268
7-3-78

DR-220

HEDL-TME 78-35

UC 79 i,p

COMPARISON OF CONTAINMENT SYSTEMS FOR LARGE SODIUM-COOLED BREEDER REACTORS

MASTER

Hanford Engineering Development Laboratory

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

HANFORD ENGINEERING DEVELOPMENT LABORATORY
Operated by Westinghouse Hanford Company
A Subsidiary of Westinghouse Electric Corporation
Prepared for the U.S. Department of Energy
under Contract No. EY-76-C-14-2170
P.O. Box 1970 Richland, WA 99352

DISCLAIMER

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

DISCLAIMER

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

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the U.S. DOE, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

Printed in the United States of America
Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22161
Price: Printed Copy \$9.00; Microfiche \$2.25

HEDL-TME 78-35

UC 79 i,p

COMPARISON OF CONTAINMENT SYSTEMS FOR LARGE SODIUM-COOLED BREEDER REACTORS

NOTICE

**PORTIONS OF THIS REPORT ARE ILLEGIBLE. It
has been reproduced from the best available
copy to permit the broadest possible avail-
ability.**

Hanford Engineering Development Laboratory

S.E. Seeman
G.R. Armstrong

April 1978

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Department of Energy, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

HANFORD ENGINEERING DEVELOPMENT LABORATORY
Operated by Westinghouse Hanford Company
A Subsidiary of Westinghouse Electric Corporation
Prepared for the U.S. Department of Energy
under Contract No. EY-76-C-14-2170
P.O. Box 1970 Richland, WA 99352

COMPARISON OF CONTAINMENT SYSTEMS
FOR LARGE SODIUM-COOLED BREEDER REACTORS

S. E. Seeman and G. R. Armstrong

ABSTRACT

Three types of containment configurations for large breeder reactors (1000 MWe/2500 MWt) were compared to determine characteristic responses of each type to selected hypothetical accident conditions. The three types of system configurations were the single, double and containment-confinement designs. Two types of postulated hypothetical accidents were applied as a basis for evaluation - the energetic core disassembly accident and the whole core melt-through accident. More than 20 separate cases were analyzed in which design and accident assumptions were varied. Based on this study the containment-confinement configuration is shown to be the most promising for limiting off-site radiation doses for large sodium-cooled breeder reactors given the specific assumptions on the initial accident conditions.

ACKNOWLEDGMENTS

Significant contributions to this report were made by personnel other than the authors. Dr. Tang Chiao, Boeing Computer Services, performed the majority of the CACECO, HAA, and COMRADEX computer calculations. These calculations comprised a significant part of the total effort, and the authors are especially grateful. Valuable assistance on the codes used was received from HEDL personnel, Roy Peak (CACECO), Dick Shire (HAA), and Hal Martin (COMRADEX). The authors wish to acknowledge the guidance provided by Carl Newton, Fast Reactor Branch, RRT, DOE.

CONTENTS

	<u>Page</u>
Abstract	iii
Acknowledgments	iv
Figures	ix
Tables	xv
I. INTRODUCTION	1
II. RESULTS AND CONCLUSIONS	5
A. SUMMARY OF RESULTS	5
B. COMPARISON OF CONTAINMENT SYSTEMS	13
1. Single Containment	13
2. Double Containment	14
3. Containment-Confinement	14
4. Modified Systems	15
C. GENERAL CONCLUSIONS AND RESULTS	16
1. Configurations	16
2. Accident Factors	17
3. Design Factors	17
III. EVALUATION CONSIDERATIONS	21
A. REFERENCE REACTOR AND CONTAINMENT SYSTEMS DESCRIPTION	21
1. Introduction	21
2. Reactor Description	22
3. Single Containment Design (Case I)	28
4. Double Containment Design (Case II)	30
5. Containment-Confinement Design (Case III)	33
B. ACCIDENT SCENARIOS	35

CONTENTS (Cont'd)

	<u>Page</u>
1. Melt-Through CDA	35
2. Combined Accident	37
C. SOURCE TERM FACTORS	38
D. MAJOR DESIGN/PHENOMENOLOGICAL CONSIDERATIONS	40
1. Vent and/or Purge Mode	42
2. Liner Vent Discharge Path	43
3. Liner Integrity	44
4. Hydrogen Formation and Control	44
5. Melt-Through Time	46
IV. DISCUSSION OF CALCULATIONS	47
A. MATRIX OF CASES	47
B. CALCULATIONAL TECHNIQUES	51
1. CACECO Code	51
2. HAA-3B Code	55
3. COMRADEX Code	56
C. SPECIFIC COMPARISONS	59
1. Basic Containment Systems; Cases I-2, II-2 and III-1	59
a. Base Cases: Energy Sources in the RCB	65
b. Base Cases: Pressures and Temperatures	67
c. Base Cases: Hydrogen Concentrations	71
d. Base Cases: Aerosol in the RCB	74
e. Base Cases: RCB Leakages	74
2. Design Assumption Effects	78
a. Liner Vent System Effects	78

CONTENTS (Cont'd)

	<u>Page</u>
b. Cavity Pool Cooler Effects	80
c. Engineered RCB Vent/Purge	82
3. Accident Assumption Effects	85
a. Liner Conditions	85
b. Hydrogen Recombination	89
c. Melt-Through vs. Energetic Accident	90
4. Special Cases	91
a. Large Sodium Pool; Case III-Pool	91
b. High Conductivity Cavity; Case I-TC	92
c. Sodium Vent to Auxiliary Cells	94
d. Very High Pressure RCB; Case I-HP	94
D. RADIOLOGICAL EVALUATIONS	95
1. Base Case	99
2. Radiological Evaluations of Single Containment Cases	109
3. Radiological Evaluations of Double Containment Cases	114
4. Radiological Evaluations of Containment- Confinement Cases	123
5. Comparison of Containment Systems on the Basis of Radiological Consequences	131
V. RECOMMENDATIONS FOR ADDITIONAL WORK	137
VI. REFERENCES	139
APPENDIX A Core Inventory Library	A-1
APPENDIX B Assumptions Used in CACECO Calculations	B-1
APPENDIX C Inputs Used for HAA Calculations	C-1

FIGURES

<u>Figure</u>	<u>Page</u>
1 Event Tree for All Cases	6
2 Event Tree for Energetic Accident Cases	7
3 Event Tree for Single Containment Cases with Melt-Thru Accident	8
4 Event Tree for Double Containment Cases with Melt-Thru Accident	9
5 Event Tree for Containment-Confinement Cases with Melt-Thru Accident	10
6 Total Decay Power After Shutdown	27
7 Cross-Sectional View of Single Containment (Case I)	29
8 Schematic Cross Section of Reactor Cavity	31
9 Cross-Sectional View of Double Containment System (Case II)	32
10 Cross-Sectional View of Containment Confinement System (Case III)	34
11 Design Options for Containment Systems	48
12 Variation in Accident Considerations	49
13 Definitions of Containment Symbols	50
14 Energy Input Rates to the RCB for CASE I-2	66
15 Energy Input Rates to the RCB for Case II-2	66
16 Energy Input Rates to the RCB for Case III-1	68
17 RCB Pressures for the Base Cases	69
18 RCB Atmosphere and Surface Temperatures--Case I-2	69
19 RCB Atmosphere and Surface Temperatures--Case II-2	70
20 RCB Atmosphere and Surface Temperatures--Case III-1	70
21 RCB Hydrogen Concentrations	72
22 RCB Hydrogen and Oxygen Concentrations--Case I-2	72

FIGURES (Cont'd)

<u>Figure</u>	<u>Page</u>
23 RCB Hydrogen and Oxygen Concentrations--Case II-2	73
24 RCB Hydrogen and Oxygen Concentrations--Case III-1	73
25 Sodium In Rate to RCB for Base Cases	75
26 Suspended Concentration of Aerosol in RCB for Base Cases	76
27 Leak Rates from the RCB for Base Cases	77
28 Total Amount of Water Entering RCB	79
29 RCB Pressure vs Time	79
30 H ₂ , O ₂ and H ₂ O in RCB Atmosphere--Case I-1	81
31 Energy Balance in Cavity Pool--Case I-2	81
32 Pool Temperatures for Various Sized Coolers	83
33 Sodium Boiloff vs Cavity Pool Cooler Size	83
34 Effect of Pool Coolers on RCB Atmosphere Temperature	84
35 Effect of Purge on RCB Hydrogen Concentration	84
36 Effect of Purge on RCB Temperatures	86
37 Hydrogen Formed in Cavity due to Floor Liner Failure	86
38 Effects of Cavity Floor Liner Failure on Hydrogen Concentration--Cases I-2, I-3(2)	88
39 Effect of Floor Liner Failure on RCB Hydrogen Concentration--Cases II-2, II-3(2)	88
40 Effect of Hydrogen Recombination Assumptions on Failed Liner--Case I-3(2)	89
41 Effect of Hydrogen Recombination Assumptions on Failed Liner--Case II-3(2)	89
42 Effect of Head Expulsion on RCB Pressure	93
43 Effect of Large Sodium Pool--Case III-Pool	93

FIGURES (Cont'd)

<u>Figure</u>	<u>Page</u>
44 Effects of High Pressure (50 psig) RCB	96
45 Contributions to Whole Body Dose from Noble Gases-- Case I-2--800 meters	101
46 Contributions to Whole Body Dose from Volatile Isotopes--Case I-2--800 meters	101
47 Total Whole Body Dose as a Function of Time at a Distance of 800 meters--Case I-2	102
48 Contributions to Whole Body Dose from Noble Gases-- Case I-2--10,000 meters	102
49 Contributions to Whole Body Dose from Volatile Isotopes--Case I--10,000 meters	104
50 Total Whole Body Dose as a Function of Time-- Case I-2--10,000 meters	104
51 Total Whole Body Dose as a Function of Time for Various Distances--Case I-2	105
52 Thyroid Dose as a Function of Time for Various Distances--Case I-2	105
53 Lung Dose as a Function of Inhalation Time for Various Distances--Case I-2	106
54 Bone Dose as a Function of Inhalation Time for Various Distances--Case I-2	106
55 Comparison of 2-hour Total Whole Body Dose for Single Containment Systems	110
56 Two-Hour Internal Dose to Various Organs--Case I-4(2)	110
57 Comparison of 30-Day Total Whole Body Dose for Single Containment Systems	111
58 Comparison of 30-Day Thyroid Dose for Single Containment Systems	111
59 Comparison of 30-Day Lung Dose for Single Containment Systems	113
60 Comparison of 30-Day Bone Dose for Single Containment Systems	113

FIGURES (Cont'd)

<u>Figure</u>	<u>Page</u>
61 Comparison of 800-Meter Bone Doses for Single Con- tainment Systems with and without Head Expulsion	115
62 Comparison of Two-Hour Total Whole Body Dose for Double Containment Systems	115
63 Two-Hour Internal Dose to Various Organs--Case II-4(2)	119
64 Comparison of 30-Day Total Whole Body Dose for Double Containment Systems	119
65 Comparison of 30-Day Thyroid Dose for Double Contain- ment Systems	120
66 Comparison of 30-Day Lung Dose for Double Containment Systems	120
67 Comparison of 30-Day Bone Dose for Double Containment Systems	121
68 Two-Hour Total Whole Body Doses for Containment- Confinement Systems	121
69 Comparison of Two-Hour Total Whole Body Doses for Base Containment-Confinement System Variations	124
70 Two-Hour Internal Doses--Case III-4(1)	124
71 Comparison of 30-Day Total Whole Body Doses for Con- tainment-Confinement Systems	126
72 Comparison of 30-Day Thyroid Doses for Containment- Confinement Systems	126
73 Comparison of 30-Day Lung Doses for Containment- Confinement Systems	127
74 Comparison of 30-Day Bone Doses for Containment- Confinement Systems	127
75 Doses vs Distance for Case I-2, Filtered, with Elevated Release	129
76 Comparison of Two-Hour Total Whole Body Dose for Selected Containment Systems	129
77 Comparison of Two-Hour Total Lung Dose for Selected Containment Systems	130

FIGURES (Cont'd)

<u>Figure</u>	<u>Page</u>
78 Comparison of Two-Hour Total Bone Dose for Selected Containment Systems	130
79 Comparison of 30-Day Total Whole Body Dose for Selected Containment Systems	133
80 Comparison of 30-Day Thyroid Dose as a Function of Distance for Selected Containment Systems	133
81 Comparison of 30-Day Lung Dose as a Function of Distance for Selected Containment Systems	134
82 Comparison of 30-Day Bone Dose as a Function of Distance for Selected Containment Systems	134

TABLES

<u>Table</u>	<u>Page</u>
1 Approximate Distances Required to Meet Guideline Dose Limits	11
2 Types of Containment Systems used in Current Breeder Reactor Plants	23
3 Summary of Reference Containment Design Parameters	24
4 Properties of FFTF and Conceptual Breeder Reactors	26
5 Release Factors for Various Isotopes	41
6 Case Matrix for Single Containment Systems	53
7 Case Matrix for Double Containment and Containment-Confinement Systems	54
8 Chronology of Accident Effects for Single Containment Base Case I-2	60
9 Chronology of Accident Effects for Double Containment Base Case II-2	62
10 Chronology of Accident Effects for Containment-Confinement Base Case III-1	64
11 Dose Limits for Site Evaluation	98
12 Contribution to 1.0-Day Internal Doses at 800 Meters--Case I-2	107
13 Contribution to 30-Day Internal Doses at 20,000 Meters--Case I-2	108
14 Contribution to 30-Day Internal Doses at 20,000 Meters--Case I-4(2)	116
15 Contribution of Isotopes to 30-Day Internal Doses at 20,000 Meters--Case III-1	128
16 Approximate Distances Required to Meet Guideline Dose Limit	135
A-1 Core Inventory Library	A-4
A-2 Input to RIBD	A-15
A-3 Radioactivity of Actinides in Spent LMFBR Core Fuel as a Function of Decay Time	A-16

TABLES (Con't)

<u>Table</u>	<u>Page</u>
A-4 Input for Sodium-23 Activation Calculations	A-17
B-1 Concrete Water Release upon Heating	B-3
B-2 Effective Addition to Concrete Heat Capacity due to: A) Latent Heat of Water Released; and B) Carbon Dioxide	B-4
B-3 Thermal Conductivity of Concrete	B-5
B-4 Carbon Dioxide Release upon Heating Limestone Concrete	B-5
B-5 Heats of Reactions of Sodium	B-6
B-6 Heat Structures	B-7
B-7 Thermal Conductivity and Volumetric Heat Capacity	B-9
B-8 Natural Convection Heat Transfer Coefficients for Nitrogen	B-10
B-9 Natural Convection Heat Transfer Coefficients for the Cavity Pool	B-11
B-10 Natural Convection Heat Transfer Coefficients for Nitrogen at 100% Density	B-12
B-11 Nitrogen Natural Convection Combined with Insulation Heat Conduction	B-13
B-12 Outside-to-RCB Leak Coefficients	B-16
B-13 RCB-to-Infinite Volume Leak Coefficients	B-16
B-14 Cell Characteristics	B-18

COMPARISON OF CONTAINMENT SYSTEMS FOR LARGE SODIUM-COOLED BREEDER REACTORS

I. INTRODUCTION

A primary function of reactor containment systems is to mitigate the consequences of release of radioactive materials from the primary system in the event of an accident. Potential accidents, postulated as bases for containment/radiological consequence studies, can vary widely in severity and probability. Those accidents that place the most stringent demands on containment performance are very low probability core disruptive events, commonly referred to as hypothetical core disruptive accidents (CDA's). Two types of CDA's, generally considered in breeder reactor safety assessments as bases for safety technology research and for risk assessments, are: 1) an energetic disassembly which is followed by core melt-through of the reactor vessel; and 2) a benign disassembly followed by core melt-through of the reactor vessel. Each follows significantly different patterns and places different demands on the containment systems. The energetic CDA followed by a core melt-through is of much lower probability than the benign CDA, but has potentially higher consequences.

Many reactor containment configurations have been considered in the international breeder reactor program. These are broadly categorized in this study as single, double, multiple, and containment-confinement configurations. At the outset of this study, no comparative assessment of the respective capabilities of these configurations for mitigating accident consequences had been made.

The containment system configurations used in the study were developed from characteristics extrapolated from existing breeder reactor plants or those similar to conceptual studies of large commercial breeders. Thus, the descriptions are not specific to any particular plant or concept.

The purpose of this report is to provide insight into the performance characteristics and requirements of basic containment configurations for commercial-size sodium-cooled breeder reactors (i.e., ~1000 MWe/2500 MWt) by the use of scoping calculations based on the energetic and melt-through CDA scenarios. These two series of events were chosen as bases for the comparisons performed in the study. No consideration is to be inferred regarding these conditions as design basis accidents for breeder reactor plants. Information from the study can provide additional insight into the configuration selection process and into the assessment of various design options given specific reactor conditions.

The objectives of this study are to:

- a) Provide a comparative assessment of various selected breeder reactor containment configurations based primarily on radiological consequences, a major factor in determining risk.
- b) Indicate the sensitivity of configuration performance to design options and to selected option variations such as vent and purge mode, cavity liner vent discharge path, and containment volume.
- c) Provide comparative estimates of various phenomenological assumptions on containment performance e.g., liner integrity, hydrogen recombination, sodium/concrete interactions.
- d) Determine the most promising containment system for large sodium-cooled breeder reactors based on the cases considered.
- e) Provide guidance in establishing base safety technology programs in support of future 2500 MWt breeder reactor plants.

To achieve these objectives, selected variations in design and source term generation factors were applied to the CDA scoping calculations to establish

their effects and importance on containment performance. In this study, critical parameters--containment pressure, temperature, and H₂ concentration, were monitored to establish the sensitivity of each to design and source term variations. The goal was to determine the doses at site boundaries for comparative purposes.

Additional aspects of the accidents which affect containment requirements were investigated. For instance, certain containment design options, such as reactor cavity pool cooling, which could significantly reduce the consequences of an accident, were inspected more closely to determine boundary design conditions necessary to achieve desired results. Accident cases were also analyzed to determine the relative contribution of selected factors to the accident scenario, e.g., the energy contribution from various sources that contribute to the thermal/mechanical loads on the reactor containment building.

Based on risk (probability of the accident times the consequence) considerations, the CDA may possibly be classified as beyond the design basis type accident.⁽¹⁾ Because these accidents are remote in probability, the environmental risk is extremely low. As a result, consequence limitations are not subject to 10 CFR 100 guidelines. Since establishing risk is based in part on consequence, the containment performance could influence such a classification. This study indicates the relative effect of containment design options on the consequences.

This study provides direction for the selection of the optimum containment system for large sodium cooled breeder reactors. However, the cost of various options has not been assessed and cost/benefit analyses were not performed. In addition, analyses of the facility structure integrity when exposed to the calculated accident conditions are not covered in this report, although assumptions are made in special cases to establish the effects of certain structural failures.

The optimum design must include the influence of the limiting criteria for a particular reactor. These criteria include the physical siting limitations

singularly applicable to a specific location, the specified time periods before containment release to the environment is allowed, and the requirements to mitigate special arbitrary source terms. The source terms used in this study will be sufficiently representative of potential arbitrary large sodium-cooled breeder reactor source terms to provide an adequate information base for a valid relative ranking of the containments, when such source terms, for example site suitability source terms, are defined and considered. It is not expected that a reasonable definition of arbitrary source terms will alter the containment rankings resulting from this study.

The first part of the study is concerned with the selection of reference reactor and containment systems. Chapter III includes physical descriptions of the reactor and the containment systems selected for use in this study. Chapter III also includes a discussion of the two types of accidents considered, the energetic core disassembly accident and the melt-through accident, and the related source terms, as well as a preliminary discussion of some design and accident factors affecting the consequence calculations reviewed in Chapter IV. Chapter IV defines the matrix of cases, reviews the analysis methods used, and then presents the specific study results: first, for the three basic configurations; second, for special accident and design assumptions, e.g., cavity liner conditions, hydrogen recombination, and cavity liner discharge path; and third, for the radiological evaluation of all the matrix cases. Supplemental information is included in the appendixes.

II. RESULTS AND CONCLUSIONS

A. Summary of Results

In studying the comparative performance of the single, double, and containment-confinement configurations more than 20 cases were considered. The event tree for the cases studied is started in Figure 1 and continued for the various accidents and containment configurations in Figures 2, 3, 4, and 5.

For clarification, the following definitions apply for this study: the single containment is basically a low-leakage air-filled enclosure (building); double containment is a low-leak inner and outer enclosure with the inner enclosure (dome over the head compartment) filled with an inert gas; the containment-confinement is a low leak inner enclosure filled with inert gas or air and a high leakage outer enclosure with the gap maintained at a slightly negative pressure. Exhaust from the outer enclosure of the containment-confinement systems is processed through a cleanup system prior to release to the environment.

Given the assumed accident scenario, the results of the study show that the containment-confinement configuration, with an engineered exhaust cleanup system, appears to be the best containment configuration for large 2500 MWt sodium-cooled breeder reactors. This conclusion is based on a comparison of the distances required to meet guideline dose limits for the containment systems studied. This summary information is presented in Table 1. If engineered safety features, such as reactor cavity pool coolers, were added to the containment system, double and single containment systems become much more attractive. The use and acceptability of such engineered safety systems will depend, in part, on the confidence level established for these systems.

The results of this study indicate a basic containment-confinement configuration that can serve as a valid starting point from which a designer can proceed with minimum modifications to reach a final optimum design. The following are general comments on this containment system:

HEDL 7708-8.2

FIGURE 1. Event Tree for All Cases.

* CACECO - ONLY CASE

HEDL 7708-8.6

FIGURE 2. Event Tree for Energetic Accident Cases.

*CACECO - ONLY CASES

HEDL 7708-8.1

FIGURE 3. Event Tree for Single Containment Cases with Melt-Thru Accident.

HEDL 7708-8.5

FIGURE 4. Event Tree for Double Containment Cases with Melt-Thru Accident.

*CACECO - ONLY CASES

**BY DEFINITION IN THIS STUDY, ALL CONTAINMENT-CONFINEMENT CASES HAVE PURGED REACTOR CONTAINMENT BUILDINGS AND FILTERED, ELEVATED RELEASES.

+ SPECIAL CASE, USED ONLY FOR INTERMEDIATE DATA POINT

HEDL 7708-8.7

FIGURE 5. Event Tree for Containment-Confinement Cases with Melt-Thru Accident.

TABLE 1

APPROXIMATE DISTANCES (IN METERS) REQUIRED TO MEET GUIDELINE DOSE LIMITS

DOSE PERIOD *	ORGAN	SINGLE CONTAINMENT CASES							DOUBLE CONTAINMENT CASES					CONTAINMENT-CONFINEMENT CASES			
		I-1	I-2	I-3(2)	I-4(1)	I-4(2)	I-5a (2)	I-5c (2)	II-1	II-2	II-3(2)	II-4(2)	II-6(1)	III-1	III-4(1)	III-3(1)	III-5b (1)
2-HOUR **	TOTAL WHOLE BODY	800	800	800	950	950	800	800	700	700	700	700	700	900	800	900	900
	THYROID	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)
	LUNG	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)
	BONE	(-)	(-)	(-)	2500	1500	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)
30-DAY ***	TOTAL WHOLE BODY	5000	4000	5000	8000	4000	2500	1100	6000	2600	3500	2600	13,000	5500	5500	18,000	4000
	THYROID	(+)	(+)	(+)	(+)	20,000	13,000	(-)	(+)	20,000	20,000	20,000	(+)	(-)	(-)	(-)	(-)
	LUNG	9000	5000	7000	10,000	5000	3000	(-)	11,000	4000	4000	4000	(+)	(-)	(-)	(-)	(-)
	BONE	850	(-)	700	17,000	2200	(-)	(-)	1,000	(-)	(-)	600	2300	(-)	(-)	(-)	(-)

* All doses calculated based on 50-year commitment
 ** For "Exclusion Area"
 *** For "Low Population Zone"

(-) Less than 600 meters
 (+) More than 20,000 meters

- General Design - The inner containment should be much larger than that used for this study; its volume should approach that of the Reactor Containment Building (RCB) of the single containment design. The outer (confinement) building should be constructed of materials (concrete, for example) which provide significant shielding for gamma radiation.
- Reactor Cavity - The reactor cavity should have a liner designed to mitigate the effects of a melt-through. Accommodation in the design should be made to vent any water released from the concrete to condenser/drain rooms.
- Inner Containment - The inner containment structure should be large enough to be considered the working RCB. In this study the RCB is defined as that building in which personnel routinely work during reactor operation. Its leakage design requirements should approach about 0.1%/day at 10 psig. Higher leak rates may possibly be acceptable. Pressure relief valve operation should be provided to allow venting prior to reaching pressures which would structurally fail the containment.
- Exhaust Cleanup System - The volume between the inner containment and the outer confinement building should be held at a slightly negative pressure during accident situations. This is accomplished by the exhaust system. The cleanup system for this exhaust flow should have the capability of handling the large quantities of sodium oxide generated during the accident. Elevated releases will provide an additional means for reduction of doses at close distances.
- Confinement Building - The confinement building should incorporate provisions in its design for reducing the direct dose at short distances. The easiest way to accomplish this would be to shield the lower portions with concrete. This would also provide external missile protection. The leakage requirements on this building can be modest, 15,000 cfm at normal operating pressure (~ -1.5 in H_2O).

B. Comparison of Containment Systems

The comparative strengths and weaknesses of each of the major containment types studied (single containment, double containment, and containment-confinement) are reviewed in this section. Options for improvement of the relative ranking of each system are also presented. The conclusions drawn are based on the accidents postulated and the assumptions made in the study. In some cases, reasonable extrapolations are made to answer the "what if" question.

1. Single Containment

For short times, the radiological consequences of using the single containment system are much like those of the other systems primarily because in two hours, the sodium pool has not yet begun to boil. However, if part of the core and sodium is expelled from the reactor head at the start of the accident, the single containment case shows a definite weakness. Even with the low leakage characteristics of the RCB, some plutonium is calculated to leak out and give rise to potential bone and lung doses that do not occur for those cases with an inner containment dome covering the head access compartment.

For longer times, the single containment system again ranks low. Thirty-day doses to the thyroid at a distance of 20,000 meters are in excess of 300 Rem.

For the large size sodium-cooled breeder reactor (SCBR) used in this study, the single containment system, without added engineering safety features, ranks lowest of the systems studied.

The addition of a 10-megawatt cavity pool cooler makes the single containment more attractive. Thyroid and lung doses are negligible, but early direct and bone doses due to the noble gases and plutonium in the head expulsion source term are still present.

2. Double Containment

The double containment design incorporates a sealed dome over the head access compartment of the single containment design. It was expected at the beginning of this study that this inner containment would significantly mitigate the accident consequences. However, once boiling commences in the cavity area, the atmosphere temperature and pressure in the domed area quickly become excessive and relief is required. Thus, after about 12 hours, the single and double containment system consequences look similar. The one important exception to this is if expulsion through the head is assumed. The initial expulsion during the energetic phase of a combined expulsion/melt-through accident is successfully controlled by the inner containment dome. For this reason, the double containment system represents a significant improvement over the single containment system during the initial stages of the accident.

3. Containment-Confinement

The containment-confinement case has engineered features that result in large dose reductions. It has the sealed head compartment which effectively handles the head expulsion accident. It cleans up all exhaust air, except by-pass leakage, and releases the cleaned up atmosphere at an elevated location (stack). These features make the containment-confinement design the most attractive when considering off-site dose factors.

The most significant drawback to the containment-confinement system is the excessive temperature that results from the sodium-oxygen reactions. Incoming sodium is constantly fed oxygen by virtue of the RCB exhaust design. The study assumed that the temperatures are not so excessive that exhaust/cleanup is affected. It may be that failure of these exhaust filter systems at some time would still result in doses that are lower than for other systems, but this is not investigated in this study.

The limiting dose for the containment-confinement system is the total whole body dose caused by direct and external radiation by noble gases. The noble gases, released from the reactor vessel early in the accident, accumulate in the inner dome and result in an early direct radiation dose outside of the RCB. At the same time, hydrogen, created by sodium/concrete reactions in the cavity, leaks into the relatively small containment dome and leads to pressurization. Sodium vapor entering the dome also adds to the pressurization. Soon the noble gases leak into the RCB where they are then removed by exhaust fans. They are not filtered out, resulting in an early stack release of noble gases. This, then, leads to early external doses as the noble gases pass downwind. The other systems promote longer holdup of the noble gases and reduce the activity level at the time the gases finally leak out. If the inner containment of the containment-confinement system is enlarged, however, this difference is diminished.

4. Modified Systems

From the results of this study, the most attractive containment system for large SCBR's appears to be a system that combines the best features of the single, double, and containment-confinement designs. This system, outlined in Chapter II, Section A, is a containment-confinement design whose inner containment resembles the RCB of the single containment case. The outer confinement structure is shielded with concrete to reduce the direct dose. Noble gases will remain for longer times within the inner containment because of its large size and low leakage characteristics.

Another modified system that appears to have merit is a double containment design with purge, cleanup, and stack release capability. This is essentially a double containment system that has the ability to become a containment-confinement system when the design pressure of the outer containment is approached. The advantage of the double containment system is its inherent low early doses. The advantage of the containment-confinement system is its low long-time doses. A concrete RCB would reduce direct doses.

A further modification that can be applied to either the double containment or containment-confinement system is the addition of an expansion volume connected to either the inner dome volume or the cavity volume. Although not studied by the authors, it has been investigated by others (Chapter IV, Section C4), and it appears to offer a reasonable method of mitigating potential releases. The expansion volume might consist of below-grade rooms already in existence.

C. General Conclusions and Results

The following represents general conclusions that were made during the study. These are separated into categories related to configuration, accident, and design.

1. Configurations

- a. Based on "exclusion distances," if only the melt-through case is considered, the two-hour whole body dose is the controlling dose in ranking the performance of the configurations studied. If the head expulsion accident is added, only single containment performance is affected; and early bone doses rise significantly.
- b. Based on the "low population zone" distances, the 30-day thyroid dose is the controlling dose in ranking the performance of the configurations studied. This is true for cases whose exhaust to the environment is not filtered, or whose design does not include pool coolers. The limiting dose in cases using filtered exhausts or pool coolers is the total whole body dose.
- c. Based on the configurations and the release assumptions studied, the containment-confinement configuration with an inner containment approaching the single containment volume provides the optimum protection.

- d. For the melt-through-only case, the double containment provides no significant or practical improvement over the single containment. If the head expulsion accident is added, the double containment provides a significant improvement over the single containment configuration when considering two-hour dose limitations, but no practical advantage when considering the 30-day dose limitations.

2. Accident Factors

- a. In all cases without pool coolers, sodium in the cavity reached its boiling point in seven to eight hours; boil-dry occurred between 135 and 150 hours (~6 days).
- b. High pressure (10 psig), low leakage containment without cavity pool coolers must be relieved at approximately 12 hours due to hydrogen or pressure buildup resulting from the melt-through accident.
- c. If hydrogen recombination is assumed, complete reactor cavity floor liner failure does not significantly affect the radiological consequences of the accident except for containment-confinement configurations. In this case, early liner failure increases the direct and external whole body doses. If the liners do not fail, the radiological consequences are independent of hydrogen recombination assumptions.

3. Design Factors

- a. The best possibilities for increasing the time to relieve high pressure containments are:

- Use of cavity pool coolers
 - Use of inerted chambers connected to the reactor cavity for expansion volume and for inherent heat sinks.
- b. An 8-MW cavity pool cooler reduces the melt-through accident to one involving only the release of noble gases. A 2.5-MW cavity pool cooler prevents boil-dry.
 - c. Venting the cavity atmosphere into typical heat transport cells can significantly increase the time of RCB venting.
 - d. Engineered vent and purge cases resulted in RCB atmosphere temperatures significantly higher, $\sim 400\text{ F}^\circ$ (222 C°), than the unpurged cases due to the increased sodium-oxygen reaction.
 - e. Venting the cavity concrete water release to the RCB atmosphere causes the undesirable effects of increasing the rates of hydrogen and pressure buildup in the RCB. It is thus desirable to vent the cavity concrete water to locations other than the RCB.
 - f. Shielding can significantly reduce the two-hour whole body dose, which is the limiting two-hour dose in all melt-through-only cases.
 - g. Cleaned and filtered releases at the time of venting can reduce the 30-day internal doses approximately in proportion to the cleanup system efficiency.
 - h. As expected, elevated release significantly reduces close-in doses.
 - i. An increased sodium inventory results in a disproportionate delaying effect on the rate of sodium heating and boil-off, e.g., an increase of three in the sodium inventory increases the time to reach boiling by a factor of eight.

- j. Increasing the thermal conductivity of the cavity concrete by a factor of three did not significantly affect the melt-through accident scenario.
- k. Judicious selection of design factors, which include sodium inventory, concrete thermal conductivity, and expansion volumes could potentially lead to a system that would remain inherently cooled for long periods, if not permanently.

III. EVALUATION CONSIDERATIONS

A. Reference Reactor and Containment Systems Description

1. Introduction

At the onset of this study, two basic decisions were necessary: 1) choose the reactor size and description; and 2) determine the types of containment systems to be investigated. It was decided to use as a basis the 2500 MWt breeder reactor plant because of its expected similarity to the commercial-sized breeder plants. A detailed description of the reactor is presented in Section A-2.

A review of existing and planned breeder reactor plants indicated that no one type of containment system can be designated as a standard for future large plants.⁽²⁻¹⁵⁾ However, all of the containment options could be roughly grouped into four categories:

- Single Containment - The RCB (Reactor Containment Building) is a high-pressure (10 psig or above), low-leakage building, with open access to the head compartment over the reactor and a normal air environment for unrestricted personnel operations. As noted in this study, the RCB refers to the containment enclosure in which personnel routinely work during reactor operation.
- Double Containment - The RCB is the same as that for a single containment, i.e., low-leakage high-pressure design. However, the head access area is sealed by a high pressure (inner containment) barrier containing an inert atmosphere.
- Containment/Confinement - The RCB is a high-leakage, low-pressure building with filtered exhaust. The head access area is sealed by a high-pressure barrier identical to that of the double containment concept.

- Multiple Containment, with Pumpback - This design uses a low-leakage, high-pressure RCB and a sealed head access area, like the double containment system. However, there is an additional building outside of the RCB. A negative pressure zone is maintained between the RCB and the external building by pumping back leakages to the main containment space. Eventual venting to a stack via an air cleaning system is also provided. This system was not evaluated in any detail in this study.

Some of the present and planned breeder reactors are categorized according to containment system in Table 2.

Only the first three basic containment types, single containment, double containment, and containment-confinement, were chosen for analysis. This choice of containment designs has been used earlier by Hilliard, et.al., in an evaluation of air cleaning system concepts for breeder reactors.⁽¹⁶⁾ Variations within the basic systems and the accident scenario/assumptions provided a large number of cases for study. In all of the cases analyzed in this study, the 2500 MWt core described in Chapter III Section A-2 was used. The basic containment systems are described in Chapter III Section C-E. A summary of design parameters of the basic systems is shown in Table 3.

2. Reactor Description

The basic reactor chosen for this work is one that was previously developed by Atomics International, their "Reference Oxide Design".⁽¹⁷⁾ It is a 2500 MW-thermal, 1000 MW-electrical, mixed plutonium-uranium oxide, loop-type reactor.

TABLE 2
TYPES OF CONTAINMENT SYSTEMS USED IN CURRENT BREEDER REACTOR PLANTS

<u>Containment Type</u>	<u>Description</u>
Single	Open head compartment and low-leakage outer containment building ^(a)
Double	Sealed, inerted high pressure inner containment barrier, surrounded by a low-leakage outer containment building ^(a)
Containment/ Confinement	Sealed, low-leakage inner containment barrier, surrounded by a ventilated low pressure outer confinement building with discharge to stack via an air cleaning system
Multiple, with Pumpback	Sealed, high pressure inner containment barrier, surrounded by one or more outer barriers ^(a) . A negative pressure zone is maintained in the outermost space by pumping back leakages to the inner containment space. Eventual venting to stack via air cleaning system is provided

(a) The RCBs in these cases are designed to withstand 10 psi or greater differential pressures.

TABLE 3

SUMMARY OF REFERENCE CONTAINMENT DESIGN PARAMETERS

Parameter	Case I	Case II	Case III
	Single Containment	Double Containment	Containment-Confinement
<u>Brief description</u>	Single, low-leakage cylindrical steel shell surrounding all primary sodium systems.	Sealed, inerted, high-pressure <u>inner</u> containment surrounding reactor vessel and head compartment. An <u>outer</u> low-leakage cylindrical steel shell surrounds the inner containment.	Sealed, inerted, high-pressure <u>inner</u> containment surrounding reactor vessel and head compartments. A ventilated rectangular building surrounds the inner containment.
<u>Inner containment</u>			
. Atmosphere	N/A	98% N ₂ /2% O ₂	* 98% N ₂ /2% O ₂
. Size	N/A	40 ft D x 50 ft H, (12.2 m x 15.2 m) hemispherical top	40 ft D x 50 ft H, (12.2 m x 15.2 m) hemispherical top
. Volume, ft ³ (m ³)	N/A	29,300 (830)	29,300 (830)
. Leak rate, %/day	N/A	100 at 15 psid**	100 at 15 psid**
<u>Outer containment</u>			
. Atmosphere	Air	Air	Air
. Size above operating floor	160 ft D (48.8 m) x 60 ft (18.3 m) straight cylinder, with hemispherical top	160 ft D (48.8 m) x 60 ft (18.3 m) straight cylinder, with hemispherical top	105 ft x 204 ft x 105 ft (32.0 m x 62.2 m x 32.0 m)
. Volume, ft ³ (m ³)	2.25 x 10 ⁶ (63700)	2.25 x 10 ⁶ (63700)	2.25 x 10 ⁶ (63700)
. Leak rate, %/day	0.1 at 10 psid***	0.1 at 10 psid***	15,000 ft ³ /min (ventilation) (7.1 m ³ /s)

*Considered, also, was an inner containment with characteristics similar to the single containment.

**A pressure difference of 103.4 kPa

***A pressure difference of 68.9 kPa

A listing of core and blanket data for this reactor, FFTF, and conceptual plants (Atoms International (AI) Babcock & Wilcox, Combustion Engineering, General Electric and Westinghouse) is made in Table 4. The AI Reference Oxide Design has the highest specific power of the plants listed in this table.

The computer code RIBD⁽¹⁸⁾ was used to calculate fission product activity, power, and energy release. Input fluxes, power levels, actinide concentrations, and burnup numbers for RIBD were taken from a report by Oak Ridge⁽¹⁹⁾. The plutonium charged to the core was assumed to have the typical isotopic concentration produced by future light water reactors.

The RIBD input numbers for the core, radial blanket, and axial blanket are shown in Appendix A. Corresponding output fission product concentrations also are listed in Appendix A. A plot of decay heat versus time is shown in Figure 2. All results are for "End of Equilibrium Cycle". The cycling scheme assumed that one-third of the reactor (core and blankets) was replaced during a 27-day downtime following the first power cycle (153 days). The second cycle is a repeat of the first. The third cycle consists only of a power cycle at the end of which the inventory is calculated.

The inventory of ^{22}Na and ^{24}Na was calculated using standard activation equations.⁽²⁰⁾ The input numbers for the activation calculations and results are shown in Appendix A.

The numbers important for radiological consequence evaluation are: 1) the decay heat following the accident; and 2) the total isotopic inventory that could be available for release.

The decay heat numbers needed by CACECO involve the total heat generated by isotopes in the core, radial blanket, and axial blanket. Figure 2 shows the total.

TABLE 4
PROPERTIES OF FFTF AND CONCEPTUAL BREEDER REACTORS(19)

	FFTF	AI ^(a) Ref. Oxide	Follow-On Plants ^(b)				
			AI	B&W	CE	GE	W ^(c)
Fuel Form	Oxide	Oxide	Oxide	Oxide	Carbide	Oxide	Carbide
Nuclear data (at full power)							
Power, MW(thermal)	400	2500	2400	2450	2465	2417	2600
<u>Core</u>							
Avg. sp. pwr., MW/metric ton (U + Pu)	158	175	115	103	122	157	161
Burnup, MWd/metric ton	45,000	80,000	67,400	100,000	100,000	100,000	100,000
Charge, metric ton (U + Pu)	2.5	12.6	19.1	23.9	17.32	13.45	13.7
Charge, metric ton (fissile Pu)	0.51	1.96	2.33	2.77	1.92	1.62	1.98
Charge, Pu/(U + Pu)	0.23	0.22	0.171	0.16	0.16	0.17	0.18
Rod avg. linear pwr., kw/ft	7.8	10.0	10.1	8.3	29.5	9.63	18.5
Rod max. linear pwr., kw/ft	13.1	16.	16.0	14.4	43.5	16.3	39.2
Max. clad. temp., °F		1230	1220	1174	1170	1300	1300
Max. fuel temp., °F	4200	4748		4000	2310		2620
<u>Axial blanket</u>							
Avg. sp. pwr., MW/metric ton (U)		5.5	5.71	2.83	10.2	12.2	
Burnup, MWd/metric ton (U)		2500	3340	2000	7800	7200	
Charge, metric ton (U)		7.32	13.1	19.1	25.97	15.0	11.
<u>Radial blanket</u>							
Avg. sp. pwr., MW/metric ton (U)		10.	7.89	8.38	24.2	6.95	3.0
Burnup, MWd/metric ton (U)		8100	12,700	9,000.	57,000	7000	
Charge, metric ton (U)		26.7	15.6	16.1	5.69	16.7	33.

^a The AI Ref. oxide reactor was chosen for this study

^b AI = Atomics International; B&W = Babcock and Wilcox; CE = Combustion Engineering; GE = General Electric; W = Westinghouse.

^c Later Westinghouse study (Ref. 14) used oxide core, 2550 MW_t.

FIGURE 6. Total Decay Power After Shutdown.

The isotope inventory library needed by COMRADEX should include all isotopes that could be available for release. The library used in this study included the following:

- 1) 100% of Core Isotopes generated by RIBD.
- 2) 100% of Core Actinides as generated by Oak Ridge⁽¹⁹⁾
- 3) 100% of Volatile, Sodium-Soluble Isotopes (As, Se, Br, Rb, Cd, I and Cs) from the Radial and Axial Blankets as generated by RIBD.
- 4) 100% of Noble Gases from Axial and Radial Blankets as generated by RIBD.
- 5) ^{22}Na and ^{24}Na as calculated by hand.

Input to COMRADEX then determines which and how much of the above groups are involved in the specific accident scenario being considered. The COMRADEX Library is shown in Appendix A.

3. Single Containment Design (Case I)

The physical layout of the single containment reference plant is shown schematically in Figure 7. The Reactor Containment Building (RCB) is a steel shell with insulation on the outside. In this study, the RCB is defined as that enclosed part of the containment system in which personnel routinely work during the reactor operation. The open head compartment and reactor vessel head dimensions were based on the FFTF and CRBR systems, respectively.

The RCB has an assumed design leakrate of 0.1%/day at a 10 psi (69 kPa) pressure differential. The floor of the RCB is concrete and releases its water to the RCB atmosphere as it is heated. Detailed information on analysis assumptions is given in Appendix B.

HEDL 7506-127.8

FIGURE 7. Cross-Sectional View of Single Containment (Case I).

The reactor vessel and reactor cavity are shown schematically in Figure 8. This same cavity/vessel design is used for all of the major cases.

Equipment and other structures in the single containment system that affect thermal calculations are listed in Appendix B. Many of these structures are the same for the double containment and containment-confinement systems.

4. Double Containment Design (Case II)

The principal physical difference between single and double containment schemes is the addition of a sealed dome over the reactor head compartment. The outer containment vessel (RCB) was taken to be identical with that provided for the single containment case. The double containment scheme is shown schematically in vertical cross-section in Figure 9. The reactor vessel and reactor cavity are also identical to those used for the single containment case.

The forty-foot diameter primary containment barrier dome, the seal on the head compartment, is made of 1-inch thick steel. The atmosphere inside the dome is inerted. The design leak rate is assumed to be 100%/day at a 15 psi pressure differential (103 kPa). This dome is removable for access to the head and for refueling operations. Heat structures important to this case are identified in Appendix B.

HEDL 7706-119.6

FIGURE 8. Schematic Cross Section of Reactor Cavity.

HEDL 7506-127.9

FIGURE 9. Cross-Sectional View of Double Containment System (Case II).

5. Containment-Confinement Design (Case III)

The containment-confinement plant is essentially a double containment system, except that the RCB is designed for negative pressure operation.

It is a ventilated RCB, continually purged. Exhaust fans maintain a flow rate of approximately 63,000 pounds of air per hour ($\sqrt{8}$ kg/s) thereby maintaining the building at a slightly negative pressure. Under normal conditions, this is an exhaust rate of 15,000 CFM. During high temperature accident situations, this results in a maximum exhaust rate of approximately 40,000 CFM.

The exhaust system discharges air through cleanup devices to a stack so that radioactive materials leaked from the primary containment are processed by filters or scrubbers. The containment-confinement scheme is shown schematically in vertical cross section in Figure 10. Features of the containment system below the operating floor are the same as Cases I and II. All leakage from the primary containment barrier is to the confinement building.

The physical size and features of the primary containment volume were assumed to be identical to those of Case II.

FIGURE 10. Cross-Sectional View of Containment-Confinement System (Case III).

Several features of Case III distinctly different from features in Cases I and II include the following:

- All RCB leakage is inward and is collected and treated before being released to the atmosphere.
- Atmospheric discharge is via elevated release which reduces closein predicted doses.
- Air cleaning is an integral part of the containment design.

B. Accident Scenarios

Two hypothetical accidents were considered in this study: 1) the melt-through CDA; and 2) the combination melt-through and energetic CDA. A brief description of each is covered in this section.

1. Melt-Through CDA

The melt-through CDA is a postulated event in which sufficient heat transport for maintaining stable coolable geometries within the vessel is lost and progressive core melting and relocation eventually leads to penetration through both the reactor vessel and guard vessel.

For this study, it was conservatively assumed that the noble gases were released from the reactor head at the start of the accident. It was further assumed that this initial release did not result in pressurization of the chamber into which it was released.

A melt-through accident would occur only if insufficient core cooling occurs. Even then, melt-through is delayed by the various structures that must be penetrated. It was assumed for this study that the melt-through of the reactor guard vessel occurs at 1000 seconds. ANL small scale experiments⁽²¹⁾

indicate that the flow of molten fuel and sodium into the reactor cavity will result in the fragmentation of the fuel and the formation of a debris bed on the cavity floor. Due to expected vigorous mixing and self-leveling characteristics, the fragmented core is expected to form a coolable debris bed on the cavity floor. This bed will remain coolable as long as it is covered by sodium. Decay heat is removed by conduction to exposed structures and to the sodium pool. Two major facets of the accident scenario tend to significantly affect the containment performance: 1) sodium boil-off; and 2) moisture release and venting from the concrete. At the time of melt-through, the sodium temperature is in the range of 1000 to 1100°F (538 to 593°C). Thus a time lapse occurs between melt-through and the time at which sodium boiling occurs.

The reactor cavity is assumed to be lined with steel which protects the surrounding concrete from direct contact with the sodium and debris bed. Heat conduction from the pool into the concrete will heat the concrete to temperatures causing water vapor release from the concrete. The water vapor can be vented to various regions within the containment system including the RCB or equipment cells. If the water vapor reaches the RCB, it can add to the energy input causing pressurization. If sodium is present in the RCB, it can react with the water to add further to the energy input and to form hydrogen, which can potentially accumulate and exceed limiting concentrations.

For this study, reactor cavity floor liner failures have been assumed in the accident sequence to determine the effects of such a condition on the containment systems. If this were to occur, the moisture from the concrete could enter the sodium pool, react to form hydrogen which, in turn, could result in cavity pressurization and leakage into the next chamber.

Regardless of whether or not the liners fail, the sodium in the pool will reach boiling and continue to boil out of the reactor cavity, carrying with it the sodium-soluble and volatile fission products into subsequent chambers, ultimately reaching the RCB. In each chamber, the pressure will rise due to: 1) the heat content of water and sodium from the cavity; 2) chemical

reactions; and 3) fission product decay. If pressurization exceeds the design limit of a chamber, either failure relief or designed relief can occur and allow leakage into the next chamber. Such sequential pressurization ultimately can result in a source term release to the atmosphere. If a hydrogen limit ($\geq 4\%$ concentration) is reached in the RCB, action may be required to relieve further accumulation. Methods considered in this study to relieve pressurization and hydrogen buildup include venting, venting and purging, and continuous purging. Potential releases can be reduced in magnitude by inherent attenuation mechanisms such as agglomeration and settling in the containment chambers and by engineered cleanup systems attached to the RCB relief mechanisms. The consequences of the source term releases to atmosphere are covered in Chapter IV.

Once all the sodium has boiled off from the reactor cavity, the fuel debris bed will melt into the concrete. This study assumes that this occurrence adds no significant contribution to the source term.

2. Combined Accident

In this case, a postulated melt-through accident is assumed to be preceded by an energetic disassembly of the core. Both the early source term release from the energetic accident and the extended release of the melt-through accident must be considered together.

In order for an energetic hydrodynamic core disruptive accident to occur, certain physical processes such as core compaction must be postulated. A recent review⁽²²⁾ indicates that such physical processes relevant to energetic accident paths cannot be realistically attained. Two potential sources of excess energy which could lead to a hydrodynamic core disassembly have been considered in breeder reactor safety studies: 1) criticality with loss of shutdown capability for an intact core; and 2) recriticality during or following whole core disruption.

In the first instance, sufficient negative reactivity insertion mechanisms, for example, fuel dispersal and Doppler effects, will act to offset positive reactivity insertion mechanisms such as sodium voiding. In both instances, the conditions required for energetic thermal interaction between molten oxide fuel and sodium are not established. In the second instance, the absence of vapor explosions and the presence of fuel boilup prevent prompt critical excursions.⁽²²⁾

For the purpose of this study an energetic CDA must result in an instantaneous source term release through the reactor head at time zero. To achieve a near instantaneous source term release, appropriate conditions must be assumed which result in sufficient work energy to breach the head seals. When the head seals are breached, the core materials which will comprise the source term (see Chapter III, Section C) are rapidly expelled into the containment systems.

At 1000 seconds after the head expulsion, the melt-through portion of the accident is assumed to occur. The scenario from this point is described in Chapter III, Section B-1. Design and performance characteristics of the containment system will determine the variations in the scenario. These characteristics and their effects are covered in Chapter IV. In short, the energy release due to sodium reactions and radioisotopic decay will tend to pressurize containments which will, in turn, leak to the next chamber and ultimately release to the atmosphere. During the leakage period, inherent and engineered removal processes can affect the source magnitude. With the resulting source term, the accident consequences can be determined (Chapter IV).

C. Source Term Factors

Basic to the evaluation of radiological consequences is the evaluation of quantity and timing of release of the various radioactive isotopes. Since this is a comparison study, the actual fractional releases are not as important as the mechanisms and timing.

The reactor inventory at the start of the accident is discussed in Chapter III, Section A and listed in Appendix A. It is convenient to separate the elements into four groups:

- I - Noble Gases
- II - Volatile, Sodium-Soluble Elements
- III - Actinides
- IV - Solid Fission Products

The noble gas group includes the isotopes of krypton and xenon. In all cases, it is assumed that 100% of these isotopes are released through leaks in the reactor head at time zero. This is conservative, since many of these isotopes have very short halflives and any reasonable retention period in the vessel would significantly reduce the direct or external dose.

Group II, the volatile, sodium-soluble elements, include the halogens (bromine and iodine) along with arsenic, selenium, rubidium, cadmium, and cesium. For the purpose of this study, it was assumed that elements of this group are released with, and in proportion to, the sodium release. Various different release models have been proposed for these elements,⁽²³⁾ but it is believed that use of these models would not significantly change the comparative results of this study.

The third group consists of the actinides, or fuel-related elements. This group includes plutonium, americium and curium. It was assumed that the isotopes in this group are released only during an energetic disassembly accident, i.e., when it is conceivable that they will exist as an aerosol or in the volatilized state. For this study, release of these isotopes from the reactor vessel is assumed to be 1% of the core inventory.

Various models, release fractions, and release mechanisms have been studied by others, but such in-depth analyses are beyond the scope of this study.^(24,25,26)

All of the other elements listed in Appendix A are classified as solids. These elements, like the actinides, are assumed to be released only during the energetic disassembly accident. A release of 1% of the inventory was assumed.

The release fractions are shown in Table 5. There are two sets, one involving a melt-through accident only, and the other combining a head expulsion (energetic) term with the melt-through source term. In both types of accidents, it was assumed that 100% of the noble gases are released, and that all of the volatile elements are available for release. In the combined accident, the actinides and solids are assumed to be released in 1% fractions.

For completeness, the basic sodium inventory is also listed. One thousand pounds of sodium were assumed to be expelled during the head expulsion phase of the combined accident.

D. Major Design/Phenomenological Considerations

During the two general types of accidents covered in this report, a number of sequential events and phenomena are considered major potential influences on establishing the minimum containment performance requirements. In some cases, singular considerations form decision points that result in significantly different paths. The consequences of the decisions are discussed in Chapter IV. These factors can be divided into two categories: 1) design-related; and 2) phenomena-related. The following factors are a partial listing:

TABLE 5
RELEASE FACTORS FOR VARIOUS ISOTOPES

	MELT-THRU ONLY	COMBINED MELT-THRU/EXPULSION	
	WITH EARLY NOBLE GAS RELEASE	MELT-THRU PORTION	EXPULSION FROM HEAD
NOBLE GASES	100%	0%	100%
VOLATILE FISSION PRODUCTS	100%	90%	10%
SOLID FISSION PRODUCTS	0%	0%	1%
ACTINIDES	0%	0%	1%
SODIUM	2×10^6 lb	2×10^6 lb	1000 lb

HEDL 7706-119.21

Design

1. Pool coolers
2. Chamber vent or purge mode
3. Liner vent discharge path
4. Atmosphere cleanup systems
5. Cavity volume
6. Containment volume
7. Sodium inventory
8. Concrete specifications

Phenomena

1. Melt-through time
2. Liner integrity
3. Na/Concrete interaction
4. Other Na reactions
5. H₂ Recombination
6. Core melt into concrete

The effects of the above design considerations covered in this study are discussed in Chapter IV, Section C-2. The effects of phenomena-related factors are discussed in Chapter IV, Section C-3. Additional information considered important for interpretation of Chapter IV results is provided in this section.

1. Vent and/or Purge Mode

When a chamber reaches certain design limits, e.g., pressure, temperature, or hydrogen concentration, the chamber is allowed to relieve the limiting condition. Three representative modes have been considered: 1) rupture disc equivalent; 2) pressure relief valve equivalent; and 3) rupture disc and purge equivalent.

The first mode is equivalent to a self-relieving chamber rupture that cannot reseal itself. In this case, the chamber pressure is relieved until equilibrium is reached with the next chamber. Depending on the initial differential pressure across the chamber wall, considerable atmospheric content of the first chamber can be released rather quickly into the second chamber. In addition, the percent composition of the first chamber will remain much the same; therefore, if a hydrogen concentration problem exists, it will not be alleviated.

The second mode is equivalent to an engineered pressure control relief system. This mode obviously tends to reduce the potential for a sudden release of a large fraction of the chamber atmospheric content. It also allows a higher aerosol concentration over a longer period, thus enhancing source term attenuation due to agglomeration and settling, and due to radioactive decay. However, this mode does not alleviate a hydrogen concentration problem.

The third mode is equivalent to the first mode with the additional capability for introducing a purge gas into the containment chamber. This mode results in the rapid exhaust of the chamber atmospheric content. Its main advantage is that it can dilute and reduce the chamber's hydrogen concentration. If the purge gas is inert, no chemical reactions will be induced by the incoming gas. If, however, the purge gas is air, considerable chemical reaction between sodium and air can occur if sodium is present. This exothermic reaction will lead to higher temperatures in the chamber. The resulting consequence of this type of occurrence is discussed in Chapter IV.

2. Liner Vent Discharge Path

A design consideration which can significantly affect the containment requirements is the location in which the water released from the reactor cavity concrete is accumulated. If, for instance, the water is vented into the annular volumes surrounding the cavity, a significant delay will occur in the time at which this moisture enters into the Reactor Containment Building (RCB). As a result, the contribution of the water heat capacity and its exothermic reaction with sodium to RCB pressurization will be delayed. In addition, a delay can occur in the time and rate of potential hydrogen production in the RCB. The significance of cavity liner discharge path is discussed in Chapter IV.

3. Liner Integrity

The reactor cavity is assumed to be lined with a low-carbon steel plating. In the ERDA Fast Reactor Base Safety Technology Program, the integrity of the reactor cavity liners is being demonstrated under conditions which may be postulated during the melt-through type accident. Assessments to date on the FFTF and CRBR have included both failed and unfailed liner cases.

If the liner does not fail, the sodium and the core debris bed will be separated from the concrete and no interactions will occur which would add energy or affect the system's structural integrity. The liners will likely bow from the cavity wall enough to allow concrete moisture venting. The sodium pool will heat to boiling and will increase the pressure in the cavity until either an equilibrium inherent leakage occurs or until an engineered pressure relief occurs. The effluent will consist of sodium vapor, volatile fission products, residual noble gases and inerting gases.

If the liner is assumed to fail, chemical reactions can occur between the sodium and the concrete. In this study, an area equivalent to the cavity floor area, about 1260 ft² (117 m²) is assumed to be exposed for reaction with the sodium. Based on tests conducted at HEDL⁽²⁷⁾, the reaction is assumed to be self-limiting and to stop after penetrating into the concrete to a depth of 2.0 in. (5.68 cm). The exothermic chemical reaction energy is added to the system thermal input. The hydrogen produced by the sodium/water reaction pressurizes the cavity and flows, along with any other cavity atmosphere constituents, as a function of differential pressure, into the next chamber.

4. Hydrogen Formation and Control

There are two general modes for producing hydrogen during the melt-through type accident. In the first mode, the cavity liner is assumed to fail. As stated in Chapter III, Section D-3, the sodium reacts with the concrete water

to form hydrogen which then can leak to the next chamber. In the second mode, the liner is assumed to remain intact. The potential for significant hydrogen production does not exist until the sodium reaches its boiling temperature and the vapor begins to leak into subsequent chambers. As the sodium enters a chamber with a high humidity, hydrogen is produced as a product of the sodium/water reaction. This condition can exist for sodium entering the RCB if moisture from the cavity concrete has reached the RCB or if the RCB concrete floor has been heated sufficiently to release significant amounts of water.

As hydrogen is formed, it accumulates until potentially flammable mixture concentrations are reached. A hydrogen limit of 4% by volume has been issued as a guide for reactor facilities⁽²⁸⁾ (6% hydrogen if the energy from burning 2% of the hydrogen is included in the system energy input). During the postulated breeder reactor melt-through accident, the potential has been shown, and is being experimentally verified, that the hydrogen produced under either of the above modes will inherently recombine to form inert products.⁽²⁹⁾ This experimental program has resulted in the following preliminary criteria which were used in this study when hydrogen recombination was assumed:

- 1) Ignition of a hydrogen-nitrogen-sodium release always occurs if the sodium concentration is greater than 6 g/m^3 and the jet temperature is at least 500°F (260°C). (Applicable to the first mode).
- 2) Hydrogen burning ceases when the oxygen volumetric percent decreases to 10%.
- 3) A sodium/nitrogen jet will not form hydrogen unless the oxygen concentration is less than the water vapor concentration. (Applicable to the second mode).

Once these certain conditions are reached, hydrogen can be formed and accumulate. In this study, when the hydrogen concentration reaches 4% by volume in an atmosphere with oxygen greater than 5% by volume, the chamber is either vented or vented and purged.

The majority of cases that were studied assumed that all cavity liners remained intact. For these cases, the formation criteria, 3) above, was not used. Instead, hydrogen was allowed to form whenever sodium and water were together. When oxygen was present, the sodium was shared in proportion to the molar ratio of water and oxygen. Hand calculations on the CACECO output verified that, for these cases, both models gave the same results. This is further discussed in Chapter IV, Section C-36.

5. Melt-Through Time

Since the inertial delay for sodium heating is primarily dependent on the thermal capacity of the sodium and surrounding structures, it is apparent that the melt-through time is not particularly significant as long as the sodium is exposed to the cavity cooling structures prior to sodium boiling. This has been shown for the CRBR size reactor^(30,31) for melt-through times up to five hours. Thus, if melt-through of the reactor and guard vessels occurs prior to sodium boiling, no significant effect on the accident scenario is expected. In this study, a melt-through time of 1000 seconds was assumed.

IV. DISCUSSION OF CALCULATIONS

A. Matrix of Cases

The variation and changes in design and accident assumptions were discussed in Chapter III, Section D. A schematic representation of the various design options is shown in Figure 11. The design choices involve: 1) ultimate destination of cavity concrete water; 2) use of a cavity pool cooler; 3) sealed or open head access compartment; 4) basic RCB specifications; 5) cleanup and relief mode for the RCB; and 6) cleanup of the final releases from the RCB. A schematic representation of the allowable variations in accident considerations is shown in Figure 12. Here the choices involve: 1) the type of accident; 2) the condition of the floor liner; and 3) hydrogen recombination assumptions.

Comparison of the large number of cases required the setup of a symbolic representation scheme. The definition of the symbols used are shown in Figure 13. The first symbol, a Roman numeral, indicates the basic containment type, i.e., "Single Containment," "Double Containment," or "Containment-Confinement." The second symbol, a number, indicates the major design variation or accident assumption being examined. In some cases, a third symbol further describes the case.

The example shown in Figure 13 is for Case I-4(1). Case I-4(1) is basically a high-pressure, low leakage RCB in which the head access area is not sealed (single containment). The system is modeled to show the effects of initial expulsion of radioactive materials from the head of the reactor. The case is further qualified by venting the water from the cavity concrete directly into the RCB.

A listing of all of the cases studied, the design and accident-related features of each case, and the descriptive symbols for each case is shown in matrix form in Tables 6 and 7. The cases listed with an asterisk(*)

FIGURE 11. Design Options for Containment Systems.

HEDL 7706-119.5

HEDL 7706-119.4

FIGURE 12. Variation in Accident Considerations.

(Example) CASE I-4 (1)

CONTAINMENT TYPE

DESIGN/ACCIDENT FACTORS

I = "Single Containment" Head Access Area is not sealed; RCB is high-pressure, low-leakage type.	1 -- Water from cavity concrete is vented to RCB.
II = "Double Containment" -- Head Access Area has high-pressure seal; RCB is high-pressure low-leakage type.	2 -- Water from cavity concrete is vented to "drain".
III = "Containment-Confinement" -- Head Access Area has high-pressure seal; RCB is low-pressure, high-leakage type.	3 -- 100% cavity floor liner failure at time 0
	4 -- Melt-through is preceded by an expulsion of radioactive materials (including 1% Pu) from the head of the reactor.
	5 -- Cavity <u>pool</u> cooler available after melt-through.
a. 2.5 MW
b. 8 MW
c. 10 MW
	6 -- Purge as necessary to control H ₂ concentration.

FIGURE 13. Definitions of Containment Symbols.

were studied using only the CACECO code. All other cases followed the CACECO-HAA-COMRADEX calculational sequence described in Section B. Additional assumptions and details common to all cases are presented in the Appendixes. An event tree of the study cases was previously presented in Figures 1, 2, 3, and 4 of Chapter II.

B. Calculational Techniques

The basic objective of this study is to compare containment concepts on the basis of radiological consequences following postulated accidents. To accomplish this assessment, extensive use was made of the CACECO,⁽³²⁾ HAA,⁽³³⁾ and COMRADEX⁽³⁴⁾ computer codes. CACECO predicted the pressure and temperature transients and leak rates during the accident. HAA, using CACECO output, determined the fallout rates of the aerosol within the containments. COMRADEX then was used to calculate the radiological consequences of the accident for each containment system. A discussion of each of the codes and how they were used for this study is presented in the following sections.

1. CACECO Code

The CACECO code⁽³²⁾ (Cavity, Cells, and Containment analysis) is designed to provide a complete pressure-temperature-leakrate history of breeder reactor containment system following a hypothetical core disruptive accident. It calculates this history from mass and energy balances that include leakage and chemical reactions. The chemical reactions include the following:

In the reactor cavity:

Sodium + oxygen to form sodium oxide

Sodium + concrete

Sodium + water vapor to form hydrogen

Sodium + hydrogen to form sodium hydride

In the RCB:

Sodium oxide + water vapor to form sodium hydroxide

Sodium + oxygen to form sodium oxide

Sodium + water vapor to form hydrogen

Hydrogen + oxygen to form water vapor (hydrogen recombination).

Reaction rates, energies and other assumptions are discussed in detail in Appendix B.

The code calculates temperature transients for equipment, roofs, walls, and floor structures which act as heat sources and sinks in the analysis. These are the "heat structures" in CACECO. Consequently, the thermal properties and boundary conditions for these structures were major considerations in the setup of each containment problem.

Basically, the same set of heat structures was used for all cases. The only differences came about when, for example, an inner containment dome added an additional thermal structure, or the flat roof of the rectangular containment-confinement RCB presented different thermal properties. A listing of all the heat structures and their characteristics and boundary conditions is included in Appendix B.

For the basic melt-through accident, CACECO spilled 1075°F (835 K) sodium into the cavity at a 2.4×10^7 pound/hour (3.03×10^3 kg/s) rate. This rate dumped all two million pounds (9.1×10^5 kg) of the primary sodium in 5 minutes. All the fission product decay heat of the reactor is assumed to be released to the sodium pool. CACECO then proceeded to follow the mass and energy transfers throughout the various allowable paths in the containment systems.

The head expulsion cases were also handled by CACECO. In these cases it was assumed that 1000 pounds (454 kg) of sodium were expelled instantly from the reactor into the atmosphere of the next chamber. For single-containment cases, this next chamber was the RCB. For other cases, it was the inner

TABLE 6
CASE MATRIX FOR SINGLE CONTAINMENT SYSTEMS

	STUDY VARIATION					RELIEF MODE			
CASE	LINERS VENT TC RCB	LINERS VENT TO DRAIN	FLOOR LINER FAILS	INITIAL HEAD EXPULSION	POOL COOLER (MW)	VENT	PRESSURE RELIEF VALVE	VENT & PURGE	FILTER RELEASE
I 1	X					X			
I 2		X					X		
I 3 (2)		X	X				X		
I 4 (1)	X			X		X			
I 4 (2)		X		X			X		
I 5a (2)		X			X (2.5)		X		
I 5c (2)		X			X (10)		X		
I 6 (1)	X							X	
I 6 (2)		X						X	
I-TC ⁺		X					X		
I-HP ⁺⁺			X				X		

* CACECO - ONLY CASES

** 1/2 OF ALL LINERS FAIL AT 12 HRS

+ HIGH THERMAL CONDUCTIVITY CONCRETE IN CAVITY

++ VERY HIGH PRESSURE (50 PSIG)RCB

HEDL 7708-8.8

TABLE 7

CASE MATRIX FOR DOUBLE CONTAINMENT AND CONTAINMENT-CONFINEMENT SYSTEMS

	STUDY VARIATION					RELIEF MODE			
CASE	LINERS VENT TO RCB	LINERS VENT TO DRAIN	FLOOR LINER FAILS	INITIAL HEAT EXPULSION	POOL COOLER (MW)	VENT	PRESSURE RELIEF VALVE	VENT & PURGE	FILTER RELEASE
II 1	X					X			
II 2		X					X		
II 3 (2)		X	X				X		
II 4 (2)		X		X			X		
II 5c (1)	X				X (10)		X		
II 6 (1)	X							X	
II 5a (1)	X				X (2.5)		X		
II 3 (5c)		X	X		X (10)		X		
III 1	X							↑ CONSTANT PURGE ↓	X
III 3 (1)	X		X						X
III 4 (1)	X			X					X
III 5b (1)	X				X (8)				X
*0 III-POOL		X							X
* CACECO - ONLY CASES + INCREASED CAVITY CONCRETE CONDUCTIVITY CASE 0 POOL REACTOR									

HEDL 7708-8.9

containment volume of the sealed head access compartment. This treatment of the expelled sodium was very conservative and resulted in much higher pressure than the time-dependent SPRAY⁽³⁵⁾ code would have predicted.

CACECO, as it now exists, requires operator intervention repeatedly in the early phases of the accident. The first interruption occurred for each case at eight hours. At this time, temperatures and pressures in the reactor cavity were high enough to require relief as might be naturally provided by component failure (head seal, etc.) or by design (rupture disc or pressure relief valve). This condition was relieved in CACECO by adding a nozzle from the cavity to the next chamber. The nozzle was sized to substantially reduce the pressure difference, but not so large as to produce instabilities in CACECO. Other user interventions were required when the hydrogen concentrations or pressure in the next chambers became excessive. A more complete description of the nozzle sizes and characteristics of pressure relief valves is included in Appendix B.

Once CACECO had produced a complete pressure, temperature and leak rate history for each chamber, selected output was fed to HAA for calculation of aerosol depletion.

2. HAA-3B Code

The HAA-3B Code⁽³³⁾ was used to determine the overall removal rate of aerosol from the various containment atmospheres. The mechanisms for removal include Brownian and gravitational agglomeration, settling, wall plating, and leakage. The introduction of the aerosol was calculated from CACECO data and was assumed to have a log-normal distribution. The leak rate fraction was also a CACECO result.

The calculation of the aerosol removal rate is important because it is assumed that isotopes released to the various containments behave similarly to the aerosol. The same removal rates are eventually applied to these isotopes. Detailed input data and assumptions used in the HAA-3B code are presented in Appendix C.

Sodium vapor leakage from the reactor vessel or cavity into the RCB is assumed to react into sodium oxide as rapidly as it is released when oxygen is present. Almost all of the work on verification of the HAA code has been conducted using sodium oxide as the aerosol material. Hence, the assumptions for distribution, shape, size, etc., have been verified. The handling of the sodium oxide aerosol input source is straightforward; use the CACECO rate data, a "standard" set of constants, and containment data, such as area, volume and temperature, to calculate the removal rates. The only real problem occurs in those cases where there is no more oxygen in the containment atmosphere to react with incoming sodium. In these cases, the sodium oxide aerosol source, at some time, becomes a sodium aerosol source. The ideal way to handle this problem would be to change the values of the data describing the aerosol characteristics to those of sodium. The problem with this approach is that the code has not been verified for this use and the necessary data for the sodium aerosol do not exist. A solution to this problem is to "turn off" the aerosol input when the oxygen is depleted. At this time, HAA will continue to calculate fallout due to plateout and settling.

The method used for this study to account for the aerosol at the time oxygen is depleted is to continue inputting the sodium as if it had sufficient oxygen with which to react. All constants, shape factors, etc., were left the same as if the incoming aerosol were actually sodium oxide. This assumption is probably conservative since fallout depletion is expected from product aerosols of the sodium/water reaction and condensation of sodium vapor.

3. COMRADEX Code

The COMRADEX⁽³⁴⁾ computer code was used to calculate the release of radioactive materials from the Reactor Containment Building (RCB) and the potential exposures at various distances from the plant. The inventory of isotopes at the time of the accident must be provided, as well as the leakage and overall fallout rates for the various containments. The isotope inventory is described in Chapter III, Section A and Appendix A. This is the result of

RIBD⁽¹⁹⁾ calculations at end-of-equilibrium cycle conditions. The effects of containment, the leakages and fallouts, are determined previously by CACECO and HAA.

COMRADEX uses this information to calculate: 1) the potential direct gamma dose from the building; 2) the potential internal radiological dose resulting from inhalation of activity in the released cloud; and 3) the potential external gamma dose from the released cloud.

The direct and external gamma doses are assumed to occur only during the accident. However, all internal doses (bone, lung, thyroid, whole body) are calculated on the basis of a 50-year commitment period. These calculations require the use of "F"-factors for the isotopes which are the Rem-per-Curie-Inhaled multipliers for each isotope. In determining these numbers, account is taken of the radiological and biological halflives of the particular isotope, the energy deposition per disintegration for each isotope, and the 50-year commitment period.

The "F"-factors for this study were based entirely on the ICRP-II⁽³⁶⁾ model ("old lung"). Values of the "F"-factors for the more important organs are listed for reference in Appendix A.

The COMRADEX modeling of the boilup phase considered the reactor vessel/reactor cavity as the first chamber of a two-chamber calculation. At the onset, 90% of the volatiles and 2,000,000 lb (9.08×10^5 kg) of sodium were assigned to that chamber. The volatile elements are those whose probable chemical form have a substantial vapor fraction at the boiling temperature of sodium; specifically, As, Se, Br, Cd, Rb, I and Cs. These were then available to "leak" from Chamber 1 to Chamber 2 (the containment building). The leak-rate of Chamber 1 was the sodium boil-off rate determined by CACECO and smoothed by hand. All volatiles were assumed to boil off at a rate proportionate to that of the sodium. There is no fallout in the initial chamber; any refluxing had already been accounted for by CACECO.

Cases that involved an inner containment (a dome over the head access compartment) treated the boilup phase the same way; the leakage fraction is calculated by CACECO (Cavity-to-RCB) and any refluxing inside the dome is accounted for by CACECO.

The COMRADEX modeling of the head expulsion phase was handled as a single chamber run for the single containment cases. The initial puff contained 100% of the noble gases, and other isotopes described in Chapter III, Section C. The leakrates were calculated by CACECO, and the fallout was calculated by HAA. COMRADEX does not allow the noble gases to fallout.

Cases that involved an inner containment needed special treatment. The initial expulsion releases all of the noble gases and the other isotopes, as described in Chapter III, Section C, into the inner domed area. COMRADEX is then used as a two-chamber case, where the first chamber is the domed area, and the second chamber is the RCB. Leakrate fractions from the dome to the RCB were calculated by hand from CACECO output, and fallout inside the dome was calculated by HAA.

The total doses due to both the boilup and head expulsion phases were obtained by adding together two COMRADEX runs.

The cases that involved only a melt-through were also modeled with a boilup phase and a head expulsion phase. The boilup phase included 100% of the volatiles, handled as previously discussed. The expulsion phases released 100% of the noble gases from the cavity into the next chamber at the start of the accident; however, no other radioisotopes accompanied this release. Leakage into the next chamber (or outside, for single-containment cases) was determined by leak rates established by the boilup phase. That is, although the noble gases could provide a direct dose, they could not leak to the next chamber until the boilup phase provided the driving mechanism.

The meteorology used for COMRADEX closely duplicated that of Regulatory Guide 1.4⁽³⁷⁾ The standard Pasquill classes were used and the wind speed adjusted so that X/Q values would approximate those of the Guide.

C. Specific Comparisons

This section deals with the physical effects of the CDA's on the various containment systems. Pressure, temperature, hydrogen concentration, and leak rates are examined and compared for the different accident and containment assumptions. First, the effects of the melt-through CDA on the base containment cases are examined. This is followed by sections dealing with the effects due to design and accident variations. The final section examines special cases where attempts were made to provide inherent accident slow-down features in the system design.

1. Basic Containment Systems; Cases I-2, II-2 and III-1

The base case for each of the major containment systems is defined as the case that involves only a melt-through CDA, all cavity liners are vented to condenser/drain rooms, and all liners remain intact. The only exception to this is the containment-confinement base case (III-1), where all of the water from cavity concrete is vented to the RCB. In this case, the location of the water vent was not a significant variable.

A chronology of accident effects for each containment system base case is given in Tables 8, 9, and 10. Since the reactor and cavity design was essentially the same for each of the cases, the accident effects are about the same up to seven or eight hours. At this time, however, enough fission product decay heat has been added to the cavity pool that its temperature has reached 1620 °F (1160 K), the normal boiling temperature of sodium. The reactor head leak rate increases and significant quantities of sodium leak into the next chamber.

TABLE 8

CHRONOLOGY OF ACCIDENT EFFECTS FOR SINGLE CONTAINMENT BASE CASE I-2

<u>Time</u>	<u>Event</u>
<u>0</u>	<ul style="list-style-type: none"> • The initiating event occurs at this time.
<u>0 to 1000 Seconds (17 minutes)</u>	<ul style="list-style-type: none"> • Sodium and structures in reactor vessel heat to 1075°F.
<u>1000 Seconds (17 minutes)</u>	<ul style="list-style-type: none"> • Entire core inventory melts through reactor vessel and spills, with sodium, on to lined cavity floor. • Pressure spike causes head seal to open slightly. • Noble gases enter RCB.
<u>1000 Seconds to 7 hours</u>	<ul style="list-style-type: none"> • Decay power heats sodium pool and cavity structures. • Gases, primarily nitrogen, in the cavity heat up, expand, and leak to the RCB atmosphere. • Sodium vapor begins to leak to the RCB and react with oxygen. • Water released by concrete walls in the cavity is vented to a condensing room where it leaves the scenario. • RCB begins to pressurize.
<u>7 to 8 Hours</u>	<ul style="list-style-type: none"> • Normal boiling temperature of sodium is reached. Pressurization of RCB and cavity temporarily prevent boiling. • High temperature cause further failures of head seals.
<u>8 to 12 Hours</u>	<ul style="list-style-type: none"> • Sodium boiling starts under pressurized conditions. • Sodium in-rate to the RCB continually increases.

TABLE 8 (Cont'd)

- Pressure in RCB increases.
- Temperature in RCB atmosphere increases, due mainly to the sodium/oxygen reaction.

12 to 14 Hours

- 10 psig RCB pressure relief valve begins operation.
- All oxygen used up by incoming sodium.
- Water starts entering RCB atmosphere from RCB concrete floor.
- Hot sodium vapor dominates energy input rate to RCB.

24 Hours

- 10 psig pressure relief valve still operating.
- Sodium-water reaction accounts for 35 % of energy entering RCB (about one-third).

148 Hours (6.2 days)

- Sodium boiling in cavity continues until pool boils dry at 148 hours.
- RCB atmosphere is essentially all hydrogen at 10 psig, 600°F.

TABLE 9

CHRONOLOGY OF ACCIDENT EFFECTS FOR DOUBLE CONTAINMENT--BASE CASE II-2

<u>Time</u>	<u>Event</u>
<u>0 to 1000 Seconds</u>	<ul style="list-style-type: none">• Scenario is the same as that for the single containment case I-2.
<u>1000 Seconds to 7 Hours</u>	<ul style="list-style-type: none">• Following melt-through, decay heat continues to warm up sodium pool and cavity structures.• Sodium vapor begins to enter head access area (sealed with dome for all "II" cases).• Sodium reacts with oxygen (initially 2%) in head access area.• RCB atmosphere temperature and pressure are relatively unaffected.
<u>7 to 8 Hours</u>	<ul style="list-style-type: none">• Normal boiling temperature of sodium is reached.• Pressurization of head access area prevents boiling.• High temperatures cause further failures of head seals.
<u>8 to 12 Hours</u>	<ul style="list-style-type: none">• Sodium in-rate to head access area increases.• Pressure and temperature increase in head access area.
<u>12 Hours</u>	<ul style="list-style-type: none">• Head access area vents to RCB atmosphere due to high temperature and pressure.
<u>12 to 13 Hours</u>	<ul style="list-style-type: none">• Increased differential pressures cause high inter-chamber flow rates.• Sodium/oxygen reaction in RCB heats and pressurizes atmosphere.• Sodium condenses and forms pool in head access area.

TABLE 9 (Cont'd)

13.5 Hours

- 10-psig RCB pressure relief valve begins operation.

13.4 to 18 Hours

- Oxygen is depleted by sodium/oxygen reaction in RCB.
- RCB concrete floor begins to heat up and release water to atmosphere.
- Hydrogen is produced by sodium/water reaction.

148 Hours (6.2 days)

- Sodium pool in cavity boils dry.
- RCB atmosphere is essentially 100% hydrogen at 10 psig, 600°F.

TABLE 10

CHRONOLOGY OF ACCIDENT EFFECTS FOR CONTAINMENT-CONFINEMENT BASE CASE III-1

<u>Time</u>	<u>Event</u>
<u>0 to 12 Hours</u>	<ul style="list-style-type: none"> • Scenario is the same as II-2, except for increased water in RCB (liners vented to RCB). • Liners venting water to RCB start increase in atmosphere temperature at 10 hours. • RCB pressure remains at -0.05 psig (constant forced venting of RCB).
<u>12 Hours</u>	<ul style="list-style-type: none"> • Head access area vents to RCB atmosphere due to high temperature and pressure. • Hydrogen formation increases, but concentration held below 6% by constant purge of confinement-type building. • RCB atmosphere temperature increases due to sodium/oxygen reaction.
<u>12 to 24 Hours</u>	<ul style="list-style-type: none"> • RCB atmosphere temperature increases to 1400°F due to continuing addition of sodium and oxygen.
<u>140 Hours</u> (5.8 days)	<ul style="list-style-type: none"> • Sodium pool in cavity boils dry.

In the single containment case, I-2, the next chamber is the RCB. So, as could be expected, the sequence of events following are more rapid than for the other cases (II-2 and III-1) that have intermediate chambers.

a. Base Cases: Energy Sources in the RCB

The easiest way to understand the controlling mechanisms/reactions of CDA's on the large sodium-cooled breeder reactors is to examine the sources of energy in the RCB atmosphere for each base case. The major inputs of energy to the RCB involve sodium. As detailed in Appendix C, sodium can add energy by: 1) carrying it directly from the cavity by vapor transport; 2) reacting with oxygen; and 3) reacting with water.

Water vapor from heated concrete can also add significant rates of energy to the RCB atmosphere. The three mechanisms for energy input by water are: 1) initially, since the sodium/oxygen reaction is consuming all of the sodium, excess water vapor reacts with sodium oxide to form sodium hydroxide; 2) as the oxygen is being consumed, the ratio of oxygen to water decreases, and the sodium/water reaction becomes significant. This reaction is also the mechanism for producing hydrogen in cases involving intact cavity liners; 3) when the concrete floor in the building begins to heat up, the hot water vapor is released to the RCB atmosphere.

Figure 14 shows a plot of the various energy inputs to the RCB for the single containment base case (I-2). Energy input is dominated in the beginning by the reaction of sodium with oxygen in the RCB. As soon as the oxygen is used up, the dominant source is the energy carried by the hot sodium vapor and other gases from the cavity to the RCB.

The double containment case (II-2) energy input rates are shown in Figure 15. The effect of the inner containment (a dome over the head access area) is to delay the high energy input rates into the RCB for about four hours. However, since the sodium input rate at this time is very high, the reactions and energy inputs occur at a higher rate. The end result is that Figure 15 looks like a compressed version of Figure 14.

FIGURE 14. Energy Input Rates to the RCB for Case I-2.

FIGURE 15. Energy Input Rates to the RCB for Case II-2.

The containment-confinement base case (III-1) exhibits some different characteristics, as shown on Figure 16. The most significant is the continuation of the sodium/oxygen reaction. The continued reaction is supported by the inflowing air used to purge the RCB atmosphere. Beyond sixteen hours, this significantly increases the energy input rate for this case.

Energy input from the hot water vapor flowing from the cavity concrete occurs early if the liners are vented to the RCB. This effect is shown graphically on Figure 16 for Case III-1.

b. Base Cases: Pressures and Temperatures

The primary effect of the energy inputs to the RCB is to increase the pressure and temperature of the atmosphere. Each case has its own characteristic pressure/temperature history based on the heat structures, leakages, and design of the particular system.

Figure 17 shows the RCB atmosphere pressure history up to 40 hours in each of the base cases. The effect of the buffer volume formed by the sealed head access area of Case II is to delay the pressure rise 4 to 5 hours. The dome seal over the head access area is assumed to fail when the atmosphere inside is above 1500°F (1089 K) and 10 psig (a differential pressure of 69kPa). Once this seal fails, Base Cases I and II become very similar.

The third Base Case, III-1, is a containment-confinement system where the RCB is held at a constant negative pressure by continuously removing air from the building, the make-up air being supplied by leaks in the RCB. The 15,000 SCFM exhaust rate is sufficient to prevent pressurization of the RCB.

Temperatures inside the RCB are dependent on energy removal as well as energy input. The base single and double containment cases have similar input and removal energy rates, so, as shown in Figures 18 and 19, the temperatures inside the RCB are similar for the duration of the accident. The containment-confinement case was modeled a little differently; since the RCB is

FIGURE 16. Energy Input Rates to the RCB for Case III-1.

FIGURE 17. RCB Pressures for the Base Cases.

FIGURE 18. RCB Atmosphere and Surface Temperatures--Case I-2.

FIGURE 19. RCB Atmosphere and Surface Temperatures--Case II-2.

FIGURE 20. RCB Atmosphere and Surface Temperatures--Case III-1.

a rectangular, industrial-type building, it has a flat roof, insulated with roofing materials. This insulation is not allowed to "sluff-off" as is assumed in the domed RCB cases. The result of this insulation is to increase the inside temperature of the roof for this case (Figure 20). The higher-than-normal wall and atmosphere temperatures are due more to the increased heat input of the sodium/oxygen reaction (Figure 15).

The atmosphere temperature for the containment-confinement case rises earlier than that for the double containment case for two reasons: first, a significant amount of hot water vapor has been vented to this atmosphere from the cavity concrete. Cases I-2 and II-2 vent this water to a condenser-drain room. Second, the constant low pressure in the RCB for the containment-confinement case promotes higher sodium leakage when the head access area pressure begins to rise.

C. Base Cases: Hydrogen Concentrations

The formation of hydrogen in the RCB atmosphere for all base cases is caused by the reaction of sodium vapor or aerosol leaking from the cavity, with water vapor in the RCB. The effect of hydrogen due to liner failure in the cavity is discussed separately in Section C-3a.

The hydrogen concentration in the RCB of the base cases is shown as a function of time in Figure 21. The containment-confinement case (III-1) tends to level off due to the constant purging of the building. If the cavity concrete water were dumped instead to a condenser/drain room, the curve would be even lower for the III-1 case. The additional curve for Case III-1* was calculated using the CACECO code that allows hydrogen recombination.

The hydrogen and oxygen concentrations in the RCB are shown for each base case in Figures 22 through 24. It is obvious from the first two figures (Cases I-2 and II-2) that hydrogen recombination assumptions will not affect the outcome because of the very rapid decrease of oxygen in the RCB. Both Cases I-2 and

FIGURE 21. RCB Hydrogen Concentrations.

FIGURE 22. RCB Hydrogen and Oxygen Concentrations--Case I-2.

FIGURE 23. RCB Hydrogen and Oxygen Concentrations--Case II-2.

FIGURE 24. RCB Hydrogen and Oxygen Concentrations--Case III-1.

II-2 were calculated using the CACECO code version that allows hydrogen formation whenever water and sodium are together. When water and oxygen are present, the sodium is reacted with each in proportion to its molar concentration.

The hydrogen concentration for Case III-1 (Figure 24) was calculated using the two hydrogen recombination models described in Section C-3b. In these cases, the hydrogen is formed due to an excess of water in the presence of sodium and oxygen. If the cavity-concrete water were dumped to a condenser/drain room, the hydrogen would be formed to a lesser extent by water released only from the RCB concrete floor.

d. Base Cases: Aerosol In Rate to the RCB

The rate at which sodium is entering the RCB is shown for each of the base cases in Figure 25. The delaying effect of the dome over the head access area is apparent. The input rate for the cases with an inner containment lags the single containment case by only six hours at the 100 lb/h (1.26×10^{-2} kg/s) input level and by less than four hours at the 1000 lb/h (.126 kg/s) input level.

The high sodium input rates force the concentration of aerosols suspended in the RCB to be nearly the same for the base cases. Figure 26 shows the suspended concentration as a function of time for the three base cases. Case III-1 is lower than the others because it is undergoing a forced exhaust.

e. Base Cases: RCB Leakage

The rate at which leakage is occurring from the RCB is shown for each of the base cases in Figure 27. An obvious difference can be seen between the high pressure-low leakage cases I and II and the low pressure-high leakage Case III. As designed, air is constantly being purged through the RCB of Case III-1; the increase in flow at ten hours for this case is a result

FIGURE 25. Sodium In Rate to RCB for Base Cases.

FIGURE 26. Suspended Concentration of Aerosol in RCB for Base Cases.

HEDL 7706-119.17

FIGURE 27. Leak Rates from the RCB for Base Cases.

of the decrease in density of the RCB atmosphere. The exhaust mechanism for this case was modeled to give a nearly constant mass flow rate. The exact model is described in Appendix B.

The other cases, I-2 and II-2, leak primarily through a pressure relief valve that limits the pressure in the RCB to approximately 10 psid. (a pressure differential of 69 kPa).

2. Design Assumption Effects

a. Liner Vent System Effects

Disposition of the water released by concrete in the cavity area has several effects on the accident. These effects are best illustrated by making a comparison of Cases I-2 and I-1. Case I-2 vents all cavity concrete water to condenser/drain rooms and water reaching these rooms no longer affects the accident. Case I-1 vents all cavity concrete water directly to the RCB atmosphere. The direct comparison is complicated by the fact that both cases also have concrete floors in the RCB that release water vapor when heated and this floor water tends to mask the effects of the cavity water late in the accident.

The cavity walls and floor release large amounts of water early in the accident. The amounts released to the RCB are shown in Figure 28. For Case I-2, the releases of water begin at a later time (11 hours) and involve only water from the concrete of the RCB floor.

The early entrance of the water to the RCB in Case I-1 has two significant effects; first, the RCB experiences an early pressure rise, as shown in Figure 29, and second, the increased quantity of water in the RCB reacts with sodium and this results in a hydrogen concentration in the RCB that exceeds 4% three hours sooner than for Case I-2. Also, although oxygen in the RCB is being depleted, the oxygen concentration in the RCB is about 13% at this time. Figure 30 shows the hydrogen, oxygen and water vapor concentration for Case I-1.

FIGURE 28. Total Amount of Water Entering RCB.

FIGURE 29. RCB Pressure vs Time.

The limiting mixture of hydrogen and oxygen that occurs for Case I-1 at approximately eleven hours is handled by venting the RCB down to ambient pressure with a nozzle large enough to prevent subsequent repressurization. This venting can be viewed as a designed relief mechanism or as a failure of the RCB containment that has the same effect.

b. Cavity Pool Cooler Effects

The primary driving force that sustains the propagation of accident conditions is the decay energy of fission products and activation products. Since most of this energy is released into the cavity sodium pool, the progression of the accident can be reduced or stopped by removing energy from the pool itself, i.e., before the energy boils the sodium into the RCB.

A partial power balance in the cavity sodium pool is shown in Figure 31 for the single containment Case I-2; it is typical for most of the cases studied. Power input to the pool is entirely from radioactive decay and is shown in the upper curve. Losses of energy by conduction and convection from the pool are shown in the lower curve. The difference between these two curves, the "excess" power, is available to heat and boil the sodium pool. This "excess" power is carried into the RCB by the sodium vapor, which can generate additional energy by reacting with oxygen and water vapor.

Depending on its size, the addition of a pool cooler in the cavity has the following important effects:

- 1) The time for the sodium pool to start boiling is delayed. In some cases, boiling does not occur at all.
- 2) When and if boiling does occur, it is at a lower rate.
- 3) Even a relatively small cooler (2.5 MW) can prevent the sodium pool from boiling dry.

FIGURE 30. H_2 , O_2 and H_2O in RCB Atmosphere--Case I-1.

FIGURE 31. Energy Balance in Cavity Pool--Case I-2.

Some of these effects are shown in subsequent curves. Figure 32 shows the pool temperature as a function of time for various size pool coolers. The 2.5-MW cooler case has 430,000 pounds (1.95×10^5 kg) of sodium remaining at the time boiling stops. The original inventory was 2,000,000 pounds (9.08×10^5 kg). The amounts of sodium eventually boiled off into the next chamber is shown as a function of pool cooler size in Figure 33. This figure indicates that a cooler between 8 and 10 MW is required if no boiling is permitted.

Figure 34 indicates the effect that various sizes of cavity pool coolers have on the RCB atmosphere temperature. The containment-confinement design (Case III-1) was used for these calculations. The effect would be essentially the same for any other containment operating in a purged mode.

c. Engineered RCB Vent/Purge

A method for reducing the hydrogen concentration in the RCB is to vent the building to atmospheric pressure and then purge the atmosphere with fresh air. The rate of purge is calculated by CACECO to hold the hydrogen concentration to 4%. Figure 35 compares Case I-1 hydrogen concentrations for purged and unpurged conditions. It should be noted that venting in the purged case occurred four hours sooner than in the nonpurged case. Waiting longer to vent resulted in hydrogen concentrations above 4% before ambient pressure was reached.

The main undesirable effect of the purge with air is the higher temperatures of the atmosphere of the RCB. The addition of oxygen continues the sodium/oxygen reaction and results in temperatures significantly higher than the unpurged case. Figure 36 compares these temperatures.

FIGURE 32. Pool Temperatures for Various Sized Coolers.

FIGURE 33. Sodium Boiloff vs Cavity Pool Cooler Size.

FIGURE 34. Effect of Pool Coolers on RCB Atmosphere Temperature.

FIGURE 35. Effect of Purge on RCB Hydrogen Concentration.

3. Accident Assumption Effects

a. Liner Conditions

The three base-case containment systems, I-2, II-2, and III-1, were calculated assuming that all cavity liners remained intact throughout the accident. For the purposes of comparison, Cases I-3(2), II-3(2), and III-3(1) were identical except for assuming a 100% failure of the cavity floor liner at the time of melt-through.

The effects of the floor liner failure are:

- 1) Additional heat is generated in the sodium pool by the reaction of sodium with exposed concrete.
- 2) Hydrogen and heat are generated in the cavity by the reaction of sodium with water released from the heated concrete.

The analysis has sodium reacting with the floor concrete at a rate of 1/2 inch per hour (1.27 cm/h) for four hours. At the end of four hours, the analysis assumes that further reaction stops due to an accumulation of reaction products. The additional heat generated by the sodium and concrete reaction amounts to only 2 to 6% of the fission product decay power over this time period. The energy of this reaction heats the sodium only about 25°F (14 K).

It should be noted that, even though the sodium and concrete reaction stops after four hours, the concrete continues to heat and release water vapor which reacts in the sodium pool. The reaction of sodium with water results in the formation of large amounts of hydrogen. Figure 37 shows the total amount formed in the cavity as a function of time. The energy input rate for this reaction is about 6% of the decay energy input rate one hour after the start of the accident.

FIGURE 36. Effect of Purge on RCB Temperatures.

FIGURE 37. Hydrogen Formed in Cavity due to Floor Liner Failure.

If the hydrogen recombination criteria described in Chapter III, Section D is used for analysis of the cases involving liner failure, the hydrogen created in the cavity does not pose a problem in the RCB. In all cases, the hydrogen concentration does not reach 4% in the RCB because when it leaks into the RCB, it reacts with oxygen to form water. Figures 38 and 39 compare hydrogen concentrations for the base single and double containment case with and without liner failure. The dip in the curve for the double containment case (Figure 39) is caused by burning (recombination) of the hydrogen in the RCB. At the start of the accident, sodium and hydrogen enter the domed area (inner containment) where the sodium condenses on the cool inner surfaces of the dome. When the hydrogen leaks out of the dome, not enough sodium is present to satisfy the requirements for recombination. Thus, the hydrogen concentration begins to increase in the RCB. At eight hours, when the inner dome is vented to the RCB, large amounts of sodium enter the RCB and satisfy the conditions for hydrogen recombination and the concentration drops to near zero. Later, at about eleven hours, the water vapor-to-oxygen ratio is such that recombination can no longer occur and the hydrogen concentration once again increases.

The basic conclusion is that floor liner failure does not significantly affect the accident if hydrogen recombination is assumed in the analysis.

b. Hydrogen Recombination

The two models used in the CACECO to describe hydrogen recombination and formation are discussed in detail in Chapter III, Section D. Basically, one model does not allow any recombination of hydrogen with oxygen, but does allow formation of hydrogen whenever sodium and water are mixed. If sodium enters a mixture of oxygen and water, the sodium is assumed to react with each in proportion to their concentrations. The other model allows hydrogen formed in the reactor cavity to react with oxygen when hydrogen/sodium mixtures, under certain conditions, are released into an air atmosphere. Formation of hydrogen on entry of sodium into a moisture-laden RCB air atmosphere is not allowed unless the RCB water concentration (molar) is equal to or greater than the oxygen concentration.

FIGURE 38. Effects of Cavity Floor Liner Failure on Hydrogen Concentration--Cases I-2, I-3(2).

FIGURE 39. Effect of Floor Liner Failure on RCB Hydrogen Concentration--Cases II-2, II-3(2).

FIGURE 40. Effect of Hydrogen Recombination Assumptions on Failed Liner Case I-3(2).

FIGURE 41. Effect of Hydrogen Recombination Assumptions on Failed Liner Case II-3(2).

HEDL 7708-8.4

The liner failure cases demonstrate the most dramatic effects of the two methods for handling the hydrogen. As stated in Section C-2a, liner failure does not significantly affect the accident if the hydrogen recombination model is used. If, however, the non-recombination model is used, the case scenario changes significantly. Hydrogen formed in the cavity does not recombine with oxygen as the hydrogen/nitrogen/sodium mixture enters the RCB atmosphere. Consequently, the hydrogen/oxygen mixture in the RCB reaches flammable limits within two hours after the start of the accident. The comparison of these concentrations are shown for two base cases in Figures 40 and 41.

The RCB hydrogen buildup in cases without liner failure is not particularly dependent on which hydrogen reaction model is used. As discussed in Section C-1c, the cases form hydrogen at the same rates, independent of which assumptions are used. As the oxygen is consumed, the concentration of oxygen becomes less than the concentration of water vapor, and this allows both models to begin forming hydrogen. The high input rate of sodium causes hydrogen to be created at a high rate for all cases not involving pool coolers.

c. Melt-Through vs Energetic Accident

The majority of cases studied involved a melt-through only, occurring 1000 seconds after the start of the accident. The base cases discussed to this point have all been of this type. The other type of accident that was analyzed involved an energetic pulse at time zero followed, at 1000 seconds, by complete core melt-through. This scenario is discussed in detail in Chapter III, Section 2. Basically, the energetic pulse is assumed to spray 1000 pounds of sodium from the reactor cavity into the next compartment. For the single containment cases, this next compartment is the RCB. For the other cases, this next compartment is the sealed head access area. In all cases, the sodium vapor is reacted immediately and uniformly throughout this chamber. The CACECO code was used to predict the resulting pressure, temperature, and leak rate histories. Included with the sodium are 10% of the volatile fission products, 100% of noble gas fission products, 1% of remaining fission products and 1% of the core fuel loading.

The pressure transients experienced within the various containment buildings are shown in Figure 42. It can be seen that the effect of the energetic pulse on the RCB pressure is much more severe in the single containment cases than for cases having a sealed head access area. For Case I-4(1), the increase in pressure after 0.3 hours is due to water vapor entering the RCB from cavity concrete.

It should be noted that the immediate, uniform burning of the sodium vapor in the RCB atmosphere is a very conservative assumption. Others ⁽³⁸⁾ handling similar cases use SPRAY⁽³⁵⁾ to model more realistically the spray of sodium vapor into the head access area. Using this method of analysis results in a 0.4 psi (2.8 kPa) initial pressurization of the RCB as opposed to the 5.7 psi (39 kPa) spike calculated by CACECO. This would reduce the initial out-leakage of bone dose-producing isotopes.

4. Special Cases

The base cases that were run at the beginning of this study pointed out one very important common result: The main contribution to the accident is the large amount of sodium that enters the RCB atmosphere in a relatively short time. After the basic cases were finished, it was decided to run several more cases in an attempt to establish the effect of inherent mechanisms on slowing down the boiloff of sodium to the RCB during the accident. The mechanisms considered: 1) large sodium pool; 2) higher thermal conductivity concrete in the cavity; 3) sodium vent to auxiliary cells; and 4) very high pressure (50 psig), low leakage RCB. Pool coolers, not an inherent mechanism, were discussed in Section C-2b.

a. Large Sodium Pool Case III-Pool

It is obvious that an increase in the quantity of sodium in the cavity pool will delay the time to start boiling. The increased amount of sodium will require more energy to heat it to the boiling point. Also, the larger cavity required to hold the sodium will provide more surface area for faster

heat loss from the pool. Since the fission product decay heat decreases with time, an extension of the time to reach boiling can significantly reduce the rate at which sodium eventually flows into the RCB.

Case III-Pool set up for the large pool study had 6 million pounds (2.72×10^6 kg) of sodium in a cavity 79 feet (24.1 m) in diameter and 59 feet (18 m) high. Other features used were similar to those of the base cases.

The delaying effect of the large sodium pool is shown in Figure 43. The sodium does not begin to boil until 56 hours after the start of the accident. At this time, the boiling rate is much slower than for the base case when it started to boil due to the reduced level of decay energy input. After 1000 hours, approximately 57% of the sodium has been boiled out of the cavity, leaving 2.6×10^6 pounds (1.18×10^6 kg) remaining, still boiling.

This large pool case clearly illustrates the disproportionate effect of increased sodium inventory on reaching sodium boiling; an increase of the amount of sodium by a factor of three increases the time to reach boiling by a factor of eight. Conversely, decreasing the sodium inventory will speed up the accident progression a disproportionate amount.

b. High Conductivity Concrete Cavity; Case I-TC

A second mechanism to reduce the amount of decay energy available for sodium boiling is to increase the heat loss capability of the cavity. This should have an effect similar to that of a pool cooler, as discussed in Section C-2b. In Case I-TC, the energy removal capability of the cavity was increased by increasing, by a factor of three, the thermal conductivity of the concrete in the cavity walls and floor. This case should also indicate the sensitivity of all cases studied with respect to the effect of concrete conductivity assumptions.

FIGURE 42. Effect of Head Expulsion on RCB Pressure.

FIGURE 43. Effect of Large Sodium Pool--Case III--Pool.

The effect of the three-fold increase in concrete thermal conductivity is to slow down the sodium heatup. However, the effect is not as significant as might be expected. The base case started boiling in approximately 7 hours; the high conductivity case started boiling in 10 hours. The base case boiled dry in approximately 140 hours, whereas the high conductivity case boiled dry in 192 hours.

In the early part of the accident, the higher thermal conductivity increased the heat losses of the pool. However, as the bulk temperature of the concrete rose, the heat transfer became dependent on transfer rates from outside cavity wall surfaces. Thus, after 100 hours, the high conductivity walls transferred energy at about the same rate as the walls of the low conductivity case.

c. Sodium Vent to Auxiliary Cells

A third mechanism to reduce or delay the sodium boilup into the RCB is to vent the sodium into the volume provided by large cells normally present in reactor containment systems and to take advantage of their passive heat absorption capability. Although this study did not examine this mechanism, others⁽³⁹⁾ did, using almost identical parameters. The containment system that they studied was similar to Case I-2, the base case single containment. The major difference was the addition of a large volume, including heat absorbing structures (steel, equipment, etc.), that connected directly to the reactor cavity and leaked to the RCB. By initially venting the reactor cavity to large HTS cells, they showed that containment integrity (no venting or pressure relief of RCB) can be maintained for 34 hours following the melt-through scenario. These results are a distinct improvement over the results of the present study.

d. Very High Pressure RCB; Case I-HP

The normal design pressure for "high pressure" RCB's in this study is 10 psig (69 kPa). A special single containment case (Case I-HP) was run to assess the

effects of increasing the design pressure of the RCB to 50 psig (345 kPa). The leak rate of this building was assumed to be 0.14 volume percent per day at design pressure.

Two iterations of this case were made: the first assumed that the cavity liners remained intact; the second assumed that one-half of all submerged cavity liners failed at 12 hours.

Results of the CACECO runs are shown in Figure 44. This figure plots the temperature of the cavity pool and the pressure of the RCB atmosphere as functions of time. The main purpose of this case was to determine the possibility of not relieving the RCB for at least twenty-four hours. As can be seen in Figure 44 both variations of this case have an RCB pressure below 50 psig at 50 hours. The atmosphere temperatures within containment are similar to the base case (I-2) previously shown in Figure 18. No consideration was given to structural failure as a result of the combination of high temperatures and pressure.

D. Radiological Evaluations

This section discusses the results of the calculations relating to radiological assessment. The first part, Section D-1, details the results from one case and explains which isotopes contribute to the various doses and when they contribute. The next three parts compare the effects of variation within each major containment system. The last part, section D-5, directly compares the major containment concepts.

The radiological evaluations for all cases discussed in the following sections were carried out by use of the COMRADEX computer code. This code must be supplied the initial isotopic inventory involved in the accident, the leak rates and fallout rates for each chamber of the system, and diffusion conditions outside of the building. The code calculates, as functions of time and distance: 1) the potential direct (gamma) dose from the building.

HEDL 7708-8.3

FIGURE 44. Effects of High Pressure (50 psig) RCB.

2) the potential internal dose resulting from inhalation of radioactivity from the released cloud; and 3) the potential external dose (gamma) from the released cloud.

The basic isotope inventory (total) is supplied by the RIBD code. The leak rate information is calculated by CACECO, and the fallout rates in the chambers are calculated by the HAA code. Techniques and assumptions used in these computer codes are discussed in Chapter IV, Section B.

The results of radiological studies of HCDA's are often evaluated in comparison to the site evaluation dose limits outlined in 10CFR100. 10CFR100 defines dose limits following a major accident on the basis of "exclusion" and "low population" distances as follows:⁽³⁾

1. An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.
2. A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from the iodine exposure.

For other organs, no limits are specified. However, 10CFR20 and ERDAM 0524 specify annual limits for occupational exposure of six times the whole body limit (5 rem) for bone dose and three times the limit for doses to other organs. Accordingly, similar proportions are used for extremely unlikely events, i.e., bone dose limit = 150 rem (6 x 25 rem) and lung dose limit = 75 rem (3 x 25 rem). These limits, including those listed in 10CFR100,

TABLE 11
DOSE LIMITS FOR SITE EVALUATION

ORGAN	2-HOUR DOSE (REM) AT EXCLUSION DISTANCE	30-DAY DOSE (REM) AT LOW POPULATION DISTANCE
WHOLE BODY	25	25
THYROID	300	300
BONE	150	150
LUNG	75	75
OTHER ORGANS	75	75

are shown in Table 11. It should be noted that no specific exclusion or low population zone distances were chosen for this study. Rather, the results are given for various distances and times.

1. Base Case

Case I-2, the base single containment system, is described in detail in this section. Detailed examination of the case provides insight into the understanding of other cases.

The whole body dose is made up of three parts: 1) The Direct Dose - a gamma dose emanating from isotopes in the building to a point outside; 2) The External Dose - a gamma dose received as a result of the radioactive cloud passing over a point some distance from the building; 3) The Internal Dose - a dose to the whole body as a result of inhalation of isotopes as the radioactive cloud passes the receptors.

The direct and external doses are received during the accident time period. The internal doses, however, are calculated to be cumulative over the lifetime of the individual. For this purpose, a 50-year commitment of these isotopes is assumed. During this time, radiological and biological half-lives are applied to continuously remove portions of the isotopes. Internal doses to the whole body and other specific organs - lung, thyroid, and bone - are due entirely to inhaled isotopes.

For all cases, COMRADEX calculated the separate doses due to noble gases (and their daughter products) and the volatiles (I, Br, Cs, etc.). This allowed a more accurate assessment of the contributors to the doses. In all cases, the noble gases were assumed to be ejected from the reactor cavity into the next compartment at the start of the accident. In the melt-through accident, this resulted in an early dose due entirely from the noble gases. The cases that included an initial head expulsion (1% Pu, 1% solids, etc.) combine the noble gases with the solids and volatiles that make up the expulsion to determine the early dose.

Case I-2 involves only a melt-through type accident. The contributions to whole body dose at 800 meters ($\sqrt{0.5}$ mi) due to noble gases are shown in Figure 45. The dose at short times is completely dominated by the direct dose from the RCB. As leakage begins to occur, the external dose from the cloud begins to build up. Eventually (at about 15 hours) the external portion of the dose exceeds the direct portion. The internal whole body dose is zero, because noble gases are not retained in the body. The total dose levels off at about one day. This implies that the portion of the cloud containing the noble gases and their daughters has passed. It should be remembered that, in all standard cases, the sodium starts to boil at approximately 7 hours, and by 12 hours, most cases are experiencing large releases. This release time is readily visible on all dose curves.

The equivalent curves for the volatiles that accompany the melt-through phase are shown in Figure 46. Obviously missing is the early portion of the direct dose curve. The volatiles remain in the cavity with the sodium until the sodium vapor pressure is high enough to begin transport into the RCB. The volatile whole body dose is due almost entirely to the internal contribution. This is due to the large quantities of iodine and cesium that are available. The external and direct doses are not significant because of the lack of hard gamma emitters in the volatiles.

Combining the results of both the noble gases and the volatiles gives the curves in Figure 47. Here it is obvious that the initial (short time) total whole body dose is due to the early noble gas release. At longer times, the volatiles and nobles contribute almost equally to the total whole body dose.

At this point, the effects of heavy shielding walls (concrete) on the RCB can be assessed. For short times, the total whole body dose will become insignificant, and for long times (about one day) the dose will be reduced by approximately 25%. This effect will become more important if distances closer to the RCB (< 800 meters) are considered.

HEDL 7706-119 7

FIGURE 45.
Contributions to Whole Body Dose from Noble
Gases--Case I-2--800 meters.

HEDL 7706-119 8

FIGURE 46.
Contributions to Whole Body Dose from Volatile
Isotopes--Case I-2--800 meters.

FIGURE 47.
Total Whole Body Dose as a Function of Time at
a Distance of 800 meters--Case I-2.

FIGURE 48.
Contributions to Whole Body Dose from Noble
Gases--Case I-2--10,000 meters.

Figures 48, 49, and 50 show the various contributors to the whole body dose at 10,000 meters (6.2 mi). Here, as might be expected, the direct dose is no longer significant, and the contribution by the volatiles outweighs the noble gas contributions by 45%.

The family of curves generated by plotting the total whole body dose for each distance as a function of time is shown in Figure 51. This type of curve is interesting for several reasons: 1) it shows at what distance the direct dose becomes negligible; and 2) it can be used to determine siting requirements.

Similar curves are shown for thyroid, lung, and bone dose in Figures 52 through 54. These three figures are based entirely on internal doses due to volatiles. In single (and double) containment cases, the noble gas isotopes and their daughters do not contribute significantly to these doses.

The thyroid doses in Figure 52 are very high because of the large amounts of radioactive iodine and the high F-factor associated with iodine. Even at 20,000 meters (12.4 miles) the 30-day dose exceeds the 300 rem guideline doses. The lung doses and bone doses shown in Figures 53 and 54 are much lower.

The relative contributions of the various isotopes to internal doses are shown in Tables 12 and 13. Both tables list only the volatile isotopes. The noble gases and their daughters do not contribute significantly to these internal doses. It can be noted that, for this particular case, inclusion of only the isotopes of iodine and cesium would have produced approximately the same internal dose.

Many of the basic conclusions for all cases studied can be drawn from this one case. It is for this reason that detailed breakdowns of the curves for each case are not presented.

FIGURE 49.
Contributions to Whole Body Dose from Volatile
Isotopes--Case E--10,000 meters.

FIGURE 50.
Total Whole Body Dose as a Function of Time--
Case I-2-10,000 meters.

FIGURE 51.
Total Whole Body Dose as a Function of Time
for Various Distances--Case I-2.

FIGURE 52.
Thyroid Dose as a Function of Time for Various
Distances--Case I-2.

FIGURE 53.
Lung Dose as a Function of Inhalation Time
for Various Distances--Case I-2.

FIGURE 54.
Bone Dose as a Function of Inhalation Time
for Various Distances--Case I-2.

TABLE 12

CONTRIBUTION TO 1.0 DAY INTERNAL DOSES
AT 800 METERS--CASE I-2 (PERCENT)

ISOTOPE	BONE	THYROID	TOTAL BODY	LUNG
AS 77	00.000	00.000	00.004	00.014
BR 82	00.000	00.000	00.012	00.008
BR 83	00.000	00.000	00.000	00.001
RB 86	00.000	00.000	00.520	00.729
SR 89	01.858	00.000	00.023	00.036
SR 90	00.055	00.000	00.000	00.000
SR 91	00.080	00.000	00.002	00.004
Y *91	00.002	00.000	00.000	00.000
Y 91	00.592	00.000	00.001	00.002
SR 92	00.001	00.000	00.000	00.000
Y 92	00.001	00.000	00.000	00.000
Y 93	00.004	00.000	00.000	00.000
CE 115	00.000	00.000	00.108	00.316
IN *115	00.000	00.000	00.003	00.009
I 131	00.000	77.431	45.729	37.271
I 132	00.000	00.031	00.025	00.018
I 133	00.000	21.022	10.147	09.310
CS 134	20.371	00.000	13.328	13.380
I 135	00.000	01.516	01.468	01.061
CS 135	08.000	00.000	00.000	00.000
CS 136	00.717	00.000	07.913	07.222
CS 137	68.567	00.000	18.322	23.704
BA 140	00.274	00.000	00.010	00.020
LA 140	00.004	00.000	00.000	00.001
CE 141	00.007	00.000	00.000	00.001
NA 22	00.000	00.000	00.004	00.026
NA 24	00.000	00.000	23.363	06.867

TABLE 13
CONTRIBUTION TO 30 DAY INTERNAL DOSES
AT 20,000 METERS--CASE I-2(PERCENT)

ISOTOPE		BONE	THYROID	TOTAL BODY	LUNG
AS	77	00.000	00.000	00.003	00.012
BR	82	00.000	00.000	00.010	00.007
PB	86	00.000	00.000	00.551	00.778
SR	89	01.847	00.000	00.024	00.039
SR	90	00.055	00.000	00.000	00.000
SR	91	00.043	00.000	00.001	00.002
Y	*91	00.001	00.000	00.000	00.000
Y	91	00.071	00.000	00.001	00.002
Y	92	00.001	00.000	00.000	00.000
Y	93	00.002	00.000	00.000	00.000
CD	115	00.000	00.000	00.097	00.286
IN	*115	00.000	00.000	00.003	00.009
I	131	00.000	82.796	46.818	38.478
I	132	00.000	00.011	00.009	00.006
I	133	00.000	16.143	07.422	06.867
CS	134	20.439	00.000	14.483	14.662
I	135	00.000	01.050	00.776	00.558
CS	135	00.000	00.000	00.000	00.000
CS	136	08.426	00.000	08.284	07.624
CS	137	68.837	00.000	19.923	25.991
BA	140	00.265	00.000	00.010	00.022
LA	140	00.006	00.000	00.000	00.001
CE	141	00.007	00.000	00.000	00.001
NA	22	00.000	00.000	00.005	00.029
NA	24	00.000	00.000	01.578	04.625

2. Radiological Evaluations of Single Containment Cases

Wherever possible the curves of all single containment cases are combined onto one graph for each organ to provide easy comparison. These curves are plotted as Dose versus Distance to enable siting analyses to be made.

The two-hour total whole body dose is shown for all single containment cases in Figure 55. Because of the lack of boiling until seven hours, this two-hour dose is similar for most of the cases. Only the noble gases contribute to this dose, a direct dose from the RCB. The only difference in cases occurs when the head expulsion cases, including some fuel, solids, nobles, and volatiles, are examined. In these cases, the whole body dose is higher at short distances because of the increased amount of gamma-producing isotopes in the RCB. The curve is heightened even more at large distances because of contributions of internal dose producers. All other cases do not release significant quantities of internal dose producers until later in the accident.

The lack of internal dose-producing isotopes at the early stage of the accident results in insignificant two-hour bone, thyroid, and lung doses for all cases except those involving a head expulsion. Case I-4(1) and Case I-4(2), yield significant early internal doses.* The internal doses to the organs are shown in Figure 56 for Case I-4(2). Application of the two-hour exclusion radius guidelines to this figure shows that the guideline of 150 rem for bone dose is met at a distance of 1500 meters (0.93 miles). This dose would be substantially reduced if a sodium spray code were used to more realistically model the initial expulsion. Subsequent calculations with the SPRAY⁽³⁵⁾ code indicate an initial 0.4 psi (2.8 kPa) pressurization of the RCB as opposed to the 5.7 psi (39 kPa) spike calculated by CACECO. Use of the SPRAY results would reduce the two-hour bone doses for Case I-4(1) by approximately 34% and for Case I-4(2) by approximately 66%. Analysis of the head expulsion case, where cavity concrete water is vented to the RCB, Case I-4(1), results in higher internal dose numbers than for Case I-4(2). The doses for I-4(1) for all organs are 80% higher than those shown in Figure 56, Case I-4(2).

*(Case I-4(1) vents cavity water release to the RCB; Case I-4(2) to condenser rooms).

FIGURE 55.
Comparison of 2-hour Total Whole Body Dose for
Single Containment Systems.

FIGURE 56.
Two-Hour Internal Dose to Various Organs--
Case I-4(2).

FIGURE 57.
Comparison of 30-Day Total Whole Body Dose
for Single Containment Systems.

FIGURE 58.
Comparison of 30-Day Thyroid Dose for Single
Containment Systems.

At 30 days, doses for the single containment cases begin to show individuality. Figure 57 shows the 30-days total whole body dose for all single containment systems. At 10,000 meters (6.2 miles) there is a factor of three between the highest and lowest doses for cases not involving a pool cooler. All of these are within the 25-rem guideline value. A comparison of cases with and without head expulsions shows that the effect on total whole body dose is insignificant.

When pool coolers are added to the system, Cases I-5a(2) and I-5c(2), the direct dose at short distances remains high due to noble gas release. However, at large distances, the 10-MW cooler case yields insignificant doses because of the very low release of iodine and cesium isotopes. The 2.5-MW pool cooler reduces the dose, but, as would be expected, not as impressively as the 10-MW pool cooler.

It is interesting to note that the failed floor liner in Case I-3(2) does not significantly increase the dose. As long as hydrogen recombination can be assumed, the effect of the failed liner is small.

The 30-day thyroid doses for single containment cases are shown in Figure 58. The results are about the same as for the whole body doses. Three notable points are: 1) the head expulsion cases are less important than their reference cases (the 10% release of volatiles with the head expulsion gives these isotopes more time to fall out before major releases are started); 2) the large (10-MW) pool cooler case is not on scale--not enough iodine is released to create a significant dose; 3) most of the cases are above the 300-Rem guideline thyroid dose even at 20,000 meters (12.4 miles). Filtering systems could be used to reduce this dose.

The 30-day lung doses for single containment systems are shown on Figure 59. The rankings are essentially the same as for the thyroid dose. The switch of case I-4(1) occurs due to the early release of actinides caused by pressurization of the RCB with water vapor. The majority of the cases are within the 75-Rem guideline values at 7,000 meters (4.4 miles).

HEDL 7706-119.21

FIGURE 59.
Comparison of 30-Day Lung Dose for Single
Containment Systems.

HEDL 7706-119.12

FIGURE 60.
Comparison of 30-Day Bone Dose for Single
Containment Systems.

The 30-day bone doses for single containment systems, shown in Figure 60, clearly show the importance of the head expulsion cases. Cases I-4(1) and I-4(2) are shifted more than one order of magnitude above the relative positions of their reference cases I-1 and I-2 (no head expulsion). The effect of early release of the Pu isotopes is very apparent. For the head expulsion case, where cavity concrete water is released to the RCB, I-4(1), doses are one order of magnitude higher than the head expulsion case where the water is released to condenser/drain rooms, I-4(2).

The cases not involving a head expulsion all have bone doses within the 150-Rem guideline value at 1000 meters (0.62 miles). The head expulsion cases, I-4(2) and I-4(1), meet the guideline value at 2300 meters (1.43 miles) and 17,000 meters (10.6 miles) respectively.

The timing of the bone dose is evident on inspection of Figure 61. This figure shows the 800-meter bone dose as a function of inhalation time, for a head expulsion case, I-4(2), and its reference case (I-2). The high doses are due almost entirely to inhalation of Pu isotopes.

Table 14 lists the contributions of each isotope to the 30-day, 20,000 meter internal doses to the important organs for Case I-4(2). Here again, the effect of the Pu isotopes on the bone dose is obvious.

3. Radiological Evaluations of Double Containment Cases

The curves showing the potential radiological consequences for double containment systems are shown in Figures 62 through 67. The two-hour total whole body dose, shown in Figure 62, is almost entirely due to noble gases and their decay products. The dome sealing the head access area promotes fallout and prevents early leakage of decay products of the noble gases; hence the whole body doses shown are due entirely to the direct gamma through the RCB.

FIGURE 61.
Comparison of 800-Meter Bone Doses for Single Containment Systems with and without Head Expulsion.

FIGURE 62.
Comparison of Two-Hour Total Whole Body Dose for Double Containment Systems.

TABLE 14

CONTRIBUTION TO 30-DAY INTERNAL DOSES
AT 20,000 METERS (PERCENT)--CASE I-4(2)

ISOTOPE		BONE	THYROID	TOTAL BODY	LUNG
AS	77	0.000	0.000	0.003	0.011
BR	82	0.000	0.000	0.008	0.007
BR	83	0.000	0.000	0.000	0.744
RB	86	0.000	0.000	0.464	0.000
SR	89	0.239	0.000	0.054	0.100
SR	90	0.185	0.000	0.000	0.010
Y	90	0.000	0.000	0.000	0.000
SR	91	0.005	0.000	0.002	0.004
Y	*91	0.000	0.000	0.000	0.000
Y	91	0.197	0.000	0.035	0.116
SR	92	0.000	0.000	0.000	0.000
Y	92	0.001	0.000	0.000	0.000
Y	93	0.003	0.000	0.000	0.003
ZR	95	0.005	0.000	0.110	0.160
NB	95	0.012	0.000	0.011	0.047
ZR	97	0.004	0.000	0.002	0.007
MO	99	0.000	0.000	0.005	0.013
TC	*99	0.000	0.000	0.000	0.000
RU	103	0.002	0.000	0.008	0.091
RH	*103	0.000	0.000	0.000	0.000
RU	105	0.000	0.000	0.000	0.000
RH	105	0.000	0.000	0.000	0.002
RU	106	0.009	0.000	0.008	0.360
PD	109	0.000	0.000	0.000	0.000
AG	111	0.000	0.000	0.000	0.002
PD	112	0.000	0.000	0.000	0.000
AG	112	0.000	0.000	0.000	0.000
CD	115	0.000	0.000	0.080	0.275
IN	*115	0.000	0.000	0.003	0.009
SN	125	0.001	0.000	0.000	0.001
SB	125	0.000	0.000	0.000	0.001
TE	*125	0.000	0.000	0.000	0.000
SB	127	0.000	0.000	0.000	0.000

TABLE 14

CONTRIBUTION TO 30-DAY INTERNAL DOSES
AT 20,000 METERS (PERCENT)

ISOTOPE		BONE	THYROID	TOTAL BODY	LUNG
TE	*127	00.000	00.000	00.000	00.002
TE	127	00.000	00.000	00.000	00.000
SN	128	00.000	00.000	00.000	00.000
SB	129	00.000	00.000	00.000	00.000
TE	*129	00.002	00.000	00.000	00.014
TE	129	00.000	00.000	00.000	00.000
I	131	00.000	82.705	39.400	36.900
TE	*131	00.001	00.000	00.004	00.006
TE	132	00.004	00.000	00.016	00.030
I	132	00.000	00.015	00.000	00.008
I	133	00.000	16.205	06.280	06.620
I	134	00.000	00.000	00.000	00.000
CS	134	01.220	00.000	12.200	14.100
I	135	00.000	01.075	00.666	00.550
CS	135	00.000	00.000	00.000	00.000
CS	136	00.500	00.000	06.970	07.320
CS	137	04.120	00.000	16.800	24.900
BA	140	00.130	00.000	00.070	00.170
LA	140	00.007	00.000	00.000	00.002
LA	141	00.000	00.000	00.000	00.000
CE	141	00.028	00.000	00.015	00.045
LA	142	00.000	00.000	00.000	00.000
CE	143	00.004	00.000	00.002	00.007
PR	143	00.027	00.000	00.053	00.030
CE	144	00.340	00.000	00.138	00.330
PR	144	00.000	00.000	00.000	00.000
PR	145	00.000	00.000	00.000	00.000
ND	147	00.008	00.000	00.004	00.011
PM	147	00.007	00.000	00.002	00.004
PM	149	00.001	00.000	00.000	00.002

TABLE 14

CONTRIBUTION TO 30-DAY INTERNAL DOSES
AT 20,000 METERS (PERCENT)

ISOTOPE		BONE	THYROID	TOTAL BODY	LUNG
SM	151	00.000	00.000	00.000	00.000
SM	153	00.000	00.000	00.000	00.000
EU	155	00.000	00.000	00.000	00.000
NA	24	00.000	00.000	00.000	00.000
PU	238	30.800	00.000	05.130	00.790
PU	239	08.220	00.000	01.310	00.170
PU	240	13.400	00.000	02.140	00.280
PU	241	36.300	00.000	04.910	00.038
AM	241	01.598	00.000	00.650	00.035
PU	242	00.040	00.000	00.006	00.000
CM	242	01.830	00.000	00.800	01.150
CM	244	00.760	00.000	00.297	00.030
NA	22	00.000	00.000	00.004	00.027
NA	24	00.000	00.000	01.320	04.420

FIGURE 63.
Two-Hour Internal Dose to Various Organs--
Case II-4(2).

FIGURE 64.
Comparison of 30-Day Total Whole Body Dose for
Double Containment Systems.

FIGURE 65.
Comparison of 30-Day Thyroid Dose for Double
Containment Systems.

FIGURE 66.
Comparison of 30-Day Lung Dose for Double
Containment Systems.

FIGURE 67.
Comparison of 30-Day Bone Dose for Double
Containment Systems.

FIGURE 68.
Two-Hour Total Whole Body Doses for
Containment-Confinement Systems.

The addition of a head expulsion, Case II-4(2), increases the two-hour whole body dose, but, again, only due to direct radiation from the RCB.

Cases are also shown where the inner dome was completely shielded. The effect of shielding this direct dose reduces the two-hour whole body dose by a factor of 20. This reduction is due to the fact that the majority of the noble gases and other isotopes remain in the inner containment dome during the effective two-hour dose release period.

The two-hour doses to the other organs are insignificant except for the head expulsion case, II-4(2). Here, the small amount of Pu and Iodine that leak out in the first two hours result in the low bone, thyroid and lung doses shown in Figure 63.

The effect of shielding the inner dome is not impressive from the standpoint of the 30-day total whole body doses. In these cases, shown in Figure 64, the direct dose accounts for only a small fraction of the total dose. The majority of the dose is due to inhalation (internal dose).

The effect of purging to reduce the hydrogen concentration, Case II-6(1), clearly increases the 30-day doses. However, since the controlled purge case releases would probably be filtered, the dose for this case would be lower had filters been assumed. Similar cases by others⁽³⁸⁾ on a different size reactor indicate a decrease in the 30-day internal doses of the same factor as the cleanup system decontamination factor.

The curves for 30-day thyroid, lung and bone doses show approximately the same relative ranking for the cases considered (Figures 65, 66, & 67). The 30-day bone dose, Figure 67, is especially interesting because it can be seen that the head expulsion case, II-4(2), dose is not significantly higher than that for the reference case (II-2). This is markedly different than the effect for single containment systems (Figure 60). The reason for the lack of increase in dose is the early retention and fallout of Pu isotopes in the sealed head access area. The relative standings of the other cases is the same as in previous figures.

4. Radiological Evaluations of Containment-Confinement Cases

The containment-confinement cases are similar to the double containment cases at times early in the accident because of the identical sealed head access areas. This inner containment effectively delays major releases into the RCB until later in the accident. However, once materials have leaked into the RCB, the exhaust system effectively clears it from the RCB atmosphere. Since the exhaust is filtered by cleanup systems in this type of containment, the internal doses are low. The base containment-confinement case (III-1) was also calculated without cleanup to assess the effect of the cleanup system. All cases with cleanup systems assumed an efficiency of 99% for solids and volatiles, and 0% for noble gases.

All the containment-confinement designs included a 100-meter stack release for all effluent. Some of the dose calculations were made without a stack in order to assess the effects of the stack.

The two-hour total whole body doses are shown in Figure 68. Almost all of this dose is due to noble gases and the external dose they produce by their passage after release. The case involving a failed floor liner, III-3(1), results in a higher dose than the unfailed liner cases. This is attributable to the increased leakages from the inner dome due to higher pressures caused by hydrogen formed when concrete water reacts with sodium. The effects of the cleanup system and the stack on this dose is shown in Figure 69 for the base case, III-1. The effects are minimal because the dose is caused by direct and external radiation from noble gases which are not filterable.

The two-hour internal doses (thyroid, lung and bone) are effectively reduced for all cases except III-4(1), the head expulsion case. Figure 70 shows these doses. The shape of the curves (lower dose, closer in) is characteristic of the effects of the stack.

FIGURE 69.
Comparison of Two-Hour Total Whole Body Doses
for Base Containment-Confinement System
Variations.

FIGURE 70.
Two-Hour Internal Doses--Case III-4(1).

The 30-day total whole body doses, shown in Figure 71, are due almost entirely to noble gases and their decay products. Again, the early dome releases caused by liner failure result in higher doses for case III-3(1). The cleanup system effectively reduces the release of isotopes that accompany the melt-through and are responsible for the internal doses.

An obvious way to improve two-hour and 30-day total whole body doses close to containment is to incorporate shielding into the RCB. This would result in dose curve shapes similar to those for the internal organs. The 30-day doses for the internal organs (thyroid, lung, and bone) are shown on Figures 72, 73 and 74.

The effects of the stack and cleanup system on all organ doses are significant. In each case, the doses at all distances are lowered to values less than those used for site consideration. For comparison, some no-filter, ground-release cases are presented. The relative contribution of the isotopes to internal doses is shown in Table 15.

The containment-confinement design offers a large reduction in dose because of its cleanup system and elevated release. The only doses of consequence are the two-hour and 30-day total whole body doses. In this design, the relatively high leakage dome allows noble gases to escape early in the accident. Once in the RCB, the gases are rapidly exhausted to the outside. Inside, they cause a direct dose through the building; outside, they cause an external dose even at large distances (see Figure 69 and Figure 71). The direct dose can effectively be reduced by the addition of shielding around the RCB (concrete, for example); but, to reduce the external dose, the noble gases must be contained within the RCB. There are two obvious solutions: the first is to design the inner containment dome to be more leak tight (for example, from 100 vol% per day leak rate down to 0.1 vol% per day leak rate). The disadvantages would be that the dome would have to be tested whenever it was removed for maintenance or refueling, and it would have to be designed for very high temperatures and pressures because of its small volume. The alternative method is to both lower the leak rate and increase the volume of the dome.

FIGURE 71.
Comparison of 30-Day Total Whole Body Doses
for Containment-Confinement Systems.

FIGURE 72.
Comparison of 30-Day Thyroid Doses for
Containment-Confinement Systems.

FIGURE 73.
Comparison of 30-Day Lung Doses for
Containment-Confinement Systems.

FIGURE 74.
Comparison of 30-Day Bone Doses for
Containment-Confinement Systems.

TABLE 15
CONTRIBUTION OF ISOTOPES TO 30-DAY INTERNAL DOSES
AT 20,000 METERS--CASE III-1

ISOTOPE		BONE (%)	THYROID (%)	TOTAL BODY (%)	LUNG (%)
AS	77	00.000	00.000	00.003	00.009
BR	82	00.000	00.000	00.008	00.005
RB	86	00.000	00.000	00.582	00.827
SR	89	01.831	00.000	00.026	00.043
SR	90	00.056	00.000	00.000	00.000
SR	91	00.017	00.000	00.000	00.001
Y	91	00.079	00.000	00.001	00.003
Y	93	00.001	00.000	00.000	00.000
CD	115	00.000	00.000	00.080	00.239
IN	*115	00.000	00.000	00.003	00.008
I	131	00.000	88.951	47.164	38.972
I	132	00.000	00.002	00.001	00.001
I	133	00.000	10.658	04.595	04.274
CS	134	20.528	00.000	15.896	16.179
I	135	00.000	00.389	00.270	00.195
CS	135	00.000	00.000	00.000	00.000
CS	136	08.021	00.000	08.618	07.974
CS	137	69.200	00.000	21.885	28.706
BA	140	00.252	00.000	00.010	00.023
LA	140	00.008	00.000	00.001	00.002
CE	141	00.007	00.000	00.000	00.001
NA	22	00.000	00.000	00.005	00.032
NA	24	00.000	00.000	00.851	02.507

FIGURE 75.
Doses vs Distance for Case I-2,
Filtered with Elevated Release.

FIGURE 76.
Comparison of Two-Hour Total Whole Body
Dose for Selected Containment Systems.

FIGURE 77.
Comparison of Two-Hour Total Lung
Dose for Selected Containment Systems.

FIGURE 78.
Comparison of Two-Hour Total Bone
Dose for Selected Containment Systems.

The extrapolation of this method would look like a single containment RCB system with an external confinement-type guard building.

To test this design, Case I-2 was recalculated assuming that all leakage was filtered and then released through a 100-meter stack. The results, shown in Figure 75, are impressive; only the total whole body doses are of significance. At short distances, even these would be drastically lowered with the addition of a concrete shield structure.

5. Comparison of Containment Systems on the Basis of Radiological Consequences

The objective of this study, to compare different containment systems and responses given the assumed accident conditions, can be best achieved by a direct comparison of the radiological consequences. Comparison will display the relative strengths and weaknesses of the different systems, and indicate possible ways to improve individual systems. For reasons of clarity, only representative systems are displayed in the figures.

Examination of the various two-hour doses, shown in Figures 76, 77 and 78, reveals that the only isotopes that significantly contribute to these early doses are the noble gases and plutonium. The total body doses (Figure 76) indicate that low leakage RCB's will reduce the two-hour dose by containing the noble gases. The dose from these buildings could be further reduced by an external shield structure, e.g., concrete, since this dose is almost entirely direct. The high leakage, forced ventilation RCB systems have a definite disadvantage here. Their whole body dose is almost entirely due to the external portion received during passage of the noble gas cloud.

Figure 78 indicates the distinct advantage an inner containment structure provides for the head expulsion cases. With no sealed head access area, (Cases I-4(2), and I-4(1)) the doses received during the early part of the accidents are significantly higher, even with low leakage RCB's.

The 30-day total whole body doses are displayed in Figure 79. The majority of the cases are fairly tightly grouped. Shielding the various buildings would greatly reduce the 600-meter dose. This is especially effective in Case I-5c(2), since in this case all of this dose is due to direct shine from the RCB. Improvements in the dose at distances greater than 1000 meters could be made by filtering the releases beginning at about 12 hours. This change would effectively remove the internal portion of the total whole body dose and in cases of type I and II, this would lower the dose by about a factor of two. Again, the confinement design encourages the release of noble gases once they escape from the inner containment. The 30-day thyroid doses shown in Figure 80 are substantial; however, adding filters to the systems at any time before 12 hours reduces this dose proportionate to the cleanup factor of the filters. This would not be completely true for I-4 cases (early head expulsion), since these involve early releases of iodine at the beginning of the accident.

The 30-day lung doses of Figure 81 show the same relative ranking. Here, as in the thyroid dose, the results would be improved with filtering. Large (8 to 10-MW) pool coolers would also reduce all of these doses.

The 30-day bone dose is shown in Figure 82. The only case that seems out of place is the single containment head expulsion case, I-4(1), in which cavity concrete water is vented to the RCB. In this case, the early high pressure created by the water vapor pressure promotes leakage of the Pu isotopes well before Case I-4(2). Both of these cases would show reduced bone dose if SPRAY were used to predict the early pressure/temperature histories due to the heat expulsion; however, their relative separation would still be maintained.

The listing in Table 16 shows the approximate distances, both exclusion radius and low population zone, that would be required for each case to meet guideline dose limits.

FIGURE 79.
Comparison of 30-Day Total Whole Body Dose
for Selected Containment Systems.

FIGURE 80.
Comparison of 30-Day Thyroid Dose as a Function
of Distance for Selected Containment Systems.

FIGURE 81.
Comparison of 30-Day Lung Dose as a Function
Distance for Selected Containment Systems.

FIGURE 82.
Comparison of 30-Day Bone Dose as a Function of
Distance for Selected Containment Systems.

TABLE 16

APPROXIMATE DISTANCES (IN METERS) REQUIRED TO MEET GUIDELINE DOSE LIMITS

DOSE PERIOD *	ORGAN	SINGLE CONTAINMENT CASES							DOUBLE CONTAINMENT CASES					CONTAINMENT-CONFINEMENT CASES			
		I-1	I-2	I-3(2)	I-4(1)	I-4(2)	I-5a (2)	I-5c (2)	II-1	II-2	II-3(2)	II-4(2)	II-6(1)	III-1	III-4(1)	III-3(1)	III-5b (1)
2-HOUR **	TOTAL WHOLE BODY	800	800	800	950	950	800	800	700	700	700	700	700	900	800	900	900
	THYROID	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)
	LUNG	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)
	BONE	(-)	(-)	(-)	2500	1500	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)	(-)
30-DAY ***	TOTAL WHOLE BODY	5000	4000	5000	8000	4000	2500	1100	6000	2600	3500	2600	13,000	5500	5500	18,000	4000
	THYROID	(+)	(+)	(+)	(+)	20,000	13,000	(-)	(+)	20,000	20,000	20,000	(+)	(-)	(-)	(-)	(-)
	LUNG	9000	5000	7000	10,000	5000	3000	(-)	11,000	4000	4000	4000	(+)	(-)	(-)	(-)	(-)
	BONE	850	(-)	700	17,000	2200	(-)	(-)	1,000	(-)	(-)	600	2300	(-)	(-)	(-)	(-)

* All doses calculated based on 50-year commitment
 ** For "Exclusion Area"
 *** For "Low Population Zone"

(-) Less than 600 meters
 (+) More than 20,000 meters

V. RECOMMENDATIONS FOR ADDITIONAL WORK

The following are work areas which could be beneficially pursued:

1. Perform a cost/benefit comparative analysis. The results of such an effort would allow a new dimension in the containment-configuration selection process. The fact that it would be a comparative study would keep the results from becoming obsolete as long as the ratio of the costs remain relatively constant.
2. Perform pool vs. loop comparative analyses. The advantages and disadvantages of each would be identified in more quantitative terms relating to both containment configuration selection and radiological consequences. This study could include a singular investigation of containment configurations for pool (or combination pool-loop) reactors similar to this document's investigation of loop reactors.
3. Perform optimization studies on those containment features, which, based on this study, offer the greatest potential for mitigating the consequences of core disruptive accidents. Some of these features are containment volumes, cavity vent discharge paths, and cavity structural and sodium parameters.
4. Continue the evaluation to include the effects which occur beyond sodium-boil dry, i.e., melting of core debris into the cavity floor.
5. Expand the evaluation to include an analysis of the effects of structural responses due to temperatures, pressures, and chemical reactions occurring during the accident. Also, expand the study to analyze the effects of structural methods (such as, using concrete as a containment structural material) on conditions within the RCB and on the radiological consequences.

VI. REFERENCES

1. "Proposed Annex to Appendix D, 10 CFR, Part 50."
2. E. R. Appleby, Compilation of Data and Descriptions for United States and Foreign Liquid Metal Fast Breeder Reactors, HEDL-TME 75-12, August 1975, Hanford Engineering Development Laboratory, Richland, WA.
3. W. B. Cottrell and A. W. Savolainen, eds., U. S. Reactor Containment Technology, ORNL-NSIC-5, Oak Ridge National Laboratory, Oak Ridge, Tennessee, August 1965.
4. W. R. Gee, Jr., et. al., Conceptual Plant Design, System Descriptions and Costs for a 1000 MWe Sodium Cooled Fast Reactor, GEAP-5678, General Electric Co., Sunnyvale, California, December 1968.
5. W. P. Staker, et. al., 1000 MWe LMFBR Follow-on-Study CEND-337 (Vol. 2), Combustion Engineering, Windsor, CT, 1968.
6. B. J. Goulding, et. al., 1000 MWe LMFBR Follow-on-Study, BAW-1328 (Vol. 1) Babcock & Wilcox, Lynchburg, VA 1968.
7. R. E. Aronstein, et. al., 1000 MWe LMFBR Follow-on-Study Conceptual Design Report, AI-12792 (Vol. IV), Atomics International, Canoga Park, 1968.
8. E. F. Beckett, et. al., 1000 MWe LMFBR Followon-Study, WARD-2000-97, Westinghouse Electric Corporation, ARD, Madison, PA 1968.
9. W. B. Cottrell, et. al., eds., US Reactor Containment Technology, ORNL-NSIC 5, dated August 1965, Oak Ridge National Laboratory, Oak Ridge, TN.
10. A. Sesonske and J. C. Yevick, "Description of Fast Reactors," Fast Reactor Technology: Plant Design, J. C. Revick, ed., MIT Press, MA, pp 677-746, 1966.
11. J. M. Laithwaite, "Containment Design in Fast Reactors," Proceedings of International Conference on Engineering of Fast Reactors for Safe and Reliable Operations, Karlsruhe, Germany, pp 454-469, October 9-13, 1972.
12. "Phase 1: Concept Selection, Prototype Large Breeder Reactor and Target Plant", NEDM-1477, General Electric Company, August 1976.
13. Liquid Metal Fast Breeder Reactor Plant Comparison, Phase 1 of PLBR Study, FBR-76-1, Rockwell International, June 1976.
14. ERDA-EPRI Breeder Reactor Project, Phase 1 Final Report, TID 27690, June 1977.

15. G. Billuris, et. al., "SEFOR Plant Design," Fast Reactors-ANS Topical Meeting, April 1967. San Francisco, CA.
16. R. K. Hilliard, et. a., Evaluation of Air Cleaning Concepts for Emergency Use in LMFBR Plants, HEDL-TME-76-41, December 1976, Hanford Engineering Development Laboratory, Richland, Wa.
17. K. Buttrey, et. al., Liquid Metal Fast Breeder Reactor (LMFBR) Task Force Fuel Cycle Study, NAA-SR-MEMO-12604, January 10, 1968, Atomics International, Canoga Park, CA.
18. D. R. Marr, A User's Manual for Computer Code RIBD-II, A Fission Product Inventory Code. HEDL-TME 75-26, January 1975, Hanford Engineering Development Laboratory, Richland, WA.
19. Aqueous Processing of LMFBR Fuels -- Technical Assessment and Experimental Program Definition, ORNL-4436, June 1970, Oak Ridge National Laboratory, Oak Ridge, Tennessee.
20. H. Ethrington, ed., Nuclear Engineering Handbook, McGraw-Hill, 1958.
21. L. Baker, Jr., "Postaccident Heat Removal in LMFBR's," Proceedings of ASME Symposium on Thermal and Hydraulic Aspects of Nuclear Reactor Safety, Atlanta, Ga, Nov. 1977, Volume 2.
22. H. K. Fauske, "The Role of Core-Disruptive Accidents in Design and Licensing LMFBR's," Nuclear Safety, Vol. 17, No. 5, September-October, 1976.
23. A. W. Castleman, "LMFBR Safety, I. Fission Product Behavior in Sodium," Nuclear Safety, Vol. 11, No. 5, September-October, 1970.
24. L. F. Parsly, Jr. and M. H. Fontana, Estimations of Fission Product Release from Melt During Concrete Penetration, Appendix G to Reactor Safety Study, WASH 1400, October 1975.
25. S. Jordan and Y. Ozawa, "Fuel Particle and Fission Product Release from LMFBR Core Catcher," Kern Forschungszentrum, Karlsruhe Laboratorium fur Aeroselphysik and Filtertechnik, ANS/ENS Conference on Fast Reactor Safety and Related Physics, Chicago, 1976.
26. J. L. Stakebake and H. N. Robinson, "Plutonium Release from Burning Sodium" Nuclear Technology, Vol. 33. April, 1977.
27. J. A. Hassberger, R. K. Hilliard, and L. D. Muhlestein, Sodium Concrete Reactor Tests, HEDL-TME 74-36, June, 1974.
28. Regulatory Guide 1.7, Rev. 1, September 1976, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident."
29. R. W. Wierman and R. K. Hilliard, Experimental Study of Hydrogen Formation and Recombination under Postulated LMFBR Accident Conditions," HEDL-TC-730, December 1, 1976.

30. "Third Level Thermal Margins in the CRBR," April 1976 Report Transmittal, Peter S. Van Nort, PMC, to Roger S. Boyd, NRC, April 22, 1976, Docket No. 50-537.
31. E. L. Gluekler, Status of Core-Meltdown Consequence Evaluation for Early Sized LMFBFR, August 1975, Re File 13853-AD-001.
32. R. D. Peak and D. D. Stepnewski, Computational Features of the CACECO Containment Analysis Code, TANSO 21-274, 1975.
33. L. Baurmash, et. al., HAA-3 User Report, AI-AEC-13088, 1973.
34. G. W. Spangler, et. al., Description of the COMRADEX Code, AI-67-TDR-108.
35. P. R. Shire, SPRAY Code User's Report, HEDL-TME 76-94, March 1977.
36. Internal Commission on Radiological Protection, Recommendations of the Internal Commission on Radiological Protection (Report of Committee 2 on Permissible Doses for Internal Radiation), ICRP Publication 2, Pergamon, London, England, 1959.
37. Regulatory Guide 1.4, "Assumptions for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," June, 1974.
38. S. F. Bankert, et. al., Containment Margins in FFTF for Postulated Failure of In-Vessel Postaccident Heat Removal. HEDL-TME 77-18, April 1977, Hanford Engineering Development Laboratory, Richland, WA.
39. R. D. Peak, et. al., Response of Liquid Metal Fast Breeder Reactor Containment to a Hypothetical Core Meltdown Accident, HEDL-SA-1262, May, 1977, Hanford Engineering Development Laboratory, Richland, WA.

APPENDIX A

CORE INVENTORY LIBRARY

APPENDIX A

CORE INVENTORY LIBRARY

Table A-1 in this Appendix presents the core isotopic inventory of a 2500 MW_{th} sodium-cooled fast breeder reactor at end-of-equilibrium cycle. Only those isotopes significant to radiological consequences assessment were used. Also added into this table are 100% of the volatiles (As, Se, Br, Rb, Cd, I, and Cs) and 100% of the noble gases from the axial and radial blankets. All other isotopes in the radial and axial blankets were not included in this study. The fission product inventory listed in this table was calculated based mainly on information presented in Reference 1.

The other data in Table A-1 are:

"CHN" Decay chains are printed in order. The first member of a chain decays into the next and so on until the chain is completed. Isotopes with very short lives are skipped. In some cases, an isotope may be a member of more than one chain; this indicates a split in the chain.

"GP" The elements are grouped by physical characteristics of the most probable state of compound. 0- solids (non volatile), 1-halogens, 2-nobles, 3-volatile solids (other than halogens and sodium), and 4-sodium.

"DECAY CONST" - Radioactive Decay Constant in s⁻¹

"GAMMA VAL" - Gamma Value in roentgen·ft²/Ci·h

F Values - "BONE", "THYROID", etc., in rem/Ci-inhaled.

Table A-2 lists the input numbers used by RIBD⁽²⁾ for the core, axial blanket, and radial blanket. For reference, the total actinide inventory of the core, taken from Reference 1, is shown in Table A-3.

The inventory of the sodium isotopes was hand calculated based on the data listed in Table A-4.

TABLE A.1
CORE INVENTORY LIBRARY

A-4

ISOTOPE	CHN	GP	DECAY CONST	INVENTORY (CI)	GAMMA VAL	BONE	THYROID	TOTAL BODY	LUNG
ZN 72	1	0	4.14062E-06	9.51368E+03	8.534E-01	0.	0.	0.	2.616E+04
GA 72	1	0	1.36552E-05	9.60008E+03	1.396E+01	1.153E+03	0.	2.541E+02	5.939E+03
ZN 73	2	0	2.94953E-02	1.15518E+04	0.	0.	0.	0.	3.592E+01
GA 73	2	0	3.94547E-05	1.26150E+04	2.092E+00	0.	0.	0.	1.073E+03
GE 75	3	0	1.39521E-04	4.27420E+04	2.210E-01	0.	0.	0.	2.050E+02
GE * 77	4	0	1.27650E-02	9.29089E+04	4.576E-01	0.	0.	0.	8.292E+00
AS 77	4	3	4.96234E-06	1.32865E+05	6.501E-02	0.	0.	1.097E+02	3.527E+03
GE * 77	5	0	1.27650E-02	5.22613E+04	4.576E-01	0.	0.	0.	8.292E+00
GF 77	5	0	1.70388E-05	7.73787E+04	6.761E+00	0.	0.	0.	0.
AS 77	5	3	4.96234E-06	7.50270E+04	6.501E-02	0.	0.	1.097E+02	3.527E+03
GE 78	6	0	1.32785E-04	3.21694E+05	1.739E+00	0.	0.	0.	2.664E+02
AS 78	6	3	1.27088E-04	3.60541E+05	5.146E+00	0.	0.	0.	0.
AS 79	7	3	1.28359E-03	7.39189E+05	2.014E-01	0.	0.	0.	4.964E+01
SE * 79	7	3	2.96975E-03	6.95946E+05	5.429E-01	0.	0.	0.	5.473E-01
SE 79	7	3	3.38150E-13	6.11081E+00	0.	0.	0.	0.	5.063E+04
AS 81	8	3	2.16606E-02	1.65877E+06	0.	0.	0.	0.	1.834E+02
SE 81	8	3	6.24451E-04	8.96757E+05	1.639E-01	0.	0.	0.	6.188E+01
SE * 81	9	3	2.01611E-04	7.45676E+04	4.525E-02	0.	0.	0.	8.767E+00
SE 81	9	3	6.24451E-04	8.97027E+05	0.	0.	0.	0.	6.471E+01
BR 82	10	1	5.43895E-06	3.97568E+04	1.639E-01	0.	0.	1.796E+03	1.160E+04
SE * 83	11	3	9.90200E-03	3.92703E+06	4.707E+00	0.	0.	0.	1.326E+01
BR 83	11	1	8.02245E-05	3.51081E+06	4.824E-02	0.	0.	0.	3.423E+02
KR * 83	11	2	1.03516E-04	2.94595E+06	4.580E-03	0.	0.	0.	0.

ISOTOPE		CHN	GP	DECAY CONST	INVENTORY (CI)	GAMMA VAL	BONE	THYROID	TOTAL BODY	LUNG
SE	83	12	3	5.13437E-04	2.94595E+06	1.400E+01	0.	0.	0.	0.
SR	83	12	1	8.02245E-05	3.51081E+06	4.824E-02	0.	0.	0.	3.423E+02
KR	* 83	12	2	1.03516E-04	3.51081E+06	4.580E-03	0.	0.	0.	0.
SE	84	13	3	3.50071E-03	1.02622E+07	0.	0.	0.	0.	2.608E+01
SR	84	13	1	3.63281E-04	1.06189E+07	1.209E+01	0.	0.	0.	0.
SR	* 84	14	1	1.92539E-03	6.07568E+05	1.294E+01	0.	0.	0.	0.
SE	85	15	3	1.77728E-02	4.76194E+06	0.	0.	0.	0.	0.
SR	85	15	1	4.02521E-03	9.87568E+06	0.	0.	0.	0.	1.185E+01
KR	* 85	15	2	4.29774E-05	1.00108E+07	8.883E-01	0.	0.	0.	0.
SE	85	16	3	1.77728E-02	1.11700E+06	0.	0.	0.	0.	0.
SR	85	16	1	4.02520E-03	2.31703E+06	0.	0.	0.	0.	1.662E+01
KR	* 85	16	2	4.29774E-05	2.34676E+06	8.883E-01	0.	0.	0.	0.
KR	85	16	2	2.04700E-09	1.43838E+05	1.260E+00	0.	0.	0.	0.
SR	86	17	3	4.30148E-07	2.87838E+05	5.818E-01	0.	0.	7.539E+03	1.022E+05
SE	87	18	3	1.23775E-01	3.85501E+06	0.	0.	0.	0.	0.
SR	87	18	1	1.24219E-02	1.44081E+07	1.704E+01	0.	0.	0.	0.
KR	87	18	2	1.52004E-04	2.03243E+07	7.397E+00	0.	0.	0.	0.
SR	87	18	3	4.67643E-19	2.86216E-04	0.	0.	0.	3.191E+03	8.752E+04
SR	88	19	1	4.35937E-02	1.12982E+07	2.412E+00	0.	0.	0.	7.173E+00
KR	88	19	2	6.87639E-05	2.88919E+07	9.070E+00	0.	0.	0.	0.
SR	88	19	3	6.52674E-04	3.01081E+07	3.100E+00	0.	0.	0.	0.
SR	89	20	1	1.54031E-01	7.05609E+06	0.	0.	0.	0.	2.750E+00
KR	89	20	2	3.65580E-03	3.18108E+07	1.297E+01	0.	0.	0.	0.
SR	89	20	3	7.60022E-04	3.93514E+07	1.377E+01	0.	0.	0.	2.397E+02
SR	89	20	0	1.54278E-07	3.15406E+07	0.	4.140E+05	0.	1.160E+04	1.804E+05

TABLE A.1 CONT'D

ISOTOPE	CHN	GP	DECAY CONST	INVENTORY (CI)	GAMMA VAL	BONE	THYROID	TOTAL BODY	LUNG
3R	90	21	1	4.33213E-01	3.45876E+06	0.	0.	0.	3.465E-01
KR	90	21	2	2.14594E-02	2.57414E+07	1.318E+01	0.	0.	0.
RB	90	21	3	4.27864E-03	3.31892E+07	1.657E+01	0.	0.	4.536E+01
SR	90	21	0	7.81647E-10	8.10802E+05	0.	1.600E+07	0.	1.149E+04
Y	90	21	0	3.00466E-06	8.68323E+05	0.	2.364E+04	0.	6.291E+02
KR	91	22	2	7.96713E-02	1.12184E+07	0.	0.	0.	0.
RB	91	22	3	1.19096E-02	2.44886E+07	0.	0.	0.	1.513E+01
SR	91	22	0	2.03100E-05	2.85640E+07	4.378E+00	6.450E+03	0.	4.095E+02
Y *	91	22	0	2.32441E-04	2.65710E+07	3.455E+00	2.130E+02	0.	8.687E+00
Y	91	22	0	1.36902E-07	2.57759E+07	2.118E-02	3.320E+05	0.	8.879E+03
KR	91	23	2	7.96713E-02	6.87579E+06	0.	0.	0.	0.
RB	91	23	3	1.19096E-02	1.50091E+07	0.	0.	0.	1.846E+01
SR	91	23	0	2.03100E-05	1.75070E+07	4.378E+00	8.838E+03	0.	4.095E+02
Y	91	23	0	1.36902E-07	1.57981E+07	2.118E-02	3.320E+05	0.	8.879E+03
KR	92	24	2	3.76707E-01	1.08313E+07	0.	0.	0.	0.
RB	92	24	3	1.53011E-01	3.48301E+07	0.	0.	0.	2.576E+00
SR	92	24	0	7.10475E-05	5.41368E+07	7.037E+00	3.799E+03	0.	1.235E+02
Y	92	24	0	5.45436E-05	5.49878E+07	1.369E+00	2.075E+03	0.	6.332E+01
KR	93	25	2	5.45780E-01	4.61310E+06	0.	0.	0.	0.
RB	93	25	3	1.19507E-01	2.64294E+07	0.	0.	0.	2.975E+00
SR	93	25	0	1.54031E-03	6.45384E+07	7.071E+00	0.	0.	1.673E+02
Y	93	25	0	1.88764E-05	6.77662E+07	9.176E-01	5.594E+03	0.	1.925E+02
ZR	93	25	0	2.31363E-14	4.11074E+01	0.	2.335E+04	0.	1.225E+03
KR	94	26	2	3.30067E+00	1.27669E+06	0.	0.	0.	0.
RB	94	26	3	2.57673E-01	1.32588E+07	0.	0.	0.	1.694E+00
SR	94	26	0	9.16852E-03	6.35311E+07	7.922E+00	0.	0.	2.938E+01
Y	94	26	0	6.08018E-04	7.37234E+07	4.239E+00	0.	0.	3.203E+02

TABLE A.1 CONT'D

ISOTOPE		CHN	GP	DECAY CONST	INVENTORY (CI)	GAMMA VAL BONE		THYROID	TOTAL BODY LUNG	
KR	95	27	2	1.38628E+00	3.02411E+05	0.	0.	0.	0.	0.
SR	95	27	0	2.66592E-02	5.63123E+07	0.	0.	0.	0.	1.894E+01
Y	95	27	0	1.10022E-03	8.11477E+07	1.706E+00	0.	0.	0.	1.280E+02
ZR	95	27	0	1.22480E-07	7.38496E+07	4.697E+00	6.276E+04	0.	1.629E+04	2.020E+05
N3	95	27	0	2.28560E-07	6.56842E+07	4.855E+00	1.320E+04	0.	4.500E+03	6.506E+04
NB	96	28	0	8.22816E-06	4.10676E+05	1.475E+01	0.	0.	0.	0.
SR	97	29	0	3.46570E+00	1.77428E+07	0.	0.	0.	0.	3.380E-02
Y	97	29	0	6.24451E-01	6.26712E+07	0.	0.	0.	0.	1.702E-02
ZR	97	29	0	1.14607E-05	9.10701E+07	8.609E-01	4.169E+03	0.	3.919E+02	1.041E+04
N3	* 97	29	0	1.28359E-02	7.88462E+07	4.841E+00	0.	0.	0.	5.277E+00
N3	97	29	0	1.56961E-04	9.17878E+07	4.432E+00	1.284E+02	0.	1.163E+01	2.994E+02
NB	* 98	30	0	2.26516E-04	9.48793E+07	0.	0.	0.	0.	0.
ZR	99	31	0	2.88808E-01	9.02210E+07	0.	0.	0.	0.	1.665E+00
N3	99	31	0	4.95100E-02	8.71558E+07	2.192E-01	0.	0.	0.	7.477E+00
MO	99	31	0	2.91637E-06	9.35641E+07	8.941E-01	0.	0.	6.238E+02	1.226E+04
TC	* 99	31	0	3.19832E-05	8.97057E+07	7.126E-01	5.225E-02	0.	8.443E+00	8.064E+01
TC	99	31	0	1.04595E-13	0.	0.	1.240E+02	0.	4.961E+01	9.152E+04
ZR	99	32	0	2.88808E-01	1.00246E+07	0.	0.	0.	0.	1.665E+00
N3	99	32	0	4.95100E-02	9.68398E+06	2.192E-01	0.	0.	0.	7.477E+00
MO	99	32	0	2.91637E-06	1.03960E+07	8.941E-01	0.	0.	6.238E+02	1.226E+04
N3	100	33	0	2.88808E-01	3.55211E+08	5.996E+00	0.	0.	0.	0.
N3	101	34	0	9.90200E-02	1.17793E+08	0.	0.	0.	0.	0.
MO	101	34	0	7.91256E-04	1.14773E+08	8.598E+00	0.	0.	0.	0.
TC	101	34	0	8.13545E-04	1.14830E+08	2.732E+00	0.	0.	0.	0.
MO	102	35	0	1.04075E-03	1.16843E+08	0.	0.	0.	0.	9.423E+01
TC	102	35	0	1.30781E-01	1.15386E+08	1.528E+00	0.	0.	0.	2.014E+01

TABLE A.1 CONT'D

ISOTOPE		CHN	GP	DECAY CONST	INVENTORY (CI)	GAMMA VAL BONE		THYROID	TOTAL BODY LUNG	
TC	103	36	0	1.38628E-02	2.69945E+08	2.660E-01	0.	0.	0.	1.936E+00
RU	103	36	0	2.02587E-07	1.13370E+08	3.103E+00	1.493E+03	0.	7.732E+02	7.437E+04
RH	*103	36	0	2.06292E-04	1.13363E+08	9.050E-03	1.583E+00	0.	8.009E-01	2.009E+01
MO	104	37	0	7.22021E-03	9.99942E+07	0.	0.	0.	0.	1.956E+01
TC	104	37	0	6.41796E-04	1.10165E+08	1.544E+01	0.	0.	0.	1.418E+02
MO	105	38	0	1.28359E-02	6.32966E+07	0.	0.	0.	0.	2.869E+01
TC	105	38	0	1.44404E-03	8.37372E+07	6.221E-01	0.	0.	0.	1.301E+02
RU	105	38	0	4.33646E-05	8.67686E+07	4.455E+00	1.339E+02	0.	6.115E+01	1.538E+03
RH	*105	38	0	1.87335E-02	2.23875E+07	7.296E-01	0.	0.	0.	7.863E-01
RH	105	38	0	5.42363E-06	8.67333E+07	4.987E-01	2.759E+02	0.	9.685E+01	2.600E+03
RU	106	39	0	2.17411E-08	3.00018E+07	0.	2.103E+04	0.	2.855E+03	1.097E+06
RH	106	39	0	2.31819E-02	3.00256E+07	1.310E+00	0.	0.	0.	4.129E+00
TC	107	40	0	2.39014E-02	3.81113E+07	0.	0.	0.	0.	2.701E+00
RU	107	40	0	2.75056E-03	5.07687E+07	1.338E+00	0.	0.	0.	1.263E+01
RH	107	40	0	5.32366E-04	5.39894E+07	2.114E+00	0.	0.	0.	6.061E+01
PD	107	40	0	3.38150E-15	4.21414E+00	0.	0.	0.	0.	0.
RH	109	41	0	7.70156E-03	4.81654E+07	0.	0.	0.	0.	1.068E+00
PD	109	41	0	1.43045E-05	2.62623E+07	3.040E-03	0.	0.	7.792E+01	2.156E+03
AG	*109	41	0	1.75035E-02	2.62581E+07	2.488E-02	0.	0.	0.	0.
PD	111	42	0	5.25106E-04	8.18004E+06	1.316E+00	0.	0.	0.	2.711E+02
AG	*111	42	0	1.06637E-02	5.80720E+06	3.676E-01	0.	0.	0.	2.956E+00
AG	111	42	0	1.07396E-06	5.89980E+06	1.503E-01	1.498E+03	0.	3.310E+02	2.508E+04
PD	112	43	0	9.57905E-06	3.07882E+06	2.149E-02	0.	0.	0.	1.122E+04
AG	112	43	0	6.15140E-05	3.09023E+06	3.805E+00	0.	0.	0.	1.587E+03
PD	113	44	0	7.70156E-03	2.07909E+06	0.	0.	0.	0.	6.812E+00
AG	113	44	0	3.63281E-05	1.85248E+06	3.782E+00	0.	0.	0.	1.567E+03

TABLE A.1 CONT'D

ISOTOPE		CHN GP		DECAY CONST	INVENTORY (CI)	GAMMA VAL		BONE	THYROID	TOTAL BODY	LUNG
PD	114	45	0	4.81347E-03	1.48879E+06	0.		0.	0.	0.	0.
AG	114	45	0	1.50683E-01	1.51848E+06	3.769E+00		0.	0.	0.	0.
AG	115	46	0	5.50111E-04	9.17696E+05	6.723E+00		0.	0.	0.	2.422E+02
CD	115	46	3	3.59886E-06	6.71081E+05	1.256E+00		0.	0.	4.092E+02	1.162E+04
IN	*115	46	0	4.27864E-05	6.14709E+05	1.100E+00		0.	1.084E+01	1.281E+01	3.458E+02
AG	*115	47	0	4.07729E-02	1.52334E+06	0.		0.	0.	0.	1.166E+00
CD	115	47	3	3.59886E-06	6.71081E+05	1.256E+00		0.	0.	4.092E+02	1.162E+04
IN	*115	47	0	4.27864E-05	6.14709E+05	1.100E+00		0.	1.084E+01	1.281E+01	3.458E+02
CD	*117	48	3	5.66291E-05	1.56108E+05	8.458E+00		0.	0.	0.	1.407E+03
CD	117	48	3	7.40534E-05	9.04595E+05	6.040E+00		0.	0.	0.	3.909E+02
IN	*117	48	0	9.92469E-05	8.34038E+05	7.646E-01		0.	0.	0.	0.
CD	118	49	3	2.29669E-04	1.10027E+06	0.		0.	0.	0.	1.807E+02
IN	118	49	0	2.62553E-03	1.00360E+06	1.076E+00		0.	0.	0.	1.580E+01
CD	119	50	3	1.22897E-03	7.41622E+05	1.066E+00		0.	0.	0.	6.912E+01
IN	*119	50	0	6.41796E-04	4.96198E+05	4.992E-01		0.	0.	0.	1.211E+02
SN	121	51	0	7.18429E-06	1.06728E+06	0.		0.	0.	0.	1.258E+03
SN	*123	52	0	2.88808E-04	4.58959E+05	1.322E-01		0.	0.	0.	3.004E+05
SN	123	53	0	6.21873E-08	6.72327E+05	7.601E-01		0.	0.	0.	0.
SN	125	54	0	8.31342E-07	3.59354E+05	5.238E-01	3.550E+04	8.121E+02	2.093E+03	7.881E+04	
SB	125	54	0	8.04553E-09	1.11757E+05	2.757E+00	1.813E+04	1.764E+01	3.497E+03	2.262E+05	
TE	*125	54	0	1.38318E-07	9.24041E+04	1.572E-02	0.	0.	7.170E+02	5.048E+04	
SN	125	55	0	8.31342E-07	1.35186E+06	5.238E-01	3.550E+04	8.121E+02	2.093E+03	7.881E+04	
SB	125	55	0	8.04553E-09	4.20421E+05	2.757E+00	1.813E+04	1.764E+01	3.497E+03	2.262E+05	

TABLE A.1 CONT'D

ISOTOPE	CHN	GP	DECAY CONST	INVENTORY (CI)	GAMMA VAL	ZONE	THYROID	TOTAL BODY	LUNG
SN 126	56	0	2.19793E-14	2.43145E+00	4.335E-01	0.	0.	0.	0.
S3 *126	56	0	6.08018E-04	1.39656E+05	9.312E-01	0.	0.	0.	0.
S5 126	56	0	6.46948E-07	1.54372E+05	1.159E+01	0.	0.	0.	0.
SN 127	57	0	9.08202E-05	7.93130E+05	9.674E+00	0.	0.	0.	1.705E+03
S3 127	57	0	2.11117E-06	1.35500E+06	3.143E+00	0.	0.	0.	7.264E+03
TE *127	57	0	7.36005E-08	1.03274E+06	1.040E-03	1.256E+04	0.	1.685E+03	1.637E+05
TE 127	57	0	2.05924E-05	1.30050E+06	9.710E-03	1.647E+02	1.252E+02	3.636E+01	8.561E+02
SN 127	58	0	9.08202E-05	4.16393E+06	9.674E+00	0.	0.	0.	1.731E+03
S3 127	58	0	2.11117E-06	7.11375E+06	3.143E+00	0.	0.	0.	8.147E+03
TE 127	58	0	2.05924E-05	6.82761E+06	9.710E-03	1.647E+02	1.252E+02	3.636E+01	8.561E+02
SN 128	59	0	1.95802E-04	1.03329E+07	2.907E+00	0.	0.	0.	5.506E+04
S3 *128	59	0	1.11080E-03	1.16833E+07	2.007E+01	0.	0.	0.	1.011E+04
S3 129	60	0	4.43638E-05	1.49397E+07	3.128E+00	0.	0.	0.	4.789E+03
TE *129	60	0	2.40193E-07	5.79501E+06	2.738E-01	1.903E+04	6.910E+03	4.709E+03	2.168E+05
TE 129	60	0	1.65033E-04	1.38586E+07	6.269E-01	5.014E+01	4.187E+01	1.958E+01	3.369E+02
I 129	60	1	1.38235E-15	5.74865E-01	2.036E-02	0.	6.698E+06	8.167E+03	7.983E+04
S3 129	61	0	4.43638E-05	8.40358E+06	3.128E+00	0.	0.	0.	5.814E+03
TE 129	61	0	1.65033E-04	7.79549E+06	6.269E-01	5.014E+01	4.187E+01	1.958E+01	3.369E+02
I 129	61	1	1.38235E-15	3.23784E-01	2.036E-02	0.	6.698E+06	8.167E+03	7.983E+04
SN 130	62	0	3.12225E-03	2.23559E+07	0.	0.	0.	0.	5.145E+01
S3 130	62	0	1.75035E-03	9.49860E+06	4.260E+00	0.	0.	0.	1.412E+02
S3 131	63	0	5.02275E-04	4.72525E+07	4.357E+00	0.	0.	0.	4.023E+02
TE 131	63	0	4.62093E-04	5.88896E+07	2.190E+00	0.	0.	0.	4.613E+01
I 131	63	1	9.97694E-07	6.34324E+07	2.432E+00	0.	1.486E+06	2.649E+03	2.090E+04
S3 131	64	0	5.02275E-04	8.33868E+06	4.357E+00	0.	0.	0.	5.054E+02
TE *131	64	0	6.41796E-06	1.00230E+07	7.303E+00	1.120E+03	1.063E+03	7.404E+02	1.143E+04
I 131	64	1	9.97694E-07	1.11919E+07	2.432E+00	0.	1.486E+06	2.649E+03	2.090E+04

TABLE A.1 CONT'D

ISOTOPE	CHN	GP	DECAY CONST	INVENTORY (CI)	GAMMA VAL	BONE	THYROID	TOTAL BODY	LUNG
SN 132	65	0	1.73285E-02	1.05323E+07	0.	0.	0.	0.	1.537E+01
SB 132	65	0	5.50111E-03	2.95035E+07	0.	0.	0.	0.	1.467E+01
TE 132	65	0	2.46845E-06	8.93021E+07	1.346E+00	3.224E+03	2.457E+03	2.014E+03	3.174E+04
I 132	65	1	8.42621E-05	1.01892E+08	1.496E+01	0.	5.288E+04	1.289E+02	8.842E+02
SB 133	66	0	4.81347E-03	2.15996E+07	0.	0.	0.	0.	9.470E+01
TF *133	66	0	2.08526E-04	3.77336E+07	1.619E+01	0.	0.	0.	2.133E+02
I 133	66	1	9.25668E-06	9.26216E+07	3.101E+00	0.	3.951E+05	5.727E+02	5.086E+03
XE 133	66	2	1.51653E-06	9.38378E+07	1.695E-01	0.	0.	0.	0.
SB 133	67	0	4.81347E-03	8.39985E+06	0.	0.	0.	0.	9.470E+01
TE 133	67	0	9.24187E-04	7.12569E+07	3.413E+00	0.	0.	0.	8.124E+00
I 133	67	1	9.25668E-06	3.60000E+07	3.101E+00	0.	3.951E+05	5.727E+02	5.086E+03
XE 133	67	2	1.51653E-06	3.64865E+07	1.695E-01	0.	0.	0.	0.
SB 134	68	0	8.15459E-01	8.40353E+07	0.	0.	0.	0.	1.408E+01
TE 134	68	0	2.75055E-04	8.94915E+07	7.920E-01	0.	0.	0.	7.440E+02
I 134	68	1	2.19626E-04	1.36784E+08	1.216E+01	0.	2.538E+04	4.326E+01	3.701E+02
CS 134	69	3	1.06623E-08	9.56216E+05	9.640E+00	3.760E+04	0.	5.660E+04	5.500E+05
TE 135	70	0	3.85078E-02	1.34090E+06	0.	0.	0.	0.	2.369E+00
I 135	70	1	2.92390E-05	3.02973E+06	1.018E+01	0.	1.231E+05	2.867E+02	1.981E+03
XE *135	70	2	7.55055E-04	2.72973E+07	2.741E+00	0.	0.	0.	0.
XE 135	70	2	2.09966E-05	3.37568E+06	1.544E+00	0.	0.	0.	0.
CS 135	70	3	9.55620E-15	8.34865E-01	0.	1.223E+04	0.	3.559E+03	3.213E+05
TE 135	71	0	3.85078E-02	5.45299E+07	0.	0.	0.	0.	2.369E+00
I 135	71	1	2.92390E-05	1.23270E+08	1.018E+01	0.	1.231E+05	2.867E+02	1.981E+03
XE 135	71	2	2.09966E-05	1.37405E+08	1.544E+00	0.	0.	0.	0.
CS 135	71	3	9.55620E-15	3.39459E+01	0.	1.223E+04	0.	3.559E+03	3.213E+05
I 136	72	1	8.35108E-03	8.63514E+07	1.339E+01	0.	0.	0.	2.700E+01
CS 136	73	3	6.17112E-07	5.90000E+06	1.250E+01	2.710E+03	0.	5.660E+03	5.000E+04

TABLE A-1. CONT'D

ISOTOPE	CHN	GP	DECAY CONST	INVENTORY (CI)	GAMMA VAL	BONE	THYROID	TOTAL BODY	LUNG
I 137	74	1	2.81764E-02	4.90632E+07	0.	0.	0.	0.	4.735E+00
XE 137	74	2	3.00842E-03	1.14622E+08	9.762E-01	0.	0.	0.	0.
CS 137	74	3	7.29710E-10	2.35081E+06	0.	5.145E+04	0.	3.163E+04	3.961E+05
3A *137	74	0	4.53033E-03	2.00177E+06	3.896E+00	0.	0.	0.	2.812E+00
I 138	75	1	1.06637E-01	2.54595E+07	0.	0.	0.	0.	6.840E-01
XE 138	75	2	8.13545E-04	9.20541E+07	8.376E+00	0.	0.	0.	0.
CS 138	75	3	3.58768E-04	1.16216E+08	1.183E+01	0.	0.	0.	0.
I 139	76	1	2.88808E-01	1.10169E+07	0.	0.	0.	0.	4.153E-01
XE 139	76	2	1.71569E-02	8.53243E+07	2.201E+00	0.	0.	0.	0.
CS 139	76	3	1.24219E-03	1.11568E+08	1.359E+00	0.	0.	0.	6.748E+01
3A 139	76	0	1.38684E-04	1.00874E+08	2.553E-01	0.	0.	0.	0.
XF 140	77	2	5.09662E-02	4.45428E+07	1.183E+01	0.	0.	0.	0.
CS 140	77	3	1.08643E-02	9.67297E+07	7.706E+00	0.	0.	0.	9.969E+00
3A 140	77	0	6.27242E-07	9.32240E+07	1.524E+00	9.007E+04	0.	7.269E+03	1.495E+05
LA 140	77	0	4.78595E-06	9.38220E+07	1.198E+01	4.765E+03	0.	8.369E+02	1.678E+04
XE 141	78	2	4.02988E-01	2.03892E+07	0.	0.	0.	0.	0.
CS 141	78	3	2.77256E-02	6.90779E+07	0.	0.	0.	0.	5.926E+00
3A 141	78	0	6.31275E-04	9.51056E+07	1.904E+00	0.	0.	0.	2.686E+02
LA 141	78	0	4.97517E-05	9.56604E+07	1.529E-01	0.	0.	0.	1.694E+03
CE 141	78	0	2.46617E-07	9.34089E+07	3.936E-01	2.070E+04	0.	1.727E+03	4.302E+04
XE 142	79	2	5.68148E-01	7.60349E+06	0.	0.	0.	0.	0.
CS 142	79	3	4.07729E-01	3.86948E+07	0.	0.	0.	0.	1.374E+00
3A 142	79	0	1.07966E-03	8.13834E+07	3.751E+00	0.	0.	0.	1.197E+02
LA 142	79	0	1.25025E-04	8.39386E+07	9.998E+00	0.	0.	0.	5.191E+02
CS 143	80	3	4.07729E-01	1.80465E+07	0.	0.	0.	0.	8.137E-02
3A 143	80	0	5.09662E-02	6.89547E+07	0.	0.	0.	0.	2.617E+00
LA 143	80	0	8.25167E-04	7.67324E+07	1.509E+00	0.	0.	0.	2.387E+02
CE 143	80	0	5.83451E-06	7.72979E+07	1.940E+00	4.132E+03	0.	3.511E+02	1.042E+04
PR 143	80	0	5.90756E-07	7.71488E+07	0.	2.293E+04	0.	1.136E+03	3.634E+04

TABLE A-1. CONT'D

ISOTOPE		CHN GP		DECAY CONST	INVENTORY (CI)	GAMMA VAL BONE		THYROID	TOTAL BODY LUNG	
CS	144	81	3	6.79549E-01	7.38447E+06	0.	0.	0.	0.	0.
BA	144	81	0	6.30127E-02	4.81496E+07	0.	0.	0.	0.	0.
LA	144	81	0	1.73285E-02	6.42435E+07	0.	0.	0.	0.	0.
GE	144	81	0	2.82083E-08	3.15716E+07	9.241E-02	7.739E+05	0.	4.756E+04	9.588E+05
PR	144	81	0	6.68538E-04	3.16706E+07	1.718E-01	0.	0.	0.	1.337E+02
CF	145	82	0	3.53071E-03	6.39180E+07	5.028E+00	0.	0.	0.	1.204E+01
PR	145	82	0	3.21971E-05	5.57926E+07	9.241E-02	0.	0.	0.	1.451E+03
GE	146	83	0	8.13545E-04	4.61536E+07	2.068E+00	0.	0.	0.	0.
PR	146	83	0	4.77369E-04	4.60271E+07	1.042E+01	0.	0.	0.	0.
ND	147	84	0	7.29978E-07	3.78303E+07	9.251E-01	1.465E+04	0.	1.152E+03	2.815E+04
PM	147	84	0	8.37372E-09	6.47901E+06	0.	6.819E+04	0.	3.383E+03	6.157E+04
SM	147	84	0	2.05411E-19	2.18676E-05	0.	0.	0.	0.	0.
ND	149	85	0	1.11294E-04	2.52656E+07	2.134E+00	2.893E+02	0.	2.418E+01	7.612E+02
PM	149	85	0	3.62597E-06	2.47350E+07	6.019E-01	3.987E+03	0.	3.142E+02	8.828E+03
ND	151	86	0	9.31640E-04	1.49193E+07	1.913E+00	0.	0.	0.	0.
PM	151	86	0	6.77954E-06	1.49335E+07	1.578E+00	0.	0.	0.	0.
SM	151	86	0	2.36333E-10	8.38598E+04	4.977E-03	2.862E+04	0.	2.317E+03	4.077E+04
SM	153	87	0	4.12289E-06	7.76226E+06	2.099E-01	2.044E+03	0.	1.546E+02	4.627E+03
SM	155	88	0	5.20375E-04	3.87394E+06	4.909E-01	0.	0.	0.	0.
EU	155	88	0	4.57911E-09	3.69817E+05	2.762E-01	5.698E+04	0.	8.134E+03	8.735E+04
SM	156	89	0	2.04829E-05	2.31470E+06	6.312E-01	0.	0.	0.	1.070E+03
EU	156	89	0	5.27793E-07	2.98786E+06	5.508E+00	0.	0.	0.	0.
EU	157	90	0	1.26670E-05	1.73125E+06	1.686E+00	0.	0.	0.	0.
EU	158	91	0	2.51685E-04	2.03705E+06	3.590E+00	0.	0.	0.	0.

TABLE A.1 CONT'D

ISOTOPE	CHN	GP	DECAY CONST	INVENTORY (CI)	GAMMA VAL	BONE	THYROID	TOTAL BODY	LUNG
SD 159	92	0	1.03516E-05	6.78171E+05	2.245E-01	6.526E+02	0.	7.115E+01	2.272E+03
COOLANT INCLUDED WITH FISSION PRODUCTS FOR DOSE CALCULATIONS									
2900000.000L3 SODIUM AVAILABLE FOR RELEASE									
NA 22	93	4	8.38914E-09	1.29730E+03	1.280E+01	0.	0.	1.390E+04	8.000E+05
NA 24	94	4	1.28704E-05	1.67569E+07	1.960E+01	0.	0.	1.280E+03	3.600E+04
FUEL INCLUDED WITH FISSION PRODUCTS FOR DOSE CALCULATIONS									
100.0000000 PERCENT OF FUEL RELEASED TO FIRST CHAMBER									
PU 238	95	0	2.46900E-10	3.86757E+05	0.	5.700E+09	0.	1.440E+08	1.880E+08
PU 239	96	0	9.01900E-13	8.97027E+04	0.	6.560E+09	0.	1.590E+08	1.770E+08
PU 240	97	0	3.33000E-12	1.46162E+05	0.	6.550E+09	0.	1.590E+08	1.770E+08
PU 241	98	0	1.69000E-09	2.09162E+07	0.	1.240E+08	0.	2.550E+06	1.650E+05
AM 241	98	0	4.79500E-11	5.19189E+04	4.386E-01	2.070E+09	0.	1.360E+08	6.260E+07
PU 242	99	0	5.79700E-14	4.43514E+02	0.	6.080E+09	0.	1.530E+08	1.700E+08
CM 242	100	0	4.93600E-08	2.57054E+06	0.	5.090E+07	0.	3.380E+06	4.050E+07
CM 244	101	0	1.19700E-09	4.32432E+04	0.	1.260E+09	0.	7.450E+07	6.480E+07
***** WHITEPE //// END OF LIST ////									
***** WHITEPE //// END OF LIST ////									

TABLE A-2
INPUT TO RIBD²

	Core	Blanket	Radial Blanket	Axial
Flux (n/cm ² ·s)		5.15 x 10 ¹⁵	1.35 x 10 ¹⁵	9.71 x 10 ¹⁴
Burnup (MWD/MT)		80,000	8,100	2,500
Charge (MT U + Pu)		12.6	26.7	7.32
No. Cycles to Burnup		3	3	3
Load Factor (%)		85	85	85
Power at 100%(MW)		2196	40	267
Cycle Length (days on/days off)		153/27	153/27	153/27
"First" Fuel		Pu-239	U-239	U-239
"Second" Fuel		U-238	Pu-239	Pu-239
F ₀ (a)		.871	1.00	1.00
F ₁ (b)		.8676	.446	.652
T(c)		0.0013	0.2694	0.1426
CON(d) $\frac{\text{gm}}{\text{MWD}}$		0.5	3.51	4.59
ENF (Mev/fission)		213.5	213.5	213.4

(a) F₀ is the fraction of the fission rate in the first fuel at time zero.

(b) F₁ is the fraction of the fission rate in the first fuel at the end of the first power step.

(c) $T = \ln \frac{F_0}{F_1}$

(d) Conversion rate of ²³⁸U to ²³⁹Pu via breeding process.

TABLE A-3

RADIOACTIVITY OF ACTINIDES IN SPENT LMFBR CORE FUEL AS A FUNCTION OF DECAY TIME (1)

Burnup = 80,000 Mwd/metric ton
 Specific power = 148.15 Mw/metric ton
 Nuclide activities are given in curies per metric ton (U + Pu) charged to core*

	Charged to Reactor	Discharged from Reactor	After Cooling Times of:				
			30 days	90 days	150 days	3 years	30 years
^{228}Th + daughters	0	0.00519	0.0060	0.00816	0.0107	0.0744	0.202
^{232}U	0	0.00595	0.00668	0.00812	0.00949	0.0249	0.0337
^{234}U	0	0.135	0.142	0.156	0.171	0.397	2.52
^{237}U	0	302,000	13,900	29.2	0.0617		
^{238}U	0.260	0.239	0.239	0.239	0.239	0.239	0.239
^{237}Np	0	0.124	0.127	0.127	0.127	0.134	0.391
^{239}Np	0	81,500,000	11,900	137	137	137	136
^{236}Pu	0	0.946	0.931	0.894	0.859	0.458	0.000643
^{238}Pu	43,800	30,700	30,800	31,000	31,000	31,000	25,200
^{239}Pu	7,940	7,120	7,150	7,150	7,150	7,150	7,140
^{240}Pu	11,400	11,600	11,600	11,600	11,600	11,500	11,500
^{241}Pu	2,970,000	1,660,000	1,650,000	1,640,000	1,620,000	1,420,000	325,000
^{242}Pu	34.2	35.2	35.2	35.2	35.2	35.2	35.2
^{241}Am	0	4,120	4,320	4,730	5,140	11,100	40,500
^{242}Am	0	239	239	239	239	236	209
^{242}Am	0	393,000	239	239	239	236	209
^{243}Am	0	137	137	137	137	137	136
^{242}Cm	0	204,000	181,000	140,000	109,000	2,130	171
^{243}Cm	0	107	107	106	106	100	55.7
^{244}Cm	0	3,430	3,420	3,400	3,380	3,060	1,090
Total	3,030,000	167,000,000	1,920,000	1,840,000	1,790,000	1,480,000	422,000

* Total Cost Loading is 12.6 MT (U + Pu)

TABLE A-4
INPUT FOR ^{23}Na ACTIVATION CALCULATIONS

	Core	Radial Blanket	Axial Blanket
ϕ (a) ($\text{n}/\text{cm}^2 \cdot \text{s}$)	5.15×10^{15}	1.35×10^{15}	9.7×10^{14}
$\sigma(n, \gamma)$ (a) (cm^2)	1.73×10^{-27}	1.73×10^{-27}	1.73×10^{-27}
$\sigma(n, 2n)$ (a) (cm^2)	4.5×10^{-31}	4.5×10^{-31}	4.5×10^{-31}
Volume of Na (b) (cm^3)	2.52×10^6	1.73×10^6	1.47×10^6
ρ Sodium (g/cm^3)	0.817	0.817	0.817
$\lambda(^{22}\text{Na})$ (a) (s^{-1})	8.45×10^{-9}	8.45×10^{-9}	8.45×10^{-9}
$\lambda(^{24}\text{Na})$ (a) (s^{-1})	1.28×10^{-5}	1.28×10^{-5}	1.28×10^{-5}

(a) Information taken from Reference 1.

(b) Information calculated from Reference 3, p. 64.

REFERENCES

1. Aqueous Processing of LMFBR Fuels, ORNL-4436, June 1970, Oak Ridge National Laboratory, Oak Ridge, Tenn.
2. D. R. Marr, A User's Manual for Computer Code RIBD-II, A Fission Product Inventory Code, HEDL-TME 75-26, January 1975, Hanford Engineering Development Laboratory, Richland, WA.
3. K. Buttrey, et. al., Liquid Metal Fast Breeder Reactor Task Force Fuel Cycle Study, NAA-SR-MEMO-12604, January 10, 1968, Atomics International, Canoga Park, CA.

APPENDIX B

ASSUMPTIONS USED IN
CACECO CALCULATIONS

APPENDIX B

ASSUMPTIONS USED IN CACECO CALCULATIONS

The CACECO^(1,2) code requires a large number of input thermal properties, reaction rates, and other physical data. This appendix presents these data as they were used by CACECO for this study.

CONTENTS OF APPENDIX B

<u>Section</u>	<u>Title</u>
I	Concrete Data
II	Chemical Reactions
III	Heat Structure Physical Properties
IV	Heat Transfer Coefficients
V	Mass Transfer Coefficients
VI	Cell Characteristics
VII	References

I. CONCRETE DATA

Included in this section are tables that give the properties of the limestone concrete that were used in this study. Typical values for other types of concretes can be found in Reference 2.

The four tables included in this section are:

B-1	Concrete Water Release Upon Heating
B-2	Effective Addition to Concrete Heating Capacity Due to: (A) Latent Heat of Water Released (B) Carbon Dioxide
B-3	Thermal Conductivity of Concrete
B-4	Carbon Dioxide Release Upon Heating Limestone Concrete

TABLE B-1
CONCRETE WATER RELEASE UPON HEATING

Temperature (° F)	Release (lb/ft ³ -Concrete)
0	0.00
190	0.00
275	3.85
285	5.78
420	6.84
700	7.51
800	8.28
1500	9.63
2000	9.63

TABLE B-2

EFFECTIVE ADDITION TO CONCRETE HEAT CAPACITY
DUE TO: A) LATENT HEAT OF WATER RELEASED; AND B) CARBON DIOXIDE

Temperature (°F)	Heat Capacity Addition (Btu/ft ³ ·°F)
0	0.00
190	0.00
190	43.99
275	43.99
275	186.89
285	186.89
285	7.62
420	7.62
420	2.32
700	2.32
700	7.52
800	7.52
800	1.85
1500	1.85
1500	0.00
1630	0.00
1630	1100.00
1710	1100.00
1710	0.00
2000	0.00
2001	0.00

TABLE B-3
THERMAL CONDUCTIVITY OF CONCRETE

<u>Temperature (°F)</u>	<u>Thermal Conductivity (Btu/h·ft·°F)</u>
0	1.3
76	1.3
500	1.1
1000	.9
1500	.7
2000	.7

TABLE B-4
CARBON DIOXIDE RELEASE UPON HEATING LIMESTONE CONCRETE

<u>Temperature (°F)</u>	<u>Release (lb/ft³-Concrete)</u>
0	0.0
1630	0.0
1710	52.5
2000	52.5

II. CHEMICAL REACTIONS

The majority of the chemical reactions analyzed by CACECO have sodium as one reactant and oxygen, water, concrete and hydrogen as the other reactants. In addition, the hydrogen-oxygen reaction is analyzed. The assumed heats of all reactions used in this study are listed in Table B-5.

TABLE B-5
HEATS OF REACTIONS OF SODIUM

<u>Reaction</u>	<u>Energy Release Rate</u>
Sodium Vapor + Oxygen	5,700 Btu/lb-Na
Sodium Vapor + Water Vapor	3,400 Btu/lb-Na
Sodium Liquid + Water Vapor	1,600 Btu/lb-Na
Sodium Oxide + Water Vapor	1,500 Btu/lb-Na
Sodium + Concrete ^(a)	72,750 Btu/ft ³ -concrete
Sodium + Hydrogen ^(b)	915-1,010 Btu/lb-Na
Hydrogen + Oxygen ^(b)	52,425 Btu/lb-H ₂

-
- (a) The sodium/concrete reaction is assumed to consume concrete at a rate of 1/2 inch (thickness of concrete) per hour. At the end of 4 hours, the reaction is assumed to cease due to an accumulation of reaction products.
- (b) These reactions were allowed only when the hydrogen recombination model was used.

III. PHYSICAL PROPERTIES OF HEAT STRUCTURES

The structures that are in and make up the containment systems can have a significant effect on the accident scenario as calculated by CACECO. A description of the physical properties of these structures is given in Table B-6.

TABLE B-6
HEAT STRUCTURES

<u>Structure</u>	<u>Description</u>
1.	Equipment in Reactor Containment Building. 6×10^6 lb of steel, 0.1 ft thick, 1.2×10^4 ft ³ .
2.	Roof and walls of Reactor Containment Building. 1.5 inch thick steel, with insulation on outside surface area = 7.04×10^4 ft ² /side. Insulation melts and sluffs off in range 600-1300°F. *
3.	Floor of containment building concrete, 6.5 ft thick, 1.76×10^4 ft ² .
4.	Head access compartment walls, Concrete, 5 ft. thick, lined with 3/8 inch steel; surface area is 2.96×10^3 ft ² .
5.	Reactor vessel head - 1.7×10^6 lb steel + 2 inch Kaylo insulation; steel is 5.67 ft thick; area = 4.2×10^2 ft ² .
6.	Reactor cavity equipment and vessels 2.4×10^6 lb steel. 0.1 ft thick; area = 4.9×10^4 ft ² .
7.	Reactor cavity walls, below sodium pool; 6.5 ft thick concrete, with 3/8 inch thick steel liner. Liner attached to steel imbedments in concrete surface area is 3.3×10^3 ft ² .
8.	Reactor cavity walls, below sodium pool; 6.5-ft thick concrete, with 3/8 inch thick steel liner. Liner attached to steel imbedments in concrete; no gap. Surface area is 3.93×10^3 ft ² .

* For containment-confinement case, this is split into two structures; roof and walls. The walls behave as shown, but the roof insulation is not allowed to burn or sluff off.

TABLE B-6 (Cond't)

<u>Structure</u>	<u>Description</u>
9.	Reactor cavity floor. 6.5-ft thick concrete, with 3/8 inch thick steel liner. Liner attached to steel imbedments in concrete; no gap. Surface area is $1.256 \times 10^3 \text{ ft}^2$.
10.	Dome over head access compartment. (not used for single containment cases) 1-in. thick steel, $2.5 \times 10^3 \text{ ft}^2/\text{side}$.
11.	Sodium pool in reactor cavity pool area = $1.256 \times 10^3 \text{ ft}^2$, depth is 31.25 ft, when $2 \times 10^6 \text{ lb}$ of sodium are dumped.
12.	Sodium pool in dome head access cell (not used for single containment cases). Area = $1.256 \times 10^3 \text{ ft}^2$.
13.	Sodium pool in RCB Area $8.6 \times 10^3 \text{ ft}^2$.

IV. THERMAL PROPERTIES OF STRUCTURES AND ATMOSPHERES

The heat transfer coefficients, thermal conductivities and heat capacities of the materials and atmospheres within the containment system must be provided to CACECO by the user. Table B-7 lists the thermal conductivity and heat capacities of all "standard" materials used in the containment model. Tables B-8 through B-10 list heat transfer coefficients assumed for atmosphere and pool situations. Table B-11 shows the method used to model the melting of the insulation for the outside of the RCB roof and walls.

An overall area-view-emissivity of $45,500 \text{ ft}^2$ was assigned to the outside of the RCB. This factor assumed emissivities of 0.7 for the outside of the RCB and 0.9 for the surroundings.

TABLE B-7
THERMAL CONDUCTIVITY AND VOLUMETRIC HEAT CAPACITY

<u>Material</u>	<u>Thermal Conductivity (Btu/ft·h·°F)</u>	<u>Heat Capacity Btu/ft³·°F)</u>
Cellular urethane foam insulation on building roof and outer wall.	1.33E-02	6.00E-01
Carbon steel at 100 to 1600 °F.	2.60E+01	5.39E+01
Kaylo.	5.00E-02	2.40E+00
Head shielding of reactor vessel. Steel at quarter therm. conduc.	6.50E+00	5.39E+01
First 30 in. of steel + concrete behind EFCO cold liner	1.04E+00	3.00E+01
Limestone concrete, 150 lb/ft ³ density. No latent heat of water.	8.00E-01	3.00E+01

TABLE B-8

NATURAL CONVECTION HEAT TRANSFER COEFFICIENTS FOR NITROGEN

NATURAL CONVECTION HEAT TRANSFER COEFFICIENTS FOR NITROGEN AT 90%, 50% AND 10% DENSITY WHICH IS SATURATED WITH SODIUM VAPOR WHICH, IN TURN, CONDENSES ON SURFACES 100° F COOLER. USED FOR THE REACTOR CAVITY AND SEALED HEAD ACCESS AREA.

<u>Temperature (°F)</u>	<u>Heat Transfer Coefficient (Btu/h·ft²·°F)</u>
0	1.052E+00
100	1.052E+00
300	9.830E-01
500	9.530E-01
700	9.510E-01
900	1.053E+00
1100	1.336E+00
1300	2.958E+00
1500	2.962E+00
1700	2.986E+00
1900	5.109E+00

TABLE B-9

NATURAL CONVECTION HEAT TRANSFER COEFFICIENTS FOR THE CAVITY POOL.
THESE COEFFICIENTS ARE A FUNCTION OF TIME TO MINIMIZE TEMPERATURE
OSCILLATIONS.

<u>Time (Hours)</u>	<u>Heat Transfer Coefficient (Btu/h·ft²·°F)</u>
0.	1.2E+00
1.00E-03	2.0E+02
1.80E+01	2.0E+02
7.20E+01	1.0E+02
2.40E+02	7.5E+01
4.80E+02	5.0E+01
4.56E+03	5.0E+01

TABLE B-10

NATURAL CONVECTION HEAT TRANSFER COEFFICIENTS FOR NITROGEN AT 100% DENSITY. USED FOR THE REACTOR BUILDING ATMOSPHERE THAT CONTAINS OXYGEN. CONSERVATIVE AFTER THE OXYGEN IS BURNED OUT.

<u>Temperature (°F)</u>	<u>Heat Transfer Coefficient (Btu/h·ft²·°F)</u>
0	1.128
100	1.128
300	1.744
500	2.211
700	2.638
900	3.051
1100	3.493
1300	4.058
1500	4.907
1700	6.257

TABLE B-11

NITROGEN NATURAL CONVECTION COMBINED WITH INSULATION HEAT CONDUCTION. THE NATURAL CONVECTION COEFFICIENTS ARE FOR OUTSIDE AIR WITH NO WIND. THE HEAT CONDUCTION IS FOR INSULATION THAT MELTS OVER THE RANGE 600 to 1300 °F. THESE ARE TO BE USED OUTSIDE THE BUILDING ROOF AND WALLS

<u>Temperature (°F)</u>	<u>Heat Transfer Coefficient (Btu/h·ft²·°F)</u>	<u>Insulation Thermal Conductivity (Btu/h·ft·°F)</u>
76	1.2	0.013
650	1.2	0.028
950	1.2	0.052
1250	1.2	0.104

Combined Coefficient

<u>Temperature (°F)</u>	<u>Heat Transfer Coefficient (Btu/h·ft²·°F)</u>
60	.14
650	.25
950	.41
1250	.61
1800	1.2

V. MASS TRANSFER COEFFICIENTS

The code analysis of leakage based on leak rate specifications uses the following equation.

$$L_c = C_t \sqrt{(\rho \Delta P)}$$

Where: L_c = Leak rate, lb/hr of cavity or RCB atmosphere.

C_t = Leakage coefficient, code input data, lb/h. $\sqrt{\text{psid} \cdot \text{lb/ft}^3}$

ρ = Atmosphere density, lb/ft³

ΔP = Differential pressure, psid

Leakage according to this equation can be either direction. The coefficients used with this equation are as follows:

- A. Cavity Leakage - The cavity leakage was based on blowing open some seals around the reactor head.

<u>Differential Pressure</u> (psig)	<u>Leakage Coefficient</u> (C_t)
0	2000
0.6	4000
1.2	6000
30	15000
50	15000

- B. Dome-To-RCB Leakage (for double containment and containment-confinement cases only) - The dome-to-RCB leak rate was sized at 100 vol%/day at 15 psig (183 lb/hr at 15 psig). This was achieved by using a coefficient, constant with pressure, of $C_t = 122 \text{ lb/hr} \cdot \sqrt{\text{psid} \cdot \text{lb/ft}^3}$. For most cases, it was assumed that pressures above 15 psid operated a pressure

relief valve which limited the pressure to 15 psid. The coefficients used for these cases are listed below:

<u>Differential Pressure</u> <u>(psid)</u>	<u>Leak Coefficient</u> <u>(C_t)</u>
0	122
15	122
16	12200
18	122000
50	122000

- C. RCB-To-Outside - All high pressure RCB's were assumed to leak at 0.1 vol%/day at 10 psig. The leakage coefficient for this leak rate is $C_t = 10.5 \text{ lb/h.} \sqrt{\text{psid/lb}\cdot\text{ft}^3}$. For cases that required a pressure relief valve the following table was used:

<u>Differential Pressure</u> <u>(psid)</u>	<u>Leak Coefficient</u> <u>(C_t)</u>
0	10.5
10	10.5
11	10500
12	105000
20	105000

- D. Containment-Confinement RCB - The leakage from the confinement type RCB to the outside world was sized to leak air into the RCB at 15,000 scfm (1000 volume %/day) when the RCB was held at -1.5 in. H₂O pressure by connecting it to an infinitely large cell whose pressure was defined as a negative 5.5 in. H₂O. The leak coefficients between the RCB and the infinite volume sized to maintain a flow of 15000 scfm into the infinite

volume under normal circumstances. If the pressure in the RCB increased (went positive) the "leak" would increase to 1600 Vol%/day. Tables B-12 and B-13 show the coefficients used to accomplish this balance.

TABLE B-12

OUTSIDE-TO-RCB LEAK COEFFICIENTS

Differential Pressure (psid)	Leak Coefficients (C_t)
0	1.1×10^6
10	1.1×10^6

TABLE B-13

RCB-TO-INFINITE VOLUME LEAK COEFFICIENTS

Differential Pressure (psid)	Leak Coefficients (C_t)	
0	6.75E+05	
0.144	6.75E+05	
0.1715	9.92E+05	
0.2166	8.84E+05	
0.2888	7.67E+05	
0.361	6.88E+05	
0.4332	6.3E+05	
0.5415	5.65E+05	
0.699	0.0	} Exhaust Failure
10.0	0.0	

Depressurization of a containment volume was accomplished by use of the CACECO nozzle equation. This equation is written below:

$$L_n = 346,538 C_n A_n \sqrt{\rho \Delta P}$$

where: L_n = Leak rate, lb/hr

C_n = Nozzle coefficient, code input data, nominally 0.817

A_n = Nozzle area, code input data, ft²

ρ = Atmosphere density, lb/ft³

ΔP = Differential pressure, psid

In all cases, the size of the nozzle was large enough to allow depressurization within one hour, but not so large as to create instabilities in the operation of the code.

After depressurization, the nozzle was generally left unchanged unless pulsing occurred. Typical values of the nozzle area were Cavity-to-RCB (or inner dome), 0.25 - 0.50 ft²; Inner Dome-to-RCB, 0.25 to 0.50 ft²; RCB-to-Outside, 1.0 ft².

VI. CELL CHARACTERISTICS

The code was used to analyze two-cell, three-cell, and four-cell configurations, depending on the containment system studied. Single containment systems involved only a cavity and an RCB. Double containment systems involved, in addition, a sealed head compartment. The containment-confinement cases involved yet an additional volume, the exhaust volume, described in Section V of this Appendix. The sizes and initial conditions in these cells are listed in Table B-14.

TABLE B-14
CELL CHARACTERISTICS

<u>Cell</u>	<u>Name</u>	<u>Free Volume (ft³)</u>	<u>Area for Pool (ft²)</u>	<u>Initial Pressure (psi)</u>	<u>Initial Temp. (°F)</u>	<u>Oxygen (%)</u>
A	Reactor Cavity	3x10 ⁴	1.256x10 ³	14.7	150	2
B	Sealed Head Compartment	2.93x10 ⁴	1.256x10 ³	14.7	150	2
C	RCB	2.25x10 ⁶	8.6x10 ³	14.7 (14.65 for Confinement RCB)	76	21
D	Exhaust "Room"	1.10 ⁴⁰	0.0	14.50	76	21

REFERENCES:

1. R. D. Peak and D. D. Stepnewski, Computational Features of the CACECO Containment Analysis Code, TANSO 21-274, 1975.
2. R. D. Peak, Users Guide to CACECO Containment Analysis Code, HEDL-TC-859, Hanford Engineering Development Laboratories, May 1977, Richland, WA.

APPENDIX C
INPUTS USED FOR HAA
CALCULATIONS

APPENDIX C

INPUTS USED FOR HAA CALCULATIONS

This Appendix lists input numbers that were typical to HAA runs made for this study. A more detailed description of the input and how it is used by HAA is given in Reference 1. The sensitivity of the output to input assumptions is discussed in Reference 2.

Item	Value	Description
ISETL	1	-Yes, Calculate Settling Term.
IPLAT	1	-Yes, Calculate Plating Term.
ILEAK	1	-Yes, Calculate Leakage Term.
IAGGB	1	-Yes, Calculate Brownian Motion Agglomeration Term.
IAGGG	1	-Yes, Calculate gravity agglomeration Term.
ISORS	1	-Yes, source term supplied.
INDIN	0	-Use Adams Moulton Intergration with variable incrementing.
KMAX	1000	-Maximum number of time steps.
ISEQ	6	-Increment of time step at which output is desired
IVOL	0	-No distribution calculations
IDEL	0	-Use input DELTA
S ϕ	0	-Initial Source rate, particles/(cm ³ sec)
SIGSOR	8.0	-Source geometric mean deviation (volume), microns **3
VSOR	0.125	-Source 50% radius cubed, microns **3
XIN(1)	100, 7.4x10 ⁷ or 5.68x10 ⁹	-Initial concentration of aerosol, particles/cm ³ . -At the beginning of each melt-through - only case XIN(1) was set to 100. For single-containment head-expulsion cases, XIN(1) was set at 7.4x10 ⁷ . For systems having a sealed head access compartment, XIN(1) was set at 5.68x10 ⁹ . For restart at later times XIN(1) was linearly extrapolated from the previous run to the restart time.

Item	Value	Description
SIGAIR	8.0	-Aerosol geometric mean deviation (volume), microns **3. For restart cases, SIGAIR for all cases extrapolated to 27.
VAIR	0.125	-Aerosol 50% radius cubed, microns **3. For restart cases, VAIR for cases extrapolated to ~216.
HIN	0.01	-Initial time step (seconds)
TOLIN	0.001	-Convergence criteria
RVL	0	-Leakage rate, per second.
AFOV	2.78×10^{-4}	-Ratio of floor area to chamber volume (cm^{-1}) used for RCB calculations. Inner containment (dome over head access compartment) used a value of 1.4×10^{-3} .
RHO	2.27	-Density of aerosol material, gm/cc.
VISC	()	-The viscosity (dyne sec/ cm^2) and temperature ($^{\circ}\text{K}$) values were averaged numbers over the lengths
TEMP	()	of the run. Our results were not sensitive to reasonable variations in these numbers.
DELTA	1×10^{-5}	-Constant, cm, used in plating equation
EPSL	1×10^{-6}	-Criteria for termination of problem.
VOL	()	-Volume (cm^3) of chamber. For the RCB, the value used was 6.37×10^{10} . For the sealed head access chamber cases the value used was 8.3×10^8 .
EPSN	6×10^{-3}	-Ratio where log-normal assumptions break down and SSM (stirred settling model) calculations begin.
EPSM	()	-Time for SMM shutoff. Set equal to boil-dry time for all cases.
ALPHA	0.25	-Density modification factor
KLIACH	0	-Use Stokes velocity correlation
EFF	1	-Gravitational collision efficiency
TSOUR(I), (), ()		-Time and source rate pairs for sodium oxide
SOURS(I)		inrate to chamber of interest. Calculated from CACECO output data (part/cc·sec)

Item	Value	Description
TLEK(I),	(),()	-Time and Leak rate fraction pairs for leak table from chamber of interest. Same as CACECO output (sec^{-1})
RLEX(I)		
IVOL	0.0	-Secondary Containment Volume (cc)

REFERENCES:

1. R. Hubner et. al., HAA-3 User Report, AI-AEC-13038, Atomics International, Canoga Park, CA, March 1973.
2. T. C. Huang and R. K. Stitt, Review and Evaluation of Current Aerosol Models for LMFBR Safety Analysis, GEAP-14054, June 1975, General Electric Company, Sunnyvale, CA.

DISTRIBUTION

UC-79 (206)
UC-79i (24)
UC-79p (38)

DOE-RL (2)

Manager
Chief Patent Attorney

DOE/RRT-HQ (6)

Director, Program Division
Assistant Director for Reactor Safety
Chief, Fast Reactor Safety
H. Alter
A. Opdenaker
S. Berk

DOE/FFTF-PO (3)

Director
Reactor Safety Field Office
Al Farabee

DOE/CRBRPP0 (1)

E. Chapple

HEDL (28)

GR Armstrong (2)	W/C-78
JM Atwood	W/C-45
T. Chiao	W/A-39
G. Franz	W/C-26
JW Hagan	W/C-78
JP Hale	W/C-76
RK Hilliard	W/221-T
RL Jensen	W/C-75
HC Martin	W/C-76
LD Muhlestein	W/A-11
RD Peak	W/C-75
SE Seeman (2)	W/C-78
DE Simpson	W/C-80
DD Stepnewski	W/C-75
Central Files (10)	W/C-110
Publications Services (2)	W/C-115

SPECIAL DISTRIBUTION

Westinghouse Advanced Reactors Division (6)

Westinghouse Electric Corporation

P. O. Box 158

Madison, PA 15230

J. R. Graham (4)

P. Bradbury

C. A. Anderson

Atomics International (2)

P. O. Box 309

Canoga, CA 91304

H. Morewitz

R. W. Keaton

General Electric FBRD (3)

310 De Guigne Dr.

Sunnyvale, CA 94086

J. McDonald

S. M. Davies

M. Temme

Argonne National Laboratory (1)

9700 S. Cass Ave.

Argonne, IL 60439

L. Baker, Jr.

Los Alamos Scientific Laboratories (1)

P. O. Box 1663

Los Alamos, NM 87545

J. Boudreau