blowdown line from the cooling tower is a point where control over a potentially contaminated stream may be lost. Please consider the need for a radiation monitor at this point.*

Answer: Our present design calls for monitoring the level of radiation in the secondary coolant line leading to the cooling tower. If the level of radiation in the line exceeds a preset value, this monitor will automatically close valves both in the blowdown line and in the cooling tower basin makeup line. In addition, an automatic sampler will sample the combined blowdown and basin drain line for periodic evaporative analysis.

210. *A provision for periodic testing under load should be included for battery (Class I) power systems. Recent experience may be cited at NRTS where an emergency battery power system provided only a small fraction of the expected endurance.*

Answer: Our current plans for periodic preventive maintenance and testing of battery (Class I) power systems include:

1. Frequent (e.g., once per shift) testing of primary pump pony motors and associated Class I power supply systems by brief application of full load current to the motors.
2. Periodic testing of battery voltage.
3. Periodic testing of electrolyte specific gravity in each battery cell.

4. Periodic application of an equalizing charge to the battery banks to assure that all cells remain equally charged.
 5. Annual full-load discharge test of all battery banks.
211. *Discuss the advisability of using throwaway or replaceable cartridge demineralizers (in the Radioactive Waste Retention System), to eliminate the complex and possibly hazardous handling of resins.*

Answer: The discussion under question 206 above, applies to all major demineralizers in the AARR facility which are subject to radioactive contamination, including those in the Radioactive Waste Retention System.

212. *The maximum credible reactivity accident envisioned is massive voiding of the ITC. It is presumed the voids are already contained in the bulk coolant above the core and are carried downward through only the ITC by the flow through the ITC. What is the mechanism for such an occurrence? Development of an internal vortex far below the flow entrance with hydraulic interference which would be imposed by the ITC experiments seems quite improbable.*

Answer: We have not been able to develop a credible mechanism by which massive voiding of the ITC could occur. We agree that the formation of an internal vortex is quite improbable, and we have taken steps in the design of the ITC assembly to prevent such an occurrence. However, we did not believe that we could positively rule out any ITC voiding accident, and because of the difficulty of defining a particular level of credible voiding, we chose to analyze the "worst conceivable" voiding case, i.e., that which would produce the greatest reactivity addition, on the basis that any realistic voiding case would have a lesser effect than this worst case analyzed.

213. *Discuss the reactivity-control worth criterion (mentioned under question 139 above) for the backup shutdown system (Criterion 9.2). Compare this criterion to the requirement that the negative reactivity worth of the backup system should exceed the maximum loaded reactivity of the core and show why this latter requirement would not be feasible for AARR. If possible, estimate the poison requirement for this alternate criterion, and compare this to be the amount of poison solution needed to satisfy Criterion 9.2.*

What provisions are there for testing or monitoring of this system, to establish the operability of all equipment?

Answer: Criterion 9.2, Part 1, Section II, should be interpreted to mean that the uniform admixture of all of the available soluble poison solution throughout the primary coolant system would result in a poison concentration in the primary coolant sufficient to keep the reactor subcritical, even with the reactor at its maximum reactivity under normal conditions (i.e., in the as-loaded condition with the primary coolant at room temperature), and with all control elements withdrawn to their positions of minimum reactivity control effectiveness.

The following provisions are made in the design for testing and monitoring operability of the backup reactivity control system. As shown in Figure IV-32, Part 1, two remotely operated ball valves are provided in the line between the poison solution storage tank and the juncture of the line with the pressurization pump suction header, and a third remotely operated ball valve is provided to drain the section of line between the two in-line ball valves. Under normal reactor operation, the two in-line valves are closed and the drain valve is open; the manual control in the control room reverses the positions of these valves in order to admit poison solution to the pressurization pump suction header for injection into the primary loop by these pumps. Leakage through either in-line valve is detected as an effluent from the drain line.

For periodic system testing, two locally hand operated block valves (normally locked open) are installed, one upstream and one downstream of the two in-line ball valves. (These hand operated valves are also shown in Figure IV-32.) By selective setting of the hand operated valves it is possible to check the operation of the ball valves, to determine sources of leakage through the in-line ball valves, and to calibrate the orifice in the poison solution supply line, all without actually injecting poison solution into the primary system and disrupting normal operation. Detailed procedures for such periodic testing will be incorporated into the AARR Operating Manual.

214.* *With reference to question 140 above: (a) Argonne has analyzed ruptures of the inlet and outlet coolant lines, and concluded that the core remains covered in both cases. Evaluation of the former case, however, shows that the ability of the core to remain submerged is dependent on the operation of a siphon-breaker in the gooseneck of the outlet line, which acts to terminate flow from the vessel. Since the operation of this siphon-breaker is critical to the course of events following a rupture of the inlet line, it should be subject to intensive review. Similarly, operation of the check valves in the pump discharge line is critical to the events following a rupture of the outlet line, and should be thoroughly evaluated.*

Answer: A description of a possible design concept for the syphon-breaker is provided under question 192 above. Discussions are provided which describe syphon-breaker action subsequent to rupture of inlet or outlet main coolant lines outside the Reactor Pool. Final design of the syphon-breaker will be made during Title II. Failure of the pump discharge check valves to close following a rupture of the outlet line is also discussed in the referenced material and shown to be immaterial with regard to the potential for uncovering the core.

(b) ...in both cases (inlet and outlet ruptures), consideration should be given to the possibility that additional water might leave the vessel due to the original momentum of the coolant stream, and the continuing (although limited) operation of the pumps. What is the likelihood that the coolant stream contains sufficient momentum to render the siphon-breaker ineffective?

*Ed. note: This question was originally transmitted to ANL in Reference (11), which indicated that no response was necessary. However, Reference (9) requested a formal response which was given in Reference (10) and is repeated here.

Answer: Preliminary flow coastdown analyses have indicated that the original momentum of the primary loop coolant is relatively insignificant compared to that stored in the primary pumps and motors (see Part 1, Section VI-D-1a). To substantiate this conclusion, a calculation of primary coolant flow decay, based upon dissipation of the original momentum of the coolant, was performed. The results indicate that the residual flow becomes insignificant within a period of time of less than 10 seconds. Integration of the flow decay curve over this time span yields a total volumetric delivery of coolant of only ~10 percent of the vessel volume above the level of the outlet pipe gooseneck, which may be conceded to be lost due to continued pump action or gravity flow and yet maintain a submerged core. Thus, it is concluded that the original coolant momentum provides no significant additional contribution to the total coolant loss already assumed in the PSAR.

(c) Argonne's conclusion that the core remains covered appears based on qualitative arguments, yet is critical to the safety of this reactor. In this connection, consideration should also be given to possible means of reflooding the reactor vessel, such as with reactor pool water, should the possibility of core uncovering not be able to be completely ruled out.

Answer: Provision for direct communication between the reactor vessel and pool, as a means of vessel reflooding in the event of a gross loss-of-coolant accident, continues to be a problem requiring further design study. Considerations related to this problem are described in Part 1, Section VIII-C. A schematic representation of a possible design concept for accomplishing this function is depicted in Figure VI-D-2 of Part 1.

(d) Similarly, the statements that hot spots constitute no more than 5% of the active fuel plate volume, that melting would be limited only to hot spots, and that a metal-water reaction would be limited to the hot spot aluminum content require further support in the light of uncertainties in the fuel heat-up rate, mechanical behavior during the establishment of reversed coolant flow by convection and the capability of natural convection flow to remove sufficient heat to limit melting to 5% of the core.

Answer: With respect to the PSAR statement, i.e., that the core hot spots constitute no more than 5% of the active fuel plate volume, this has been extracted directly from ORNL statements on the design of the HFIR core. See, for example, page 18 of ORNL-3573.* Regarding the question limiting the extent of meltdown and subsequent metal-water reactions to the core hot spots, it should be noted that the PSAR conclusions in this respect are based upon a consideration of the "most likely" sequence of events following a gross loss-of-coolant accident. This is clearly indicated by the discussions in Section VI-D-2b of Part 1; there the inability to perform a decisive, quantitative analysis is conceded, and the consequences of more extensive core meltdown and associated metal-water reactions are examined. In addition

*F. T. Binford, "The High Flux Isotope Reactor-Accident Analysis," ORNL-3573-Draft (February 1965).

to this cited material, further discussions relating to this problem area are presented under question 193 above.

It should be noted, however, that despite all of the arguments presented against the likelihood of extensive fuel melting, the Maximum Credible Accident analysis presented in Section VII of Part 1 is based upon assumptions of complete core meltdown, complete metal-water reaction of the aluminum content of the fuel plates, and complete combustion of all evolved hydrogen. Since the plant is shown to meet normally accepted standards of reactor plant safety even on the basis of these extremely pessimistic assumptions, a substantial extra measure of assurance is provided that the facility could not unduly affect the safety of the public under realistic accident conditions.

(e) Argonne has shown that even in the event of a 100% core meltdown, off-site personnel would not receive exposures greater than those specified in 10 CFR 100. It would seem appropriate to limit doses calculated to be received by Laboratory personnel, considering post-accident personnel control measures which would be taken, to the emergency values specified by 10 CFR 100. Any such measures planned should be fully described including such considerations as training, practice, equipment requirements, and overall reliability of the plans.

Answer: It is shown in Table VII-A-2 of Part 1, that Laboratory personnel situated at distances greater than 350 meters from AARR would receive less than the limiting doses suggested in 10 CFR 100* during any two-hour period following occurrence of the assumed Maximum Credible Accident in AARR. From Figure III-A-2 and the tables in Section III-B, Part 1, it can be seen that the area having a 350-meter radius surrounding AARR has a Laboratory personnel population between 250 and 300 persons, exclusive of personnel housed within the AARR complex itself. All but about 80 of these persons are located at radii between 240 and 350 meters, wherein the whole-body dose is still below the two-hour limiting dose of 10 CFR 100, and the two-hour inhalation dose to the thyroid is no more than 10% higher than the 10 CFR 100 limiting dose.

Established Laboratory evacuation procedures will certainly be sufficient to assure removal of all persons from within the AARR exclusion area (1500 meters radius), which lies entirely within the Laboratory site, in substantially less than two hours. These procedures are generally described under question 83 above. As pointed out in the latter reference, the Laboratory rush-hour exodus, which is essentially uncoordinated, is completed 15 minutes after the close of work, and this time could be shortened considerably by proper emergency coordination. Special measures, such as shielding walls, are provided to protect AARR personnel in the Laboratory and Office Building during the period estimated as necessary for their evacuation, and special procedures will be established to assure that the evacuation can, indeed, be accomplished within this period.

*25 rem whole-body dose; 300 rem inhalation dose to the thyroid.

For the 30 persons who might be within the Reactor Containment Building at any one time, three airlocks, widely separated in the building, are provided for egress. It is likely that even these people will be able to leave the building, and the area, before receiving more than the 10 CFR 100 limiting doses, since in the realistic case the buildup of radiation in the building will be relatively slow. (In the assumed MCA, instantaneous dispersion of fission products throughout the building was assumed.)

215. *The fuel handling techniques (described under question 145 above) are satisfactory, but a drawing of the grappling method, as noted (in question 203) above, should be provided.*

Answer: See question 203 above.

E. Questions from Reference (12)*

216. *Question 3(b) above - Reference to Criterion 5.2 in the revised PSAR is not an adequate answer to this question. Quantitative limits should be set on the allowable reactivity that may be introduced via the experimental facilities. Discuss this reactivity insertion and the effects it would have on the accident analysis dose calculations.*

Answer:¹³ We are confused that this question is being asked again. In the RDT attachment to Reference 9, items (1) and (4) under the heading "Other Considerations" state that Criterion 5.2 of the PSAR is satisfactory in applying suitable limits to the experimental program and to the energy potential of experiments. The above question (from DRL) states that Criterion 5.2 is not adequate for these purposes. Please clarify this point so that we may determine how best to answer the question, or whether, indeed, further response is necessary.

217. *Question 52(a) above - Provide a more quantitative discussion of Pu^{239} and Pu^{240} doses resulting from an accident release.*

Answer:¹⁶ The reactor core initially contains 9.4 kg U^{235} in highly enriched U_3O_8 . The U^{238} content is approximately 0.8 kg. In the average core spectrum of the fresh reactor, the rate of production of Pu^{239} atoms by neutron capture in the U^{238} is 1.1 atoms Pu^{239} per 100 atoms of U^{235} fissioned. Thus, in the course of a typical full core cycle of approximately 20 days at 100 MW, ~ 2.0 kg of U^{235} are fissioned and ~ 25 gm Pu^{239} are produced. A small fraction of this Pu^{239} is lost to fission. A still smaller fraction of the Pu^{239} captures neutrons to produce Pu^{240} .

The total inventory of Sr^{89} in the reactor core, after 20 days at 100 MW, is 9.4×10^5 curies, with a total potential dose of 4.2×10^5 rads per inhaled curie. The curie-inventory of 25 grams of Pu^{239} is 1.5 curies, with a total potential dose of 4.1×10^{10} rads per inhaled curie. Thus, assuming that the same percentages of Sr^{89} and Pu^{239} are released to the Reactor Containment Building, the Pu^{239} contribution to the potential radiation dose rate is only one-sixth as much as the Sr^{89} contribution. It is concluded that the plutonium isotopes generated from neutron capture by the U^{238} in the reactor core would contribute very little to the radiation dose rates in the postulated MCA.

In a similar manner, the radiation-dose-rate potentials, under MCA conditions, of proposed target materials will be evaluated. This is discussed in Section VII-C-4, Part 1.

*Ed. Note: These questions were raised by DRL and answered in five separate ANL transmittals, References 13-17.

218. *Question 52(c) above - Describe those design features of the rabbit systems that assure fission products originating in the reactor core cannot escape directly from the containment building via a rabbit system. Consider the system from the viewpoint of accidents (less than the design basis accident), both within the rabbit system and the core.*

Answer:¹⁵ Automatic isolation valves are provided for all rabbit system containment penetrations as shown on Fig. IV-26 and Fig. IV-27 of the 12/9/66 revision of the PSAR. These valves are opened only for sample passage, and are to be control-interlocked to close upon detection of high radiation level in the building ventilation or primary cooling systems. In this sense, there is no different (or special) concern associated with a rabbit system isolation valve than with, say, a ventilation system isolation valve.

219. *Question 58 above - Will AARR use identical manufacturing drawings for the control rods and drives as used for HFIR?*

Answer:¹⁵ It is the intention of ANL to use the HFIR drawings for the fabrication of the control rods and the control rod drive mechanisms. Minor changes may be required based on (a) the operating experience of HFIR, (b) improvements that would ease fabrication, and (c) meeting safety and design criteria of the AARR as given in Section II of Part 1. The control rods will actually be manufactured at Oak Ridge as will some parts of the control rod drives.

220. *Question 102 above - It is not clear that the response to this question has addressed restriction to the channels due to non-preferential bowing. What thermal and hydraulic information is available to evaluate this situation?*

Answer:¹⁵ The only available information on the subject of flow blockage (or restriction) as a result of thermal - and hydraulic - induced plate deflections is given in Section IV-B-1 of Part 1. While the subject of flow blockage (or restriction) is not specifically addressed, it may be inferred that plate deflection under the influence of thermal and hydraulic forces is thought to be non-preferential (i.e., essentially uniform).

221. *Question 105(a) above - What is the basis for setting this trip level at 1.3? What is the design margin? A more complete answer to this question is required.*

Answer:¹³ The basis for setting the trip level at 1.3 was arbitrary, but based on (a) the ORNL-calculated power level for incipient boiling, and (b) the actual trip level setting used in the HFIR.

222. *Question 106 above - Address this question from the standpoint of burnout propagation.*

Answer:¹⁵ The answer to question 106 pointed out the fact that parallel-channel considerations were not of concern for the HFIR core thermal analysis. In view of this, we do not understand the present question,

which presumably is again concerned with parallel-channel effects.

223. *Question 107 above - This answer addresses the gross setback reactor flow. In the case of gross reactor flow transients what analyses have been made considering the response of the core to pulse changes in the reactor flow?*

Another aspect of this question is related to local flow changes. If one or more channels were blocked, how would this be known? Discuss the consequences of blocking one channel, two channels, etc., as asked in question 125 above.

Answer:¹⁵ The above question refers to question 107 above. Question 107 was only concerned with cooling of the beryllium reflector, while the present question (including the reference to question 125) seems to raise new questions concerning the transient behavior of the core. We do not understand the above question in view of the primary reference to question 107.

224. *Question 111(c) above - Provide an analysis for the accidental dropping of a newly spent fuel element on its side including consideration of:*

(a) *Criticality aspects involving*

1. *two elements of the same core*
2. *an element of one core with an assembled second core*
3. *an element falling on its side on top of two spent cores*

Answer:¹⁷

1. Criticality is not possible with two elements of the same core if the beryllium reflector is replaced by water, even in the complete absence of control rods. This is discussed in Part 1, Section VI-C-1. Obviously, if one of the two elements were on its side, the degree of subcriticality would be even greater. These remarks apply whether or not the fixed neutron absorber is in place.

2. The case of an element falling vertically onto an assembled core in storage is discussed in Part 1, Section VI-C-1. It is concluded that the reactivity interaction is negligible. The same conclusion would be drawn if the fallen element were to come to rest on its side. The separation between the meat of the horizontal element and that of the vertical core in storage is at least 6-1/2 in. of water, which is essentially infinite separation, for practical purposes.

3. An element cannot cover a significant cross-sectional area of two spent cores due both to the lattice spacing in the Spent Core Storage Pool and the fuel element length. In this particular case, cores are located on 2 ft-6 in. centers (infinite separation) leaving about 13 in. of water between two adjacent stored cores. Since the active

fuel length is only 20 in., it is clear that the reactivity interaction should be no greater than in case (2) above.

(b) Breaking of control blade locks

Answer:¹⁷ We believe this question is inapplicable to the HFIR-type core as stated in response to question 41(b)(1) above, since there are no control blade locks and since control rods are not removed with the fuel elements.

(c) Impairing heat transfer capability by laying the core on its side

Answer:¹⁷ This question was treated to some degree in Part 1, Section VI-C-2. However, more recent calculations pertaining to a fuel element with a 6 hr cooling time (the minimum for fuel handling) indicate that some melting would occur in times of the order of a few minutes if the fuel element fell on its side. While it is impossible to predict at this time the remedial actions that might ensue following such an accident, it is not unreasonable to expect that the element could be righted in a fairly short time and prior to onset of melting.

It is not possible to predict the disposition of the molten fuel should corrective action not be taken in time to prevent melting. This question will, however, be studied in more detail during the Title-II design stage and will be discussed in the FSAR.

(d) Consequences of fuel melting for cases a-1, a-2, and a-3 cited above.

Answer:¹⁷ If it is established by means of detailed calculations that the dropping of a newly spent core and any subsequent melting could lead to a criticality incident, the fuel handling procedures and/or fuel storage racks will be modified as required to prevent such an incident following a dropped element accident.

Using the design basis accident philosophy, submit your analyses of the above situations with associated dose calculations, where applicable.

Answer:¹⁷ Regarding the last part of this question, we believe that since a criticality in the Spent Core Storage Pool can be prevented by suitable means (see part (d) above), any radiation doses would be a result of the melting of a single fuel element only. There is no doubt in our minds, however, that any computed doses to the general public would be considerably lower than those computed for the MCA. The following reasons support this thesis:

1. The upset fuel element would be covered by about 30 ft of water. Since melting of the element, if it did occur, would probably proceed in a non-violent manner, the fission product retention in the pool should be very large; hence, the fission product release to the building would be very small.
2. Since the pool is designed to cool a 6-hr-old core, there is no apparent way that thermal energy could be added directly to the

containment building atmosphere as a result of a meltdown in the pool.

3. The gross fission product inventory from one spent fuel element with a 6-hr cooling time is only about 0.3%, and the iodine inventory about 20%, of the respective inventories of the full core at the onset of the MCA.

4. The reactor building is normally maintained at a pressure which is negative with respect to ambient. If, as a result of a meltdown, sufficient activity were released from the pool to cause automatic building isolation (and resultant evacuation), the building would be initially sealed at a negative pressure. It appears, therefore, that any overpressure in the containment building would be limited to the barometric load as a consequence of building isolation. The net conclusion, therefore, is that the very modest (0.5 psi max.) containment building overpressure results in very little outward leakage of fission products; the major dose contribution to non-AARR ANL personnel or the general public would be that due to stack release prior to isolation. This dose would be very low by comparison with calculated MCA doses (c.f., Section VI-E-2, Part 1).

5. If the exhaust air activity is not high enough to cause building isolation, it is clear that doses to the general public would be inconsequential.

6. The question of doses to operating personnel is more difficult to answer. Clearly, the doses would be less than would be expected as a result of the MCA, provided the MCA occurs as postulated (see Section VII, Part 1).

The above statements are intended to show that the melting of a single fuel element in the pool would present the AARR facility management with a severe cleanup problem. We believe that there is no question that the doses resulting from a fuel handling accident are significantly lower than those which might result from the AARR MCA.

225. *Questions 120 and 124 above - Consider the experimental facilities as a source in the discussions of the consequences of an MCA. (see Question 216 above.)*

Answer:¹³ See response to question 216 above.

226. *Question 128 above - What are the design margins against failure of these systems?*

Answer:¹⁵ The use of the words, "significant safety factor," in the answer to question 128 was unfortunate. One might infer that the fission-product barriers listed in question 128 are, in fact, designed to withstand a particular reactivity excursion. This is definitely not the case. The only listed fission-product barrier whose design criteria are even remotely related to credible (or even hypothetical) reactivity excursions is the reactor containment building (RCB). Even in this case, however, the postulated MCA (see Section VII-B, Part 1) which governs the design of the RCB has no relationship to the magnitude of a reactivity excursion, since the initiation of the

postulated MCA has never been related to a specific reactivity accident. Nevertheless, some comments can be made regarding a so-called reactivity excursion yield margin (REYM), defined here as the nuclear energy release calculated to be required to breach the weakest part of the primary system divided by the maximum credible nuclear energy release.

(a) Maximum credible nuclear energy yield

The maximum credible reactivity accident envisioned for the AARR is that resulting from massive voiding of the bulk fluid in the ITC. Analysis of this accident (presented in Part 1, Section VI-B-3), shows that even in this extreme case, the reactor core is protected from damage through safety-plate scram. This analysis, using the pessimistic assumption of 3.2% $\Delta k/k$ reactivity addition available, indicates a total fission energy release less than 20 MW-sec and maximum hot spot temperatures well below the melting point. It should be noted that, despite the fact that this reactivity insertion is called "credible," we cannot envision any mechanism capable of introducing this amount of reactivity by way of massive voiding of the ITC (c.f., Section VI-B-3, Part 1).

(b) Nuclear energy release to rupture primary system

Recent calculations by ANL have indicated that the weakest portion of the primary coolant system, in terms of the strain energy required to rupture any major component thereof, is the reactor vessel. This is consistent with the conclusion reached for the HFIR primary coolant system.^a

As the magnitude and/or rate of the reactivity insertion initiating a nuclear accident increase, so too do the nuclear energy yield and the fraction of the core participating in a metal-water reaction. It is shown in Part 1, Appendix E, Table E-2, that the reactor vessel can contain a nuclear excursion having a nuclear energy yield of 850 MW-sec and an accompanying metal-water reaction energy of 1500 MW-sec.^b

(c) Reactivity excursion yield margin (REYM)

From the above discussion, it is seen that the REYM for the AARR primary coolant system is 850/20, or 43. This number does not appear to us to have any great significance, although it is certainly indicative of a "margin" between what might be considered a credible situation and that situation which would result in breaching of the primary coolant system via the bursting of the reactor vessel.

227. *Question 132 above - If further analysis raises the possibility of a core meltdown greater than 5%, what would be the likelihood*

^aF. T. Binford, et al, "The High Flux Isotope Reactor Accident Analysis," ORNL-3573, April 1967, p. 232.

^bRecently, the reactor vessel material was changed from Inconel to stainless steel. The effect of this change on the vessel containment potential is not known at this time.

of a meltthrough? Discuss the potential for recriticality at the upper shield plug. Is there need for a core disperser or other anticriticality device?

Answer:¹⁷ (a) Likelihood of meltthrough

The "likelihood of a meltthrough" following a core meltdown cannot be determined or even estimated until models of the accident leading to the meltdown and the subsequent fuel movement are defined. The power level required, the fuel distribution, the presence or absence of water in the vessel, etc., all affect the answer to this question. However, it can be pointed out that the AARR system differs from power reactors (in this context) in two important ways. The first difference is that there is a two-foot thickness of metal between the bottom of the core and the interior of the building. The bottom head of the vessel is a flat, solid stainless steel forging approximately 12 in. thick, and the shield plug between this head and the core bottom is also approximately 12 in. thick. The second difference is that AARR is a pool reactor and the reactor vessel is always covered with water which serves as a heat sink. These factors lead us to conclude that the likelihood of a meltthrough is extremely low.

(b) Recriticality

A core reassembly on the top shield plug would constitute a geometry far from optimum (i.e., not spherical). On the basis of a preliminary review of experimental criticality data, we believe that such a reassembly would be grossly subcritical. We will look at this problem in more detail when the plug design is completed, and the results will be presented in the FSAR.

(c) Core disperser

Based on the comments in (b) above, there does not appear to be any need for a core disperser or other anticriticality device. If the final reactor design should indicate otherwise, such a device will be provided.

228. *Question 133 above - In the revised PSAR (Part 1), sufficient quantities of fission products escape from the stack to yield a dose of 2 rem to the thyroid at 1500 meters. At 100 meters from the containment building, this dose could be much higher, and fumigation conditions might increase this even further. Evaluate the contribution of a stack release to the 100 meter dose based on the release quantities already identified.*

Answer:¹⁶ A very conservative analysis of the inhalation doses due to stack release prior to isolation of the Reactor Containment Building is given in Section VII-C-2, Part 1. It is noted there that within the range from the RCB to a distance of 1500 meters, the calculated maximum incremental inhalation dose to the thyroid is 11 rem, "regardless of the choice of meteorological category." This would arise from Category-A meteorological conditions at ~300 meters. At 100 meters,

the dose would be smaller, and very small by comparison with the calculated 2-hour dose rate to the thyroid (see Table VII-C-1, Part 1).

229. *Question 136 (HFIR question No. 5) - Discuss the results of experiments on longitudinal movement of fuel elements mentioned in the answer to this question.*

Answer:¹⁵ Based on count-rate measurements, with all control rods fully inserted, it has been determined that the reactivity effect of moving one fuel element longitudinally with respect to the other is negligible.^a

230. *Question 136 (HFIR question No. 2F) - That part of the question regarding pressures that might be generated from local fuel melting has not been answered. What information is available on pressure surges vs. the amount of plate experiencing burnout or local melting? The answer to HFIR question 2F indicates there may be conditions under which the reactor cannot be scrammed. Discuss these conditions further.*

Answer:¹⁷ We believe that there is no technically acceptable answer to this question. The work done for the AEC by TRW Systems, Inc., on "Kinetic Studies of Heterogeneous Reactors" was intended to generate this particular information. However, the project termination presently precludes answering this question. We also do not believe that analytical models would be of any use without experimental verification.

That part of the reply to HFIR question 2F dealing with control rod scammability (or lack of scammability) has been considered in some detail in response to question 251 below.

231. *Question 114 (b) above - The control room ventilation system is described as an emergency load in Criteria 13.3 and 13.5. What is the reason for omitting this load from the listing of Class-I loads?*

Answer:¹³ Criterion 21.3, Section II, Part 1, defines emergency loads as those loads which can tolerate brief losses of power, and indicates that emergency power to those loads will be furnished by two emergency generators (Class-II power). Criterion 21.2, Section II, Part 1, states that uninterrupted (Class-I) power will be furnished to those vital loads (e.g., emergency cooling pony motors) which presumably cannot tolerate even momentary losses of power.

Criterion 13.5 is still considered valid, and it should be noted that the current design provides for control room pressurization on an emergency basis.

The emergency pressurization system is connected to Class-II (emergency)

^aPersonal communication, R. D. Cheverton, Oak Ridge National Laboratory.

power, which is consistent with the definition of emergency power given in Criterion 21.3. Therefore, this pressurization system is not contained in the listing of Class-I loads. Note, however, that it is contained in the listing of Class-II loads under question 114(b).

232. *Question 88(a) above - There is no mention in the list of Class-I items of the spent fuel and waste storage areas, or the engineered safeguards. This list should be reviewed and necessary additions considered.*

Answer:¹³ According to Criterion 1.4, Section II, Part 1, Class-I structures, systems and components are those required following an earthquake in order to ensure that the plant can be shut down, that decay heat can be removed, that containment will not be breached and that malfunctions cannot increase core reactivity. With this frame of reference, let us consider the specific items in question.

Spent Fuel Storage Area - All spent cores are located in the Spent Core Storage Pool which is entirely within the containment building. This pool is a massive concrete structure completely lined internally with stainless steel. The east and west walls and the floor are 6-ft thick and the south wall is 5-ft thick. The possibility of gross pool rupture following an earthquake is considered so remote as to be classed incredible. Nevertheless, it has been shown (Part 1, Section VI-C-3) that even if such a rupture were to occur, if the most recently stored core had been shut down for about 30 hr or more, no fuel melting would occur, since that core could be adequately cooled by natural circulation of air. For these reasons, the Spent Core Storage Pool is not considered a Class-I item.

Waste Storage Area - In accordance with the definition of Class-I systems given above, waste storage systems are not considered Class-I items. The penetrations through the containment building for those systems, however, are considered to be Class-I, and are so classified in Item 1 of the questioned listing.

Engineered Safeguards - As far as the AARR Project is concerned, there is only one engineered safeguard (i.e., a system required to function following a MCA), and that is the containment building with its concomitant isolating features. These are definitely considered Class-I and are so classified in Items 1 and 9 of the questioned listing.

233. *Questions 153 and 155 above - On the premise that missiles may penetrate the containment, and that primary system piping associated with the experimental facilities near the top of the reactor vessel pool may be vulnerable, what is the basis for concluding that the reactor pool does not need a protective cover?*

Answer:¹⁴ It is conceded, in the discussion under question 153, that missiles falling through the pool water might possibly damage small lines above the reactor vessel associated with the experimental rabbit systems. However, only the portion of each rabbit assembly extending a short distance above the reactor vessel cover, containing the actuating slide valve, is part of the primary coolant system during normal operation. This is shown in Figures IV-23 and IV-26 of Part 1.

Other small piping above the reactor vessel is connected to the primary system only during actual transfer of a rabbit sample (an operation taking only a few seconds). Ruptures in this latter piping would therefore not constitute primary system rupture except in the extremely unlikely event that the tornado-generated missile were to enter the pool and sever a line at the exact moment when a rabbit sample is in transit.

The severing of one of the rabbit assemblies at its point of connection with the reactor vessel cover (an unlikely event since these assemblies are heavy machined parts, and not merely lengths of pipe) would result in a small primary system breach on the reactor inlet side, within the Reactor Pool. The case of a total breach of a 24-in. reactor inlet pipe within the Reactor Pool, wherein the rate of system pressure loss would be far greater than in the case of the small break, is fully analyzed in Section VI-D-2b of Part 1. It is concluded that automatic scram would occur upon initial loss of pressure, and subsequently no core melting would occur because of continued coolant circulation by the primary pumps. Operation of the pumps by the pony motors (on Class-I or Class-II power, supplies for which are protected from damage under tornado conditions) would be sufficient to remove decay heat and prevent melting.

The design of AARR with regard to protection against tornado damage is based upon the criterion (see Criterion 1.6, Section II, Part 1) that the function of components required for safe, orderly plant shutdown and for removal of decay heat after shutdown will be unaffected by wind velocities up to 500 mph. Since it has been shown (see questions 153 and 155, and the above) that tornado-generated missiles entering the Reactor Pool could not cause damage that would result in inability to shut the reactor down and remove decay heat, it is concluded that a protective cover is not required on the Reactor Pool in order to satisfy Criterion 1.6.

234. *Question 156 - The codes referenced in the answer do not give the requested loadings. The answer submitted for this question should be reexamined.*

Answer:¹⁵ Loads that will be considered in the design of the piping systems will include the following:

1. dead loads.
 - a. weight of pipe and fittings.
 - b. weight of valves and other devices mounted on piping.
 - c. weight of insulation.
 - d. weight of fluid being transported.
2. internal pressure.
3. thermal expansion.

4. forces due to settlement of components and equipment.

5. seismic loads.

235. *Question 157 - The stress and strain criteria for all loading combinations have not been specified. If stresses, for example in case III A, are to go beyond yield, further definition of the criteria will be required.*

Answer:¹⁴ It is indicated, from the design accomplished to date, that it will not be necessary to permit the use of stresses in the plastic range. Case III A will, therefore, be amended to keep the stresses developed by the combination of loadings, including the loading of a double earthquake, within the elastic range.

236. *Question 158 - The seismic design criteria for the facility should be specified. It has been mentioned that the foundation design may be changed. Are the present facility and component design specifications sufficiently conservative to allow for potential changes in the magnitude of the design loadings which reflect foundation changes.*

Answer:¹⁶ Although our seismic consultant, Dr. Merit White, has not yet completed his design criteria report, we are now able to furnish values of design earthquake response spectra and values of damping factors which will be used in the design of containment and other Class-I items.

The design earthquake motion is similar to that recorded at El Centro, California during the 1940 Imperial Valley earthquake, except that the maximum ground acceleration is 0.075 g, rather than 0.33 g. Spectral curves for acceleration, velocity and displacement for the design earthquake are given in the following figures. The table lists values to be used for design damping factors.

Damping Factors for Design Earthquake

<u>Item</u>	<u>% of Critical Damping</u>
Reactor Containment Building and Foundation	5.0
Internal Concrete Structures	2.0
Steel Assemblies	
(a) Bolted or Riveted	2.5
(b) Welded	1.0
Class-I Piping Systems	0.5

In addition to the above material, Dr. White has furnished values of vertical and horizontal accelerations at various elevations throughout the reactor building which will be used in the design of Class-I components and piping. Design specifications for certain equipment such as

ACCELERATION SPECTRUM FOR AARR

VELOCITY SPECTRUM FOR AARR

DISPLACEMENT SPECTRUM FOR AARR

the reactor vessel have allowed additional conservatism to accommodate these design values now established.

237. *Question 1.7 - The only loads discussed are dead loads. Consider other loads, and describe the meaning of the term "appropriately combined." Provide a more complete description of the preliminary foundation design for structures containing Class-I systems and components which are located adjacent to the containment building.*

Answer:¹⁴ The following is a list of the various live loads, and their reduction factors, that are carried by the foundation mat of the containment structure:

(a) Containment Building experiment floor:

1. 4,000 lb/sq ft for the first 10 ft out from the face of the reactor biological shield.
2. 2,000 lb/sq ft for the distance between 10 ft and 30 ft out from the face of the reactor biological shield.
3. 1,500 lb/sq ft for the distance between 30 ft out from the reactor biological shield and the containment building wall.

These heavy live loadings allow for the movement and placement of experimenter equipment and shielding on and over the experiment floor. The long term live load of the experimenter shielding has been taken as a summation of weights totaling 300 KIPS per beam tube and with a c.g. 10 ft out from the face of the biological shield and on the beam tube centerline.

(b) Containment Building service floor area at elevation 715 ft - 200 lb/sq ft with a 100% reduction factor.

(c) Containment Building service floor area at elevation 728 ft -

1. Pool/Canal Demineralizer Cell - 400 lb/sq ft with a 75% reduction factor.
2. Coolant Piping Access Room - 400 lb/sq ft with a 75% reduction factor.
3. H. and V. Equipment Area - 160 lb/sq ft with a zero reduction factor.
4. All grating areas - 125 lb/sq ft with a zero reduction factor.

(d) Containment Building Instrument Balcony - 150 lb/sq ft with a zero reduction factor.

(e) Containment Building balconies and platforms at elevation 782 ft - 8 in. - 100 lb/sq ft with a zero reduction factor.

(f) Active Material Handling Building -

1. Pump and Operating Valve Room - 400 lb/sq ft with a 75% reduction factor.
2. Waste Resin Area - 400 lb/sq ft with a 75% reduction factor.
3. Elevation 780 ft - 6 in. - 1000 lb/sq ft with a 50% reduction factor.
4. Elevation 764 ft - between column rows L and K - 150 lb/sq ft with a zero reduction factor.
5. Elevation 764 ft - between column rows J and K - 100 lb/sq ft with a zero reduction factor.
6. Elevation 758 ft between column rows H and J - 100 lb/sq ft with a zero reduction factor.

The term "appropriately combined" alludes to the method of combining the various loading cases in a manner that will produce maximum moment or maximum shear at any particular section of the foundation mat.

With the exception of the Active Material Handling Building, all structures, including their foundations, are divorced from the Containment Building. The foundations of the structures housing Class-I systems are divided into two types which are:

- (a) isolated spread footings which are used for points of support adjacent to the containment building foundation mat; and
- (b) caissons for other points of support.

The isolated spread footings will be so proportioned as to minimize differential settlement. It is not known at this time what the sizes of the various isolated and caisson type footings will be. This can only be determined as the Title-II design progresses.

238. Question 180 - The pressure level of the hydrostatic test, and the basis for the selection of this level, should be specified now.

Answer:¹⁶ The exact value of pressure level of the periodic hydrostatic test for the reactor pressure vessel (and, of course, the primary system) has not yet been set. It will depend on several factors as follows:

- (a) The test pressure level must be greater than the operating pressure of 600 psig.
- (b) Consideration must be given to the instrument trip settings for overpressure relief.
- (c) The effect of frequent, cyclic pressure testing of the reactor pressure vessel on cumulative fatigue damage must be evaluated

to satisfy both Code requirements and additional ANL safety requirements.

The final choice of exact test pressure level cannot be determined at the present time due to the fact that the instrumentation system design has not yet progressed far enough, and, further, due to the fact that the reactor vessel material has been changed, requiring a complete redesign of the reactor vessel.

239. *Question 181 - Discuss the feasibility of 100% radiographic inspection of the pressure vessel.*

Answer:¹⁶ Aside from the question of radiation protection to the inspector, essentially all of the reactor vessel shell pressure boundary could be radiographed 100% with the vessel in place. The only surface for which 100% radiography would not be feasible would be the lower 47 in. diam. extension cylinder housing the drive rod shrouds. The pool seal and shield plug adapter become essentially a permanent part of this extension piece and the placement of either film or source would not be feasible in this area. It would be possible to ultrasonically examine this entire lower extension piece from the interior of the vessel. Other welds, such as the attachment weld of the core support ledge ring to the vessel shell, would probably require multiple film techniques and, although exceedingly difficult, radiography would be possible.

240. *Question 182 - This answer is incomplete. Discuss blowdown forces, Class-I criteria, etc.*

Answer:¹⁶ The detailed design of the reactor vessel supports and internals support structures will also consider stresses and forces resulting from primary coolant pipe breaks, design and maximum credible earthquakes and gamma heating. The preliminary design phase considers these effects to the extent necessary to assure provision of sufficient conservatism in order to accommodate these forces. The effects of a major break in the primary coolant pipes will include consideration of the break in a horizontal or vertical leg of the piping system. Effects of lateral loads are greatly reduced by the use of a vertical guide structure at the top of the vessel, and the design criteria for this lateral restraint will include all the above factors.

241. *Question 190 - Discuss the "conceivable cases" where loss of Class-II and Class-III power could occur simultaneously.*

Answer:¹⁴ Simultaneous loss of all Class-II and Class-III power can occur only after a series of low probability events, as follows:

1. Case A - AARR loses Class-II power and neither diesel will start. This situation is considered improbable, but if it should occur, the reactor would be scrammed in sufficient time to prevent any abnormal situation from arising in the core. Power from the Class-I (uninterrupted) electrical supply would carry all critical loads until such time as the core can self-cool (see Criterion 21.2, Section II, Part 1).

2. Case B - Only one emergency generator is available following a loss of Class-III power, and a bus fault occurs in the Class-II switchgear after Class-III power is reestablished. Until Class-III power returns, the single operating emergency generator carries the entire Class-II-load (see question 114 (b) above). When Class-III power is finally re-established, this generator will be paralleled with the utility power for a short period of time (less than one minute) and then disconnected from the system. If a bus fault should occur in the 480-volt emergency switchgear during the time that the generator is operating in parallel with utility power, it will result in a simultaneous loss of power to all of the Class-II system and the common parts of the Class-III system. Nevertheless, it should be noted that Class-III power would still be available to that portion of the facility not connected to the Class-II-switchgear. This includes, for example, the primary pumps.

242. *Question 191 - Discuss the possible mechanisms whereby the hydraulic shock absorber might fail. What are the weak points in this system? What would be the consequences of this accident?*

Answer:¹⁵ The hydraulic shock absorber, which operates during the last 6 inches of travel of the outer control rod drive engagement mechanism, could fail in two extreme cases.

1. A dirt (or crud) build-up in the shock absorber cylinder could keep the rod from completing its full scram.
2. The shock absorber cylinder could be dry. This is very unlikely, as the reactor would have to be drained below the dashpot level, before the scram signal was initiated, for this to occur.

In either case, the shock absorber assembly would receive an impact loading and the control-rod-to-extension-tube latch would receive shear loading. The forces and stresses incurred on the rod drive engagement mechanism during a malfunction of the dashpot are to be analyzed during Title II.

If, as a consequence of a dashpot failure, the control rod separates from the control rod extension tube, the mechanical structures are sufficiently strong to sustain a control rod drop from its full-out to full-in positions. Also, it should be noted that if an outer control rod should hang up during its last 6 inches of travel, the reactor would still be shut down, since Criterion 8.2, Section II, Part 1, requires the reactor to be shut down by at least 1% in reactivity if a single control rod is stuck in its full-out position.

243. *Question 192 - What are the analytical and experimental bases for concluding that the recently modified syphon breaker system will operate as designed?*

Answer:¹⁵ The provisions for breaking the primary system syphon are not finalized. Alternate methods are being considered. One of these alternates involves a 2"- or 3"-line between the suction and discharge lines near the vessel with an automatic, normally closed valve. Any break in the primary system main line will result in a pressure loss

which will open the valve and allow air flow to break the syphon. A preliminary analysis of this system indicates it is feasible. The choice of the system to be used will be made during the current Title-II design phase, and will be based upon analyses and/or experimental verification, as required. The Final Safety Analysis Report will describe the chosen system in detail, and will include a failure analysis to demonstrate reliability.

244. *Question 195 - Justify the location of the poison injection system. Why isn't it designed to feed directly into the pressure vessel? Why can't a small, full pressure line be designed to minimize the possibility of a rupture? In the event of a pipe rupture with the present system, what are the power requirements and methods of forcing poison into the core? Is more Class-I power required?*

Answer:¹⁴ The proposed method of introducing soluble poison into the reactor core was selected because it provides the simplest, most reliable system for the circumstance when introduction of soluble poison is most likely to be effective in controlling the reactivity of the core.

In order to be effective, the poison must be uniformly distributed in the primary water. This is best accomplished by introducing it into the flowing stream at a point where turbulence through several bends and fittings will provide thorough mixing. Since the system is not designed to be fast-acting, the delay time in reaching the core is not significant (see response to question 247 below).

The only advantage of adding poison solution directly to the vessel is to allow for the virtually incredible situation in which more than one of the control rods are stuck in their positions of minimum effectiveness, while simultaneously flow is completely interrupted, due either to a rupture of the reactor inlet line or to failure of both primary pumps. In this situation, cooling of the core is reduced to utilization of the water in the vessel with no forced cooling. It is very unlikely that under these circumstances, sufficient poison could be introduced into the vessel in time to prevent serious damage to the core.

A system for introducing the solution of neutron poison directly into the reactor vessel would require additional valves and piping, at least one penetration of the pressure vessel, and internal piping for distribution within the pressure vessel, thereby decreasing system reliability.

Power requirements for this system are not increased over those required for emergency core cooling. The emergency pressurizing pump and d.c. pony motors on the primary pumps are on Class-I power. Valves are operated by air accumulators, so that there is no additional emergency power requirement for operation of the backup shutdown system.

245. *Provide justification for the use of wake factors at low wind velocities. What are the doses at nearby facilities 15 minutes after an MCA?*

Answer:¹⁶ There is little experimental information to support or contradict the application of the building-wake factors at low wind velocities. One set of measurements has been reported by N. F. Islitzer, on the MTR-ETR complex of buildings. (See N. F. Islitzer, "Aerodynamic Effects of Large Reactor Complexes Upon Atmospheric Turbulence and Diffusion," IDO-12041, May 1965.) Since this is a loose complex of buildings rather than the more-compact complex of AARR buildings, definitive quantitative correlations cannot be inferred. Nevertheless, the data reported by Islitzer tends to support the argument that there are indeed substantial effects of building wake even at low wind velocities.

To a first approximation, doses to the thyroid after a total exposure time of 15 minutes after a MCA are one-eighth of the 2-hour doses, at dose points near the RCB. An improved approximation is that the 15-minute dose is one-third larger than the direct time proportionality indicates. In this calculation, the time required for the leakage plume to reach the dose point has been ignored.

246. *Containment Pneumatic Testing - It is a Code III requirement that the containment vessel be pneumatically tested at least at the design pressure. It is claimed that a test will be made at 125% of design pressure. The figure given under questions 168 (a) and (b) above, and in the April 6 letter, Crewe to Dunbar,* as 125% of design pressure is 6.8 psig. However, the design pressure specified in the PSAR is 7.5 psig. The MCA pressure is 5.45 psig, and it is seen that 125% of 5.45 is 6.8 psig. Thus, there is an important question as to the correct pressure at which the containment vessel is to be tested.*

Answer:¹³ The most recent calculation of the peak containment overpressure following a MCA yields a pressure of 5.45 psi, using the most pessimistic assumptions, and including the effect of burning the hydrogen at the peak of the pressure-time history. In accordance with Criterion 17.2, Section II, Part 1, the containment building is being designed to withstand a pressure of 1.1 times the calculated peak overpressure, or 6.0 psi. The design pressure, in accordance with the A.S.M.E. Boiler and Pressure Vessel Code, Section III - Nuclear Vessels, is 0.9 times 6.0, or 5.4 psi. The pressure at which the building will be tested is 1.25 times the design pressure, or 6.8 psi.

247. *Engineered Safeguards - The following systems are considered to be engineered safeguards for the AARR: core flooding system, secondary shutdown system, and the internal cleanup (refrigeration) system. Provide preliminary rather than conceptual design information on all these systems. Show that all engineered safeguards are Class-I design, will operate satisfactorily under MCA conditions, are testable, and have high reliability, availability and redundancy.*

Answer:¹³ As indicated in our reply to question 232 above, our understanding of an engineered safeguard is a system whose operation is

*Ed. Note: Letter transmitting Reference 8.

required following a MCA in order to mitigate the consequences thereof to the general public. None of the systems mentioned in the question are in this category. (In fact, the AARR does not have a core-flooding system.)

While we would hope and expect that the internal cleanup system and the refrigeration system would function following a MCA, none of the dose calculations in the PSAR take credit for these systems, and therefore we do not consider them as "engineered safeguards" in the reactor safety sense.

As far as the secondary shutdown system is concerned, that system is a slow-acting one and would have absolutely no bearing on the outcome of a MCA. It has been noted by the HFIR organization that "... we consider such a secondary shutdown system to be unnecessary in the sense that it is our opinion that the accidents in which the system might be of use are sufficiently improbable as to constitute no significant risk in operation."^a The testability, reliability, and availability are the same as that of HFIR since the system is the same as that of HFIR.

248. *Potential Accident Mechanism - Discuss the likelihood and consequences of a design basis-type accident which might be initiated by a steam explosion leading to asymmetric core displacement, shearing of several beam tubes, rupture of internal beam tube diaphragms, and uncovering of the core below the center line.*

Answer:¹⁷ We believe that this question cannot be answered in its entirety. For example, how does one assess the likelihood of an accident resulting from a steam explosion if the magnitude of the pressure pulse causing damage cannot be evaluated (see our reply to question 230 above)? In spite of the fact that this accident cannot be analyzed, we have discussed the fact that a beam tube rupture cannot result in uncovering the core (see our response to question 264 below). As a matter of interest, we do not see how a steam explosion of the magnitude implied in the question could "displace" the core without disassembling it, in which case, the question of "uncovering of the core" becomes academic.

249. *Internal Cleanup System - Dose Calculations*

(a) *In calculating whole body doses, it has been assumed that only 1% of the solid fission products are released. A 2% release has been assumed in HFIR calculations. Considering burnup, fuel size, enrichment, and maximum temperature involved in the MCA, discuss the basis for selection of the AARR solid fission product release fractions.*

^a"The High Flux Isotope Reactor Accident Analysis," F. T. Binford, et al, ORNL-3573, April 1967, p. 172.

Answer:¹⁶ It is noted that in the final safety analysis report for the HFIR the assumption was made that 2% of the solid fission products "escape from the immediate vicinity of the affected portion of the fuel." This assumption is stated on page 7 of ORNL-3573 ("The High Flux Isotope Reactor Accident Analysis"). It is also pointed out there that the percentage (2%) chosen for the solids is higher than the percentage usually recommended. In fact, the reference cited in the HFIR Accident Analysis (Parker, Trans. A.N.S., 6 (1), p. 120, June 1963) states, "The fact that no universally-accepted release values exist does not imply that this (fission-product-release) research has not been fruitful; in fact, the large mass of data, if consolidated almost without respect to its intended specific application, will substantially support the 100, 50 and 1% releases for noble gases, halogens, and solids used ... by the Division of License and Regulation..."

ANL has no basis for selecting a higher percentage than the one (1%) usually postulated for the fraction of solid fission products available for release from a reactor containment building. Nevertheless, it may be easily seen, from Table VII-C-1 of Part 1, that doubling the escape percentage of solid fission products from the reactor core would result in no worse than doubling the bone doses which are presently listed in Table VII-C-1. At the exclusion radius (1500 meters), the presently listed doses to the bone are: 2-hour dose of 2.5 rem; 30-day dose of 35 rem, in the absence of refrigeration; and a 30-day dose of 10.2 rem, with a half-capacity refrigeration system in operation.

AISITE-II calculations indicate that the 30-day direct doses from the leakage plume are increased slightly (~10%), and that the 30-day direct doses from the Reactor Containment Building (RCB) are increased by somewhat less than 50%, when the percentage of solid fission products released is increased from 1% to 2%. These AISITE-II calculations are for the idealized case of a steady leakage rate from the RCB of 1%/day.

Footnote (a) to Table VII-C-1 of Part 1 indicates that the direct doses from the RCB have been computed with the very conservative assumption of reactor operation at 100 MW for 90 days, which should provide enough conservatism to cover also the requested case where 2% of the solid fission products escape into the RCB.

(b) *All dose calculations to date have neglected the containment air cleanup system. What fission product reduction factor is claimed for this system?*

Answer:¹³ Since all dose calculations have neglected the containment air cleanup system, the fission product reduction factor claimed for this system is zero.

250. *Accident Propagation - Justify the assumption that a loss of coolant accident would result in no more than a 5% meltdown of the core. On what basis is it concluded that this meltdown could not propagate? Is experimental verification necessary? Are parallel channel burnout studies or thermal hydraulic tests necessary?*

Answer:¹⁴ We believe this question has been answered in response to question 214 (d) above.

251. *Control Rod Scrammability - Discuss control rod scrammability under the following situations:*

(a) *blowdown forces from a pipe break accident.*

Answer:¹⁶ We believe that this question has been answered in response to question 193 above.

(b) *flow blockage leading to partial meltdown and a metal-water reaction.*

Answer:¹⁶ Loss of control rod scrammability is considered credible only if a large fraction of the fuel zone of the core becomes molten. The outer control rods could then become jammed, due either to propagation of the melt through the outer fuel element side plate and inner control rod or to structural deformations produced by a steam explosion and/or rapid chemical energy release.

If one concedes that fuel melting would occur in a completely-blocked channel at full power, there is no identifiable mechanism whereby this melting could result in a net positive reactivity insertion. At worst, the inner control rod would be automatically withdrawn to compensate for the loss of reactivity occasioned by the moderator heatup and fuel plate expansion in the blocked channel(s). If this occurred, however, the outer control rods would certainly be scrammable, since damage to the outer control rod channels could not be effected without severely damaging the inner control rod channel, which is presumed to allow unhindered motion of the inner control rod.

The more likely results of a flow blockage would be either the self-shutdown of the reactor due to the inherent negative reactivity feedback effects (assuming no safety system action), or the automatic scramming of the reactor due to high gamma activity in the primary coolant. In the latter case, if scrammability had been affected by the fuel plate melting, the former shutdown mechanism would still act, and could be supplemented by the backup soluble poison injection system.

(c) *Can the core be scrambled in the event of a steam explosion?*

Answer:¹⁶ With respect to the question of control rod scrammability in the event of a steam explosion, it is noted that the side plate of the outer fuel element provides a strong barrier to control rod damage. The static pressure required to burst this side plate is ~400 psi. During a steam explosion, the dynamic pressure levels within the fuel zone, required to cause side plate bursting would have to be considerably greater than the static bursting pressure. Such an explosion would preferentially expel molten metal and water in the axial direction, thereby causing core disassembly and self-shutdown.

252. *Design Loads - Provide further information on the design loading combinations to be used for Class-I structures and components. The design codes do not answer this question.*

Answer:¹⁶ With respect to the containment building, Table 157-1 (question 157 above) covers the subject of load combinations with the available information. For Class-I items other than containment the design loading conditions will combine the effects of the seismic design loads with all other functional loads to assure that the component behaves elastically. For example, the design loadings for the reactor pressure vessel will include the effects of a primary pipe break, design earthquake loadings and all functional loads - pressure, thermal, pipe reactions, etc. These loads when combined, shall not result in any primary stresses exceeding the yield point of the material at the temperature involved.

It is recognized that design codes do not generally provide for all loading combinations to be employed in the design of critical Class-I structures and components. What the codes do provide, however, are allowable stress limits or values of minimum yield strength for the particular material under investigation. It has always been our intent to employ conservatism in the design of Class-I items including the containment building. Inasmuch as the MCA and earthquake are assumed to occur simultaneously, any Class-I structure or component which is affected by the MCA will be designed to accommodate both the MCA and earthquake effects.

253. *Load Combinations for Containment Structure - Evaluate the case where the containment is subjected to a 300 mph wind and a vacuum of 1/2 atmosphere, in a format similar to that in Table 157-2 (question 157 above). Show that the containment maintains functional integrity under these conditions. On what basis is it assumed that the reactor will be maintained in a safe shutdown condition under 500 mph tornado wind velocities?*

Answer:¹⁴ We do not understand the intent of this question since we are not guaranteeing the integrity of the above-grade portion of the containment structure under the influence of tornadic conditions. Question 153 above itemizes in greater detail the basis for the assumption that the reactor will be maintained in a safe shutdown condition in the event of a direct tornado strike on the AARR facility. (See also response to question 233 above.)

254. *Pressure Vessel Design - Provide the loading combinations to be used in design of the pressure vessel. Use the format of Table 157-2 (question 157 above). Since the pressure vessel design will go beyond current codes, provide assurances that the design approaches will not be in conflict with standards presently being written.*

Answer:¹⁶ The following table summarizes the various loading conditions for the reactor pressure vessel:

LOAD COMBINATIONS FOR REACTOR VESSEL

	<u>D.L.</u>	<u>P.L.</u>	<u>H.L.</u>	<u>T.L.</u>	<u>E.L.</u>	<u>MCA</u>	<u>T.P.L.</u>
Case I	X	X	X	X			
Case II	X	X	X	X	X	X	
Case III	X	X	X	X	2 E.L.	X	
Case IV	X						X
Case V	X	X	X	X	(Plus thermal/pressure transients)		

where D.L. - Effects of dead load

P.L. - Internal design pressure

H.L. - Hydraulic forces internal to vessel

T.L. - Effects of thermally induced loads and stresses

E.L. - Effects of earthquake loads

MCA - Effects of maximum credible accident (pipe rupture - jet forces on vessel, etc.)

T.P.L. - Initial hydrostatic test pressure load (1.25 x design pressure)

ASME Code allowable stress intensities will be permitted (except in the belt course and closure head bolts where ANL has added conservatism) for Cases I and V. For Case II, the primary stresses will be limited so that they will not exceed the yield strength of the material. For Case III, deformations and strain will be limited to ensure the structural and functional integrity of the reactor pressure vessel.

With respect to the statement concerning assurance that the AARR reactor vessel design will not be in conflict with standards presently being written, we assume this refers to the proposed RDT-S-918, - "AEC Reactor Coolant System Vessel Requirements." We have examined the 5th revision (dated 5 April 67) of this proposed standard which appears to apply to the more conventional water-cooled power reactors employing stainless-steel-clad ferritic-steel vessels. However, with the exception of specific references to ferritic vessels in this proposed standard, it appears that the AARR vessel design does not conflict - at least in principle - with this proposed draft standard.

255. *Core Support Structure - The joining of the support lugs to the reactor vessel should be reviewed for the manner in which the criteria for 100% radiography of the pressure vessel may be satisfied.*

Answer:¹⁶ Actually the core support structure consists not of lugs, but of a continuous annular ring fully welded to the inside surface of the vessel. The detail used is acceptable for full penetration welds as stated in Section III, ASME Code, Paragraph N-462.3(c).

256. *Pressure Vessel Head - The head is a region of maximum stress and radiation. Why shouldn't definite requirements be met for material in this region?*

Is there enough holding power in the bolts to assure that the head can't come off the vessel? Show that this incident cannot occur.

Answer:¹⁶ The top head of the reactor vessel is subjected to high stresses, particularly due to the presence of many vertical penetrations. In general, however, the top head is not subjected to high levels of radiation (in fact, the height of water separating the quick-opening breech plug and the core is, in part, set to afford sufficient shielding to ensure long life of the rubber O-rings used to seal the breech plug).

We do not understand the question regarding material requirements. Prior to the change to a stainless steel vessel, ANL had placed very stringent requirements on the top head material, particularly with respect to low temperature impact properties. In fact, ANL went further than code requirements in trying to obtain good impact properties throughout the entire top head forging thickness.

As to the subject of bolt design, it is felt that considerable conservatism was used in the design of the bolting for the Inconel vessel. ANL had limited the primary stresses in the bolts to 20,000 psi for material which has a minimum yield strength of about 85,000 psi at room temperature, and only slightly less at operating temperature. Thus, the bolts would have a safety factor, against gross yielding, of slightly greater than four, while the vessel shell would yield on a gross basis at a pressure 1.5 times the design pressure. It is felt that the bolting arrangement provided for the Inconel vessel would have resulted in vessel failure prior to bolt failure.

257. *Accident Analysis - Provide an analysis of core cooling capability and core damage vs pipe break size. Is the largest possible break the worst case?*

Answer:¹⁴ We believe this question has been answered in question 196 above.

258. *Experiment Cooling System - From the viewpoint of reactor safety (in terms of reducing the probability of rupture of the primary coolant system), discuss the desirability of providing a separate cooling system for experiment cooling.*

Answer:¹⁵ We assume that "experiment cooling" refers to the vertical, encapsulated experiments, i.e., the hydraulic rabbits and static

irradiation facilities. Separate cooling systems for these facilities were considered. One of the main reasons for using primary coolant to cool the experimental facilities' structure and samples is the inherent reliability and simplicity of the system.

With the present scheme, loss of primary water through failure of the water-cooled irradiation facilities is precluded by two factors:

(a) Appropriate valving in the hydraulic rabbits prevents primary water from leaving the reactor vessel (the coolant flow sleeve valve assembly, Fig. IV-30, 12/9/66 Revision of PSAR, is considered an extension of the reactor vessel) at all times except during sample transport. In the case of the static irradiation facilities, they are completely within the reactor vessel, although instrumentation leads may be brought out through vessel head penetrations.

(b) If a rabbit tube or static facility sample basket should rupture, there is no free volume available into which primary water could flow, since all available space is taken up by either primary coolant or samples.

259. *Core Design Changes - If the core design is changed to reduce costs, will these changes be proof-tested via a concurrent thermal and hydraulic test program?*

Answer:¹³ We have no plans at this time to change the core design for any reason.

F. Questions from Reference 18.^{*}

260. *Question 216 above - Although RDT has accepted Criterion 5.2 of the PSAR as applying suitable limits to the experimental program and to the energy potential of experiments, DRL has not. Argonne should be aware that RDT and DRL are conducting independent reviews, as required, and that there may be differences of opinion as to the adequacy of various matters. Further, we are informally advised that a DRL staff member specifically discussed Criterion 5.2 with AARR project management at a recent meeting, and that DRL's information requirements were apparently understood at that time. Thus, there seems to be no valid reason for confusion at Argonne over this question being restated. Please answer the question as it appeared in reference 12.*

Answer: Argonne acknowledges that it clearly understands the question posed by the DRL staff. However, we would again like to point out that the answer desired cannot be given and we believe the information desired does very little, if anything, to define potential hazards. The information desired by the DRL staff is a specific table of limits - quantities of all dangerous isotopes such as plutonium - that would apply to all experimental uses in connection with the reactor. As has been pointed out before, this is not only not feasible, but such limits or lists when used in lieu of experiment review can introduce real safety problems.

The safety of using one gram of plutonium as oxide dispersed in a stainless steel matrix versus one gram of plutonium nitrate in solution, or the safety of a material in the Internal Thermal Column versus its use as a target in a beam tube are so widely different that only individual review and analysis is practical. As far as we know this is the procedure followed at essentially all test and research reactors such as the MTR, ETR, TREAT, EBR-II, etc.

In summary, we do not believe there is confusion at Argonne as stated, but nevertheless we cannot answer the question as stated in Reference 12. We believe that the only practical solution is as stated in Criterion 5.2 of Part 1.

261. *Question 221 above - This answer does not discuss the design margin on which the trip level for reactor scram at high neutron flux/primary coolant flow ratio has been set. Further, this trip level should not be considered only as a function of the core, but should be based on the reactor system as well. Therefore, please answer this question based on AARR, not just the HFIR.*

Answer: The setting of the high flux-to-flow ratio trip is tentative, and is the same as the ORNL value of 1.3 for HFIR. The minimum steady-state power margin, at the trip level, for full reactor coolant flow, has previously been stated in response to question 105 above.

^{*}Ed. Note: These questions were raised by RDT after their review of ANL responses in Reference 13. Answers were furnished in Reference 19.

For reactor transients, it is not considered meaningful to apply the concept of an overall design margin. Rather, all relevant portions of the reactor system, including trip level settings and individual safety channel response times, will be analyzed for anticipated, but inadvertent transients, to ensure that the plant design will conform to the applicable safety criteria, particularly Nos. 4.4 and 4.8. It should be noted that this is a ratio trip and therefore reacts to both upward changes in power and downward changes in flow.

262. *Question 225 above - See question 260 above.*

Answer: See reply to question 260 above.

263. *Question 232 above - Class-I structures, systems and components are those required to ensure plant shutdown, decay heat removal, and containment integrity not only under earthquake conditions, but also under other extreme environmental conditions such as hurricanes, flooding, tornadoes, etc. Further, the integrity of other structures in addition to the containment vessel, but also containing large amounts of fission products, must also be maintained. With this frame of reference, please justify why the spent core storage area is not listed as a Class-I item, considering that a spent core that had been shut down for less than 30 hours presumably cannot be adequately cooled by air, and could melt (at least partially) if the storage pool ruptures.*

In addition, the waste storage area may contain significant amounts of radioactive materials from various sources. Show that this area should also not be considered as Class-I, based on the occurrence of only acceptable activity releases following loss of integrity under any of the environmental conditions noted above.

Answer: There are two canals used primarily for fuel handling and storage and we believe we may have confused the issue with nomenclature. One canal is inside the containment building and is actually a structure in common with the Reactor Pool. This pool is a Class-I structure and it is here that spent cores are stored for approximately six to eight months. The other canal is outside the containment building and is not in a Class-I structure. This outer canal is used for preparing cooled cores for shipment, for loading coffins, etc., and since only cooled cores are handled in this canal it is not a Class-I structure.

The waste tanks are not actually waste storage facilities in the usual sense of the word but rather collection tanks. In normal service they collect laboratory wastes, decontamination wastes from coffins, ion exchange rinses, etc., which are collected and transported in unshielded tank trucks to the ANL waste handling facilities. In the event of a serious core meltdown hot wastes and hot resins would be collected in these facilities for packaging and shipment. Since the residence of such hot wastes will be of short duration, very low frequency - hopefully never - and suitably cooled, we do not believe these need to be Class-I structures.

264. Question 247 above - Please be advised that the Division of Reactor Licensing considers the secondary shutdown system, internal cleanup system and refrigeration system of sufficient importance to be designated engineered safeguards. The basis for this position is described below. In addition, it should be noted that engineered safeguards may be considered to include those systems necessary to prevent the occurrence of accidents, as well as those required to mitigate their consequences.

As for the core flooding system, we had not previously been advised of your decision not to include this system in the AARR. Considering the accident mechanism postulated in Reference 12 (asymmetric core displacement, rupturing of beam tubes, and uncovering of the core below the center line), show that a core flooding system is not a necessary safeguard.

The original dose calculations following an MCA showed that the 15 minute doses at the CP-5 facility could be as high as 3000-4000 rem to the thyroid. A later calculational model, employing wake factors, reduced this figure to less than 300 rem. If it should be determined that the use of wake factors is not justified, it will be necessary to show that the internal cleanup and refrigeration systems are not absolutely necessary safeguards to reduce doses to acceptable levels.

As for the secondary shutdown system, we realize it is slow acting and would have little bearing on the outcome of an MCA. However, it is conceivable that this system might be necessary under the conditions of a hypothetical accident that could result in power oscillations, possibly leading to containment pressures higher than that in the MCA. It appears that the secondary shutdown system might be the only method of shutting the plant down under these conditions. Please comment on this accident, and the importance of the backup shutdown system.

For all the engineered safeguards discussed above, if it is determined that a system should be required considering the conditions and situations outlined, please provide the information on reliability, availability, construction, etc., as requested in the original question.

Answer: The AARR has never had a core flooding system, separate from the primary coolant system, that would permit the adding of water to the reactor vessel in the event of a primary system rupture.

A beam tube (aluminum internal sleeve) rupture would not result in the uncovering of the core, since the secondary high pressure seal, located at the end of the beam tube adapter which is within the biological shield, is designed to withstand full system pressure. There would be a momentary pressure reduction until primary water filled the volume within the ruptured beam tube. Thereafter, the pressurizing system would make up the water lost to the beam tube volume and maintain system pressure. The core would not have been uncovered. Hence a core flooding system would not be necessary for

this situation. The same conclusions could be drawn if any number of beam tubes were to rupture simultaneously, since the total volume of all beam tubes is small compared to the primary system volume.

If the rupture is a complete severance of the outer sleeve and inner sleeve, the rupture would be inside the pool; hence pool water would be available to flow into the reactor vessel. A separate system would seem to be necessary only if one were to postulate the accident proposed in Reference 12, occurring simultaneously with the failure of all three pressurizing pumps and the complete failure of the canal. We believe this combination of failures is less realistic than the MCA.

In the earlier PSAR, for the heavily loaded SS-UO₂ system, a 2-hour inhalation dose of 1661 rem to the thyroid was tabulated (Table H-IX). This was calculated for the case of a leak rate of 0.1%/day from the RCB.

In Part 1, Section VII, the leak rate for the first 2 hours is assumed to be roughly 0.7%/day, on the basis of the pressure calculated for the MCA, without refrigeration or internal cleanup. Also, in this case, a plume-meander factor of 0.6 was applied, and a building-wake factor of approximately 0.05. Applying the product of these three factors

$$\frac{0.7\%/day}{0.1\%/day} \times 0.6 \times 0.05 = 0.21$$

to the 2-hour dose quoted above (1661 rem), the result is ~350 rem, which is in good agreement with the value (334 rem) tabulated in Part 1 as the 2-hour dose at 100 meters. (See Table VII-C-1.) Thus the earlier data are consistent with the data in the revised PSAR, when the revised model is accounted for properly.

We can find no mention of 15-min doses at CP-5 in the range of 3000-4000 rem to the thyroid, in the original dose calculations. If the revised assumption for the leak rate from the RCB is applied for the case of the first two hours, namely, 0.7%/day, and if the plume-meander factor of 0.6 is applied but not the building-wake factor, the result would be a 2-hour dose of

$$\frac{0.7\%/day}{0.1\%/day} \times 0.6 \times 1661 \text{ rem or approximately } 7000 \text{ rem}$$

In the first 15 min, therefore, the total inhalation dose to the thyroid would be approximately

$$\frac{1/4 \text{ hr}}{2 \text{ hr}} \times 1-1/3 \times 7000 \text{ rem or roughly } 1200 \text{ rem}$$

During the first 15 min following a MCA, the MCA-postulated leakage rate from the RCB would not be significantly affected by the refrigeration system. The recirculation system of air filtration

involves a maximum cleanup rate of ~ 4000 cfm, by comparison with a RCB air volume of roughly one million cubic feet. Therefore, such an internal cleanup system would have very little effect on the radiation doses to the public in the first 15 min following the postulated MCA. In summary, neither the refrigeration system nor the internal cleanup system would appreciably reduce the 15-min dose at 100 meters following the postulated MCA and therefore we do not believe that they should be classed as engineered safeguards.

At the June 30, 1967, meeting at ANL, ANL was informed that the "hypothetical accident that could result in power oscillations" refers to the discussion given in Part 1 for the case of an inlet pipe rupture inside the Reactor Pool and no reactor scram (Section VIII-C). Under these conditions, the backup shutdown system might be at least partially effective in shutting down the reactor, although the poison solution would be diluted by mixing with the pool water prior to entering the reactor vessel (see also response to question 268 below). However, it was noted in the Introduction to Section VIII that any accidents listed as hypothetical accidents are considered to be incredible. In this sense, it is ANL's understanding that systems which might tend to prevent a hypothetical accident, or mitigate the consequences thereof, are not considered essential to the safe operation of the facility and therefore such systems are not engineered safeguards unless they also prevent or mitigate the results of the MCA or design basis accidents.

265. *Question 249(b) above - We fully realize that all dose calculations have not taken credit for the containment air cleanup system. However, information on the fission product reduction capability of this system is of importance in connection with the evaluation not only of the MCA, as indicated in question 247 above, but also with the evaluation of the less serious, but more probable type of accidents. It is therefore requested that the question be answered as stated.*

Answer: The recirculating-filter system consists of two units, each unit consisting of three filters in series. The filter string is a high efficiency filter followed by an activated charcoal filter which in turn is followed by a second high efficiency filter. The capability of the individual units will probably be 99.9 plus % for 0.3μ DOP for each high efficiency filter and 30 to 50% removal of methyl iodide and 90% for other forms of iodine for the charcoal traps. The actual overall efficiency is of course a function of the source material. However, the question states that "the question be answered as stated" - and the answer to that question is as originally stated - no credit is taken in the dose calculations for any cleanup that might occur due to operation of the recirculating filter system.

266. *Question 259 above - Although you have no plans at this time to change the core design, would you conduct thermal and hydraulic proof tests in the future if, for any reason, core design changes should be considered desirable?*

Answer: Under the present definitions as we understand them, "proof test" means test to failure and the answer to this question is no. Any significant change in core design would require an addendum to the PSAR (or FSAR), and the question of what tests might be appropriate would be examined at that time.

G. Questions from Reference (20).*

267. *Question 237 above - Provide the mathematical basis by which the various loads will be "appropriately combined" to produce maximum moment or shear strength. Provide further information on the preliminary foundation design for structures containing Class-I systems and components. How will the isolated spread footings be designed so as to minimize differential settlement?*

Answer: The basic equation for obtaining the soil pressure beneath the containment building foundation mat that will be used in the design of that foundation is $P/A \pm Mc/I = f_{\text{soil}}$. The various loading cases will be combined in such a manner as to produce a maximum moment or a maximum shear condition at any particular section of the foundation mat. The first term of the above equation will always include the Dead Load of the structure and may, as the case might be, include the following loads:

- (a) Dead Load variable (experimenter shielding and water in the Reactor Pool and various canal portions)
- (b) Live Load
- (c) Pressure Load
- (d) Snow Load
- (e) Earthquake Load

The second term will include the eccentric Dead Load moment and may, as the case might be, include the eccentric moment caused by the following:

- (a) Dead Load variable
- (b) Live Load
- (c) Snow Load
- (d) Earthquake Load

In addition to these loadings the second term of the basic equation may, as the case might be, include the moment due to Wind Load or Earthquake Load as applied laterally to the containment building.

The design of the foundations of the Electrical Equipment Building and Laboratory and Office Building are not yet completed. The preliminary calculations of the containment building foundation mat are available. It is not clear from the statement, "provide further information," as to exactly what is desired.

*Ed. Note: These questions were raised by RDT after review of ANL responses in References 14-16. Answers were furnished in Reference 19.

The isolated spread footings will be designed to deflect vertically in as close a relationship as possible to the predicted settlements of the containment building foundation at the closest adjacent point of the foundation mat. This will be accomplished by the varying of the isolated footing area. The area of the footing will be selected by use of the Boussinesq analysis and/or the Newmark chart which involve the "bulb of pressure approach" to predicting settlement.

268. *Question 244 above - Please comment on the current location of the poison injection system in connection with its effectiveness in the event of a pipe rupture accident.*

Answer: If the rupture is in the inlet pipe to the reactor the poison solution may or may not find its way into the reactor vessel. If the rupture is in the pool, some poison might get into the vessel, except the solution would be diluted with pool water. If the break is outside the pool, no poison could get into the vessel.

If the break is in the outlet pipe, poison solution could be added to the vessel. If the pipe break occurs between the poison storage tank and the injection or pressurizing pumps no poison will be injected.

269. *Question 250 above - Provide further discussion on the possibility of meltdown propagation following a loss of coolant accident. Although it is claimed that the maximum credible accident would not lead to unacceptable consequences to the public, this does not mean that lesser accidents are to be considered much less cause for concern. Plant integrity (damage) is also a consideration.*

Answer: This question has been treated on three previous occasions. References to the subject material, ordered chronologically are:

(1) Part 1, Section VI-D-2b, (2) Questions 193 (a) and 196 above, and (3) Reference 10.

In this material, it has been stated that, so long as the core zone of the reactor remains flooded, the "most likely" consequences of primary system rupture are those concomitant with melting limited to the core hot-spots. The rationale leading to this position is developed in the above referenced section of the PSAR, and insofar as is presently possible, supported there by reference to experimental data.

Each time the subject has been reviewed it has been conceded that specific assumptions made regarding the complex events expected to occur during a major loss-of-coolant accident cannot presently be decisively defended. Indeed, it presently appears to be impossible to develop a defensible "middle-ground," between what the PSAR states as the most likely consequences of the subject accident and the pessimistic model used for MCA calculations (Part 1, Section VII, complete core meltdown, 100% Al-H₂O reaction, etc.). Major obstacles, in this respect are: (1) lack of appropriate experimental data; (2) lack of "tested" analytical models and associated computer codes;

and (3) lack of a detailed plant design at this stage of a project.

Resolution of only item (3) of the foregoing is considered to be within the scope of the AARR project. Presumably items (1) and (2) are receiving extensive attention within the LOFT program and its peripheral activities. It is understood that the LOFT program schedule is such that initial loss-of-coolant tests will begin early next year. We are also aware of the efforts being expended to develop computer codes, comprising mathematical models intended to describe, and form bases for extrapolation from, LOFT test results. We will continue to monitor the progress of the LOFT experiments and analytical programs and when the AARR design becomes definitive, use the best available data and computational tools to perform more detailed loss-of-coolant studies.

In summary, we are in full agreement with the RDT statement that the consequences of accidents, less severe than that assumed in the MCA model for the AARR, also cause considerable concern. However, recognizing the hazards that might result from making design decisions based upon analyses employing questionable assumptions, a pessimistic approach appears to be the only one reasonable at this time. Refinements in analysis, which tend to yield more optimistic results, must be relegated to a secondary role, and such calculations performed only when the output of the considerable effort involved can be defended adequately. With regard to the final RDT statement above, there is no known credible mechanism whereby plant (assume this means containment) integrity will be violated, due to a loss-of-coolant accident. Other plant damage, in addition to the initiating event, may be expected, but is considered inconsequential in case of a major accident.

270. Question 253 above - It is difficult to see why maintenance of the functional integrity of the reactor containment building is not necessary for safe shutdown of the reactor following a tornado strike. Show that safe shutdown could be accomplished if the containment fails (collapses), or that failure would be limited only to cracking, deformation, etc.

Answer: A detailed discussion of this question is given in response to question 153 above. All critical components required for shutdown are located either below grade, or within pools or structures located under many feet of water. The entire primary system with the exception of the inlet and outlet lines to the reactor and the pressure vessel itself are located in the below grade service floor and are protected by the six-foot thick concrete experimental floor. The control rod drives, the reactor control room, and all critical power supplies and cabling are also located below grade in Class-I structures. The reactor vessel and connecting piping is inside the canal structure and covered by a minimum of seventeen feet of water.

271. Question 218 above - The information provided about the rabbit system containment penetrations satisfactorily answers the question. Please provide similar information regarding the static irradiator facilities.

Answer: Static irradiation facilities do not penetrate containment; hence, the question does not appear to be pertinent to those facilities.

272. *Question 219 above - Provide a brief discussion on the control rod modifications that may be required to meet seismic design criteria. (See question 281 below.)*

Answer: It is not known at this time what specific design changes, if any, may be required as a result of the seismic design criteria. This information will be developed as the Title-II design progresses and as various stress and dynamic analyses are completed.

273. *Question 223 above - Please answer the questions on core behavior disregarding the reference to question 107.*

Answer: No analyses have been made considering the response of the core to pulse changes in reactor flow. No mechanism for the production of significant flow pulsations, accompanying gross reactor flow transients, can be envisioned for the completely water-filled primary coolant loop of the AARR (note this is a cold "hard water" system with no heated pressurizers or surge tanks and no pneumatic surge suppressors).

If only one, or perhaps a few, core coolant channels were blocked, the event would be undetectable presuming no fuel melting occurred. The occurrence of significant melting would be detected by the high activity level in the primary coolant. If a large number of channels were blocked, the resulting flow and/or power changes would indicate the incident.

An analysis of the maximum conceivable consequences of propagation of local core flow blockage is presented in Section VI-D-1c of Part 1. Although meltdown propagation is assumed in that analysis, it is by no means considered probable. It is considered likely that only fuel plates experiencing loss of flow on both surfaces will melt. This is based upon the tendency of pressure effects to result in a lateral plate deflection in the direction of a blocked channel, thereby improving the heat transfer status of its neighboring, unblocked channel. As a consequence, the likelihood of propagation is not considered to increase with the number of channels assumed to be blocked initially.

274. *Question 226, Section b above - Describe the basis on which it has been concluded that the weakest portion of the primary coolant system, in terms of the strain energy required to rupture any major component thereof, is the reactor vessel.*

Answer: The calculations and results pertaining to the strain energy required to burst the primary system (excluding the reactor vessel) is reported in the ANL Reactor Development Program Progress Report, April 1967, ANL-7329, p. 75. Note that the reactor vessel material change noted in footnote (b) to the reply to question 226, may invalidate the conclusions given.

275. *Question 234 above - Discuss the manner in which the referenced loadings are combined.*

Answer: Title-II design will include seismic analysis of the primary coolant piping system. The specific manner in which the seismic effect will be combined with the other loads has not yet been determined. However, the criterion will be the retention of the containment capability of the primary system and no violation of building containment due to pipe reactions.

276. *Question 258 above - Discuss the possibility of depressurization (blowdown) following a rupture in the hydraulic rabbit system. How many separate lines penetrate the pressure vessel from this system?*

Answer: The hydraulic rabbit system transfer station is within the containment building. The line leading from the reactor vessel to the transfer station has an inside diameter of 5/8 in. A complete rupture of this line, coincident with sample transfer, cannot result in a reactor depressurization, since the pressurizing system can make up leaks from a 3/4 in. diameter opening (see Section VI-D-2b, Part 1). It should be emphasized, however, that a rupture of this line is of no consequence, except during the momentary period when a sample is being transferred to or from the reactor vessel. There are planned, initially, four hydraulic rabbits in the reflector and five in the ITC for a total of nine. Each has a separate penetration in the vessel.

277. *Question 217 above - This answer provides an adequate discussion of doses from plutonium produced in the core. Please consider also any plutonium that may be inserted into the irradiation facilities, etc.*

Answer: See response to question 260 above.

278. *Question 251(a) above - It appears from the answer that blowdown might occur before reactor scram is initiated. Please clarify this situation.*

Answer: It is considered to be inconsequential whether control-rod de-latching occurs prior to "blowdown" (depressurization) of the AARR primary coolant loop. This is based upon the short time required to depressurize the completely water-filled, low temperature system comprising the AARR primary coolant loop. The loss of only approximately 60 gallons of water is required to cause complete pressure loss and negligible steam formation is expected within the system during blowdown.

279. *Question 251(b) above - Regarding the statement that "there is no identifiable mechanism whereby this melting could result in a net positive reactivity insertion," ORNL has reported in the HFIR accident analysis that the removal of 30% of the fuel results in a positive reactivity insertion. Please comment on this apparent discrepancy. What is the pressure that would be required to cause side plate bursting?*

Answer: The statement referred to is correct inasmuch as it is made in the context of describing reactivity changes which might result from local flow blockage. According to the ORNL analysis (pages 260-261, ORNL-3573), if local melting occurs and propagates, in a highly idealized manner such that a spherical cavity ~ 6 cm in diameter is formed, with the removed fuel completely replaced by water, a maximum reactivity gain of $0.03\% \Delta k/k$ (0.4 cent) could result. This gain of reactivity is considered negligible and would be compensated by competing inherent effects, moderator density change, etc. As the size of the assumed cavity grows beyond 6 cm, a reactivity loss is incurred.

Although it is true that ORNL has calculated a substantial reactivity gain ($\sim 2.5\% \Delta k/k$) for removal of as much as 30% of the fuel volume, the calculation is based upon removal uniformly throughout the core and replacement of removed fuel with water. We are unable to conceive of any accident situation that could lead to this condition.

The static bursting pressure of the outer side plate is ~ 400 psi, as stated in the answer provided. No calculation of dynamic pressure levels necessary to burst the side plate have been made but they will be in excess of 400 psi value.

280. *Question 228 above - Consider the effect of fumigation conditions on the calculated doses.*

Answer: To permit evaluation of effects of meteorological conditions which presumably might result in larger doses close to the RCB than have been inferred from Category-A conditions, a detailed quantitative definition of "fumigation conditions" is required.

281. *Question 236 above - DRL's seismic consultant reports that the minimum ground acceleration should be 0.1 g. Please discuss the basis on which the maximum design quake has been set at 0.075 g. What design changes would be required to accommodate 0.1 g acceleration?*

Answer: The basis for 0.075 g is the following:

Earthquake Exposure

Comparison of the seismic histories of (a) California - Western Nevada (200,000 square miles), and (b) the North-Central United States (Ill., Ohio, Mo., Mich., Ind., Wis., Ky., Western Tenn., Iowa, Ark. - 488,000 square miles) reveals that in terms of total number of earthquake shocks per unit area per year over a period of 30-50 years the activity in (a) is about 50 times that in (b). In terms of activity at magnitudes of earthquake greater than $M = 3$ (about the lower limit of significant damaging earthquakes) the difference of activity is far greater.

As a conservative basis for establishing a design standard for the Argonne region the exposure there is assumed to be 1/50 the average for the California-Western Nevada area as determined from the seismic history of that area. This assumption is considered conservative for two reasons, (a) the fact mentioned earlier that for shock magnitudes

>3 the observed frequency in the North-Central region is far less than would be predicted on this basis, and (b) that included in the North-Central region is an area - Western Tennessee - of relatively high seismic activity and which is relatively distant from Argonne.

On this basis, the following expression for the probability of earthquake occurrence within the North-Central region has been obtained:

$$N = 10^{-6}(0.0054 x^3 - 0.000665 x^5 + 0.0000725 x^7),$$

where N = expected number of earthquakes of magnitude $>M$ per year per square mile and $x = 8.7 - M$. It is assumed that M has an upper limit of 8.7.

Design Standard and Survival Probability

A structure has a pattern of resistance to vertical and horizontal loads. In case a particular earthquake motion had furnished the design loads it would be called the design earthquake and would be described by its spectrum of response (dependence of the response of a simple elastic system to this particular motion on the period and internal damping of the system).

However, the same response spectrum, or portions of it, would be produced by earthquakes of various magnitudes at appropriate distances from the structure being designed. Consequently, for a particular design the probability of survival is the probability that none of possible damaging earthquakes occur within their critical distances from it. If the design spectrum is described by the so-called "undamped spectral intensity" (its average value over a certain range of period times that range) with symbol SI_0 the following relation between magnitude M and distance r can be employed.

$$r^2 = \frac{h_o^2}{SI_0} \left(10^{-4.44} + 1.07M - .027M^2 - \frac{h_o^2}{h_o^2} \right),$$

where h is the depth to region of slipping and for simplicity is taken to be 15 miles, while $h_o = 15$ miles (Ref: Hudson and Housner, "Analysis of strong motion accelerometer data from the San Francisco earthquake of March 22, 1957," Bull. Seism. Soc. Am., 47, 1957).

As design earthquake it is proposed to adopt a shock similar in time history to that recorded at El Centro on 18 May 1940 but of 0.225 its strength. The proposed design earthquake has a maximum acceleration of 0.075 g (compared to 0.33 g at El Centro) and an SI_0 of 1.9. Then the preceding equation becomes

$$r^2 = 120 (10^{-4.44} + 1.07M - .027M^2 - 1).$$

The probability of survival of a structure designed for a certain SI_0 can be shown to be

$$p_s = e^{-\pi T \int r^2(M) n(M) dM},$$

where $n(M) = dN/dM$, and T is the period of time under consideration. In the present case this becomes

$$p_s = e^{-.00181T}.$$

The probability of not surviving is $p_f = 1 - p_s$. For the Argonne area some typical values are given below.

T (years)	1	20	50	100
p_s	.998	.965	.914	.835
p_f	.0018	.035	.086	.165

We have been notified by AEC-CH that the second question under this item should be 0.2 g for the design earthquake rather than the 0.1 g of the question. It is not possible to determine without extensive analysis what design changes would be required if the design earthquake were changed to one with a 0.2 g ground acceleration. The static and dynamic analysis of the containment building, its foundations, and all Class-I structures and equipment components would have to be analyzed for the new conditions and design changes determined on that basis.

We would like to again point out that the plant design is not basically to survive the design earthquake but that safe shutdown and/or containment capability will survive twice the design earthquake occurring simultaneously with the MCA.

282. *Question 239 above - If the lower 47 inch diameter extension cylinder housing the drive rod shrouds cannot be radiographed in place, would it be feasible to do this at the time of manufacture?*

Answer: The lower 47 inch I.D. extension cylinder shell could be radiographed at the time of manufacture but it is not clear how this would affect reinspection capability.

283. *Question 254 above - It is not clear as to what Case-III deformations and strain will be limited. (See last sentence, next-to-last paragraph.) Does this mean limited to yield?*

Answer: The limits to be placed on deformations and strains for the reactor vessel will depend on the particular area or component of the vessel under consideration. Critical areas, such as support lugs, core support ledge, seals and penetrations, will receive much of the effects of an earthquake and have been designed with considerable conservatism. These critical areas will undoubtedly be able to take twice the design earthquake with little or no plastic deformation. Since the vessel, as presently under redesign, appears to be considerably heavier

than previously, it may require a reevaluation of these conservatisms to assure such a behavior. However, the basic intent of the Case-III limits is not necessarily to limit strains to the yield strain, but rather to assure function of the vessel during and after this loading situation.

284. *Question 255 above - Discuss the manner by which the criterion for 100% radiography of the pressure vessel may be satisfied in the case of the continuous annular ring welded to the inside surface of the vessel.*

Answer: The fabrication sequences for the continuous annular ring supporting the vessel internals will probably provide for radiography in the following manner:

- (a) A continuous groove, 1/2 in. deep, will be cut on the inner surface of the vessel at the elevation of the core support ring (ledge). This groove will be about 5 in. high. (If the shell course in this region is a forging this groove may be omitted.)
- (b) Starting in the groove a weld build-up will be placed such that after machining a continuous ring 2 in. wide is available on the I.D. of the vessel. This weld build-up will be fully x-rayed (radially).
- (c) The core support ledge annular ring will have its outer periphery prepared for welding to the weld build-up laid on in (b).
- (d) The core support ledge will then be welded to the weld build-up and this weld will be fully x-rayed (axially).
- (e) Paragraph N-462.3(c) may not be employed.

H. Questions from DRL.*

285. *Why are not wind loads included for Cases-III and IIIA in Table 157-2 (question 157 above)?*

Answer: This is an oversight: both Cases-III and III_A do include wind loads.

286. *Question 254 above - For the pressure vessel design, the legend for effects of MCA (jet forces, etc.) is not specified. Explicitly what forces or effects is ANL considering?*

Answer: The specific loadings to be considered as MCA effects are:

- (a) Jet forces from pipe breaks in either inlet or outlet primary coolant pipes (both in horizontal and vertical legs).
- (b) Depressurization in vessel.
- (c) Transient temperatures in vessel material.

287. *Question 254 above - Clarify the stress and strain limits for Case III of the pressure vessel design.*

Answer: For 304L austenitic stainless steel which is used for the reactor vessel, the value of the yield strength is an arbitrarily defined point, usually that point corresponding to the 0.2% offset line intersecting the actual stress-strain curve. Unlike an ideal elastic-rigid plastic, therefore, the effect of exceeding the yield point is not continued plastic deformation without increased stress, but rather increased deformation requiring increased stress to produce it. Finally, this material possesses very considerable ductility and can absorb considerable amounts of energy beyond the arbitrary yield strain with relatively little increase in deformation. Although the detailed stress analysis for the vessel will not be completed for several weeks or months, once the critical areas are defined these areas will receive a detailed analysis in order to assess the effects of earthquake loads (plus other loads), and to demonstrate that any "yielding" occurring will not in any way impair the safe shutdown and structural integrity of the reactor vessel. Application of a limit loading analysis as part of this work is under consideration.

Since the earthquake load is but one of many loads to be considered it is anticipated that "yielding" as such, will probably be localized (at nozzles and other prominent discontinuities) and should be only minor excursions into the "plastic" domain of this material. This is due to at least two reasons: (a) the vessel is quite conservatively designed in the complex belt course containing the beam tube nozzles and in other critical areas such as the closure flanges; and (b) the manner in which the vessel is supported (both support structure lugs and top lateral guide structure) which prevents the heavy masses (top

*Ed. Note: These questions were raised by DRL in a meeting on July 28, 1967. Answers were furnished in Reference 21.

closure head and internals) from causing significant loadings on the vessel shell.

Summarizing, it is our intent that rather than to specify an arbitrary amount of "beyond yield" deformation (if any) which will be allowed, we will provide complete detailed analysis of each such strain condition to satisfy the criterion, requiring neither loss of function nor loss of structural integrity.

288. *Clarify what ANL interprets as tornado design criteria.*

Answer: The as-built containment building will not necessarily retain its containment capability following a tornado, but any tornado damage will not interfere with the ability to shut down the reactor or to keep it safely shut down.

IV. References

1. Letter, K. A. Dunbar, AEC-CH, to A. V. Crewe, ANL, September 12, 1966.
2. "Preliminary Safety Analysis Report on the Argonne Advanced Research Reactor - Supplement 1" (November 9, 1966) - Unpublished.
3. Letter, A. V. Crewe, ANL, to K. A. Dunbar, AEC-CH, January 19, 1967 (transmittal of revised PSAR).
4. "Preliminary Safety Analysis Report on the Argonne Advanced Research Reactor - Index to Responses to AEC Questions" (January 23, 1967) - unpublished.
5. Letter, K. A. Dunbar, AEC-CH, to A. V. Crewe, ANL, January 31, 1967.
6. Letter, A. V. Crewe, ANL, to K. A. Dunbar, AEC-CH, February 9, 1967.
7. Letter, K. A. Dunbar, AEC-CH, to A. V. Crewe, ANL, March 21, 1967.
8. "Preliminary Safety Analysis Report on the Argonne Advanced Research Reactor - Supplement 2" (April 5, 1967) - unpublished.
9. Letter, K. A. Dunbar, AEC-CH, to A. V. Crewe, ANL, April 28, 1967.
10. Letter, A. V. Crewe, ANL, to K. A. Dunbar, AEC-CH, May 12, 1967.
11. Letter, K. A. Dunbar, AEC-CH, to A. V. Crewe, ANL, April 5, 1967.
12. Letter, K. A. Dunbar, AEC-CH, to A. V. Crewe, ANL, May 22, 1967.
13. Letter, A. V. Crewe, to K. A. Dunbar, AEC-CH, May 24, 1967.
14. Letter, W. M. Manning, ANL, to K. A. Dunbar, AEC-CH, June 7, 1967.
15. Letter, W. M. Manning, ANL, to K. A. Dunbar, AEC-CH, June 12, 1967.
16. Letter, W. M. Manning, ANL, to K. A. Dunbar, AEC-CH, June 15, 1967.
17. Letter, W. M. Manning, ANL to K. A. Dunbar, AEC-CH, July 18, 1967.
18. Letter, K. A. Dunbar, AEC-CH, to W. M. Manning, ANL, June 19, 1967.
19. Letter, W. M. Manning, ANL, to K. A. Dunbar, AEC-CH, July 7, 1967.
20. Teletype, M. Shaw, AEC-RDT, to K. A. Dunbar, AEC-CH, June 23, 1967.
21. Letter, M. Levenson, ANL, to A. E. Mravca, AEC-CH, August 7, 1967.

APPENDIX A

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

OCT 12 1967

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: ARGONNE ADVANCED RESEARCH REACTOR

Dear Dr. Seaborg:

At its ninetieth meeting, October 5-7, 1967, the Advisory Committee on Reactor Safeguards reviewed the proposed Argonne Advanced Research Reactor (AARR) which is to be constructed at the Argonne National Laboratory. A Subcommittee meeting was held in Washington, D. C. on September 29, 1967. During its review, the Committee had the benefit of discussions with representatives of the Argonne National Laboratory, the Division of Reactor Development and Technology, and the AEC Regulatory Staff and of the documents listed.

The proposed AARR is a light-water-cooled and moderated, beryllium-reflected, flux-trap reactor. Many of the features of the reactor are similar to those of the High Flux Isotope Reactor (HFIR), which was previously reviewed and discussed in Committee letters, dated May 9, 1960, July 25, 1960, July 15, 1965 and May 11, 1966. In particular, the fuel elements, core geometry, and control rods and drives are identical to those used in HFIR. Several features, however, differ significantly. The pressure vessel of the AARR is constructed of 304-L stainless steel. The number and size of the neutron beam tubes have been increased, and a larger number of rabbits and positions for long-term irradiations have been included in the internal thermal column. A reinforced concrete containment structure designed to withstand an internal pressure of 6 psig encloses the reactor.

The seismic design criteria are not yet completely defined. The Committee believes that questions related to these criteria should be resolved by the applicant with the Regulatory Staff before the containment base slab is poured.

The present review is the first of two reviews to be conducted at the construction phase. Although general design criteria are available, firm designs of several features, such as the containment structure, the pressure vessel, and instrumentation, will not be available for review until after contracts are let for design and construction. The ACRS will wish to review the designs as early as possible.

On the basis of the information presently available, it is the opinion of the ACRS that there is reasonable assurance that a reactor facility of the type proposed can be constructed and operated at the Argonne National Laboratory site without undue hazard to the health and safety of the public.

Dr. Herbert S. Isbin, Dr. Harry O. Monson, and Dr. David Okrent did not participate in the above review.

Sincerely yours,

N. J. Palladino

N. J. Palladino
Chairman

References:

1. Volume I, Preliminary Safety Analysis Report on the Argonne Advanced Research Reactor, dated March 31, 1966, Revised December 9, 1966.
2. Volume II, Appendices to the Preliminary Safety Analysis Report on the Argonne Advanced Research Reactor, dated March 31, 1966, Revised December 9, 1966.
3. Supplement 1 to the Preliminary Safety Analysis Report on the Argonne Advanced Research Reactor, dated November 9, 1966.
4. Index to Responses to AEC Questions, dated January 23, 1967, and Corrected Figure IV-G-3 to Preliminary Safety Analysis Report on the Argonne Advanced Research Reactor.
5. Supplement 2 to the Preliminary Safety Analysis Report on the Argonne Advanced Research Reactor, dated April 5, 1967.
6. Supplement 3, Compendium of Safety Considerations (Questions and Answers) Raised During the DRL Review of the Argonne Advanced Research Reactor PSAR, undated (received August 8, 1967).

Appendix B

SEISMIC DESIGN CRITERIA

As noted in Appendix A preceding, the seismic design criteria for the facility had not been agreed upon by the Laboratory and the AEC Regulatory Staff at the time of the ACRS consideration of the Project. Subsequent agreement was reached, however, and the acceptable criteria are given in the following table (originally published in "Reactor Development Program Progress Report - January 1968," ANL-7419, p. 140):

TABLE III.A.2. Allowable Stress Limits for AARR Facility

Loading Condition	Liner Plate		Pipes, Tanks, Switchgear, etc.	Reinforcing Steel	Concrete
	P _m	P _m + P _b			
FOR REACTOR CONTAINMENT BUILDING					
Case III DL + LL + PL + EL	S _m	1.5 S _m	-	0.5 f _y	0.60 f' _c
Case IIIA-DL + LL + PL + 2 EL	0.90 S _y	0.90 S _{CS}		0.90 f _y	0.85 f' _c
FOR ALL OTHER CLASS-I STRUCTURES, SYSTEMS, AND COMPONENTS					
Case III-DL + LL + PL + EL	-	-	0.90 of yield or equivalent ^a	0.90 f _y	0.85 f' _c
Case IIIA-DL + LL + PL + 2 EL	-	-	Generally not to exceed yield ^b		0.95 f' _c

^aThe "equivalent" value for austenitic pressure vessels remains to be defined.

^bIf yielding is to be permitted at local regions, analysis will be performed to determine failure modes, total deformation allowable will not exceed 50% of the deformation that would cause failure

Symbols for Table III.A.2

- DL = dead loads.
- LL = live loads.
- PL = design pressure load.
- TL = design temperature load.
- EL = design earthquake load (based on maximum horizontal ground acceleration of 0.1 g).
- 2 EL = maximum earthquake load (based on maximum horizontal ground acceleration of 0.2 g).
- P_m = primary membrane stress.
- P_b = primary bending stress.
- f_y = yield stress (minimum guarantees for reinforcing bars).
- f'_c = minimum ultimate compression strength of concrete.
- S_y = yield strength at temperature as specified in Table N-424, Section III, ASME Nuclear Vessel Code.
- S_{CS} = collapse stress as defined in Fig. 2 on p. 6 of Criteria of Section III, ASME Nuclear Vessel Codes (1964).
- S_m = allowable membrane stress as specified in Section VIII, ASME Boiler and Pressure Vessel Code times a factor of 1.10.