IDO-16981
Preliminary Edition
Special Distribution
Issued: April, 1964

PRELIMINARY SAFETY ANALYSIS REPORT
LOFT FACILITY

Editor

J. M. Waage

Authors

F. W. Ahrens
G. A. Dinneen
R. D. Fletcher
O. M. Hauge
D. J. Hine
G. B. Homer

N. R. Horton
S. E. Jensen
G. B. Matheney
W. J. Neal
H. A. Traue
J. M. Waage

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.

ACKNOWLEDGMENT

Valuable assistance in the preparation of this report was given by STEP-TAN Design Engineering; by the USAEC Health and Safety Branch; by the U. S. Weather Bureau; and by the architect-engineer, Kaiser Engineers. The authors also wish to acknowledge Mr. George Wehmann, USAEC-ID, for his continuing surveillance of the various efforts during the preparation of this document.

SUMMARY

As part of the Atomic Energy Commission's nuclear safety programs, engineering tests will be carried out at the National Reactor Testing Station to investigate the consequences of a loss of coolant accident with a pressurized water power reactor. These tests will be conducted in a new facility to be constructed and which will be designated as LOFT (Loss of Fluid Test). A preliminary safety analysis has been performed to evaluate the potential hazards of conducting this test program with the emphasis placed on determining the radiological hazards associated with the planned tests and on determining the potential hazards of the maximum credible accident. The basis for the analysis was the loss of coolant from a pressurized water reactor which had been operating at 50 Mw(t) in this facility for 800 continuous hours. It was assumed that the reactor was operated and the test conducted during the winter months when pasture grazing by cattle is relatively low.

The safety analysis of the planned loss of coolant test is based upon estimates of fission product release, transport, and diffusion which could reasonably be predicted from existing experimental data. The meteorology at the NRTS is well known and has been used in computing the downwind radiological effects. The results of this analysis indicate that the only radiological dose in excess of the limits set for normal release to the atmosphere is that which can possibly occur to the thyroid gland of a child from the ingestion of radioiodine if no precautionary measures are taken. This dose results from the consumption of contaminated milk obtained from cattle grazing in pasture areas where the iodine has deposited. This dose will have its maximum value at the site boundary in the Howe area and is based upon the assumption that the test is allowed to continue for five days before being terminated.

Monitoring devices will be provided both inside and outside the LOFT containment building to assure early detection of a fission product release that could result in radiological doses above tolerance. A spray system is also provided that will permit sufficiently rapid pressure reduction in the containment vessel to reduce possible releases to the atmosphere to a safe level.

In addition, procedures have been developed by ID-AEC which are available to insure that acceptable levels are not exceeded. These procedures, as they apply to control of off-site doses, include the purchase of milk, limitations on cattle grazing and personnel evacuation.

The safety analysis of the maximum credible accident is based upon the postulation of a pipe break in both the primary and secondary coolant systems during adverse meteorological conditions. It is further assumed that a missile is generated as the result of the rupture with sufficient energy to penetrate the containment vessel. The same assumptions used in the analysis of the planned test were used to describe the fission product release and transport to the containment; however, the effects of pressure reduction sprays were included in estimating the release to the atmosphere. The results of this analysis indicate that the doses that could be received at the nearest site boundary are below the guide limits set for accidental releases to the atmosphere.

The report presents, in addition to the radiological analyses, brief discussions of the experimental program and design philosophy and descriptions of the experimental assembly and LOFT facility. In addition, the safety analysis of the power operation and post-test phases of the program are summarized. It has been concluded that the loss of coolant test program can be carried out safely in the LOFT facility as presently conceived and currently located.

PRELIMINARY SAFETY ANALYSIS REPORT
LOFT FACILITY

CONTENTS

SUMMARY	111
I. INTRODUCTION	I-1
II. TEST PROGRAM	II-1
A. Introduction	II-2
B. Test Outline	II-2
1. Phase I - Containment Vessel Leakage and Pressure Tests	II-2
2. Phase II - Coolant Blowdown Tests in the Containment Vessel with a Dummy Core	II-3
3. Phase III - Power Operation	II-6
4. Phase IV - Loss of Coolant Test with a Radioactive Core	II-10
C. Termination of the Experiment	II-16
1. Pressure Reduction and Fission Product Removal	II-16
2. Decontamination	II-17
D. Post-Test Examination	II-18
III. DESIGN CONSIDERATIONS	III-1
A. Design Parameters	III-4
B. Design Responsibilities	III-11
IV. EXPERIMENTAL ASSEMBLY DESCRIPTION	IV-1
A. General	IV-1
B. Reactor Description	IV-1
1. Pressure Vessel	IV-1
2. Vessel Internals	IV-4
3. Reactor Core	IV-4
4. Control Rods and Drives	IV-5
5. Shield Tank	IV-5
6. Summary of Design Features	IV-6
C. Heat Removal Systems	IV-6
1. Primary Coolant System	IV-6
2. Emergency Coolant System	IV-14
D. Instrumentation	IV-15
1. Nuclear Instrumentation and Control	IV-15
2. Process	IV-23
3. Experimental Instrumentation	IV-25
4. Containment Vessel	IV-35
V. NUCLEAR DESIGN	V-1
A. Reactor Physics	V-1
B. Reactor Control	V-6
C. Gamma Heating	V-6

VI.	SITE DESCRIPTION	VI-1
A.	General	VI-1
B.	Environmental Conditions	VI-1
	1. Geography	VI-1
	2. Meteorology	VI-2
	3. Geology	VI-4
	4. Seismology	VI-4
C.	Population Distribution	VI-5
	1. On-Site	VI-5
	2. Off-Site	VI-5
D.	Land Usage	VI-8
E.	Radiological Surveillance	VI-8
VII.	FACILITY DESCRIPTION	VII-1
A.	General	VII-1
B.	Buildings and Facilities	VII-1
	1. Containment Building	VII-1
	a. Access Opening	VII-3
	b. Penetrations	VII-4
	c. Shielding	VII-5
	d. Building Equipment	VII-10
	2. Services Building	VII-12
	a. The Remote Control Room Structure	VII-12
	b. Personnel Entrance	VII-13
	c. Decontamination Equipment and Utility Room	VII-14
	d. Electrical Equipment Room	VII-14
	e. Sample Room	VII-14
	3. Support Facilities	VII-15
	a. Examination Area	VII-15
	b. Transportation System	VII-18
	4. Experimental Facilities Systems	VII-19
	a. Secondary Coolant System	VII-19
	b. Sampling Systems	VII-26
	c. Documentary Systems - Cameras	VII-33
	d. Pressure Reduction Sprays	VII-34
	e. Decontamination System	VII-35
	f. Halogen and Particulate Removal System	VII-36
	g. Radiological Grid	VII-38
	h. Controlled Leakage	VII-40
	i. Leakage Test Systems	VII-40
	5. Utility Systems	VII-44
	a. Water Systems	VII-44
	b. Waste Systems	VII-47
	c. Plant and Instrument Air System	VII-51
	d. Heating and Ventilating System	VII-51
	e. Steam System	VII-54
	f. Electrical Power	VII-54
	g. Communication and Alarm Systems	VII-57

VIII.	SAFETY ANALYSIS CRITERIA	VIII-1
A.	General	VIII-1
B.	Criteria	VIII-1
1.	Operator Action	VIII-1
2.	Credibility of Accidents	VIII-2
3.	Criteria for Design	VIII-7
4.	Philosophy	VIII-10
5.	Divided Control Room Philosophy	VIII-12
6.	Emergency Power System	VIII-12
7.	Critical Systems and Equipment	VIII-13
IX.	PRELIMINARY ANALYSIS OF REACTOR OPERATION	IX-1
A.	Basic Operating Philosophy	IX-1
B.	General Safety Considerations	IX-1
C.	General Hazards	IX-2
D.	Reactivity Accidents	IX-4
1.	Startup Accidents	IX-4
2.	Control Rod Withdrawal at Power	IX-4
3.	Control Rod Drop Accident	IX-5
4.	Cold Water Accident	IX-5
E.	Mechanical Failures	IX-7
1.	Fuel Cladding Failures	IX-7
2.	Loss of Coolant Flow Accidents	IX-7
3.	Secondary Coolant Steam Line Break Accident	IX-7
4.	Steam Generator Tube Rupture Accident	IX-8
5.	Loss of Coolant Accident	IX-9
F.	Radiological Hazards During Normal Operation	IX-9
X.	ENGINEERING ANALYSIS OF THE LOSS OF COOLANT TEST	X-1
A.	General	X-1
B.	Pre-Test Checkouts	X-1
1.	Containment Leakage	X-2
2.	Pressure Reduction Sprays	X-2
3.	Exhaust Filter System	X-3
4.	Fission Product Sampling System	X-3
5.	Services Building Ventilation	X-3
C.	General Test Sequence	X-3
D.	Administrative Controls	X-5
E.	Coolant Blowdown	X-6
1.	Introduction	X-6
2.	Effects of Coolant Blowdown on Primary System Components	X-7
3.	Effects of Coolant Blowdown on the Reactor Dolly	X-14
4.	Effects of Coolant Blowdown on the Containment Building	X-14
F.	Core Meltdown	X-16
1.	General	X-16
2.	Core Heatup Following Blowdown - Phase I	X-17
3.	Core Rearrangement - Phase II	X-23
4.	Final Core Geometry - Phase III	X-25

G.	Fission Product Release, Transport, and Plateout Phenomena	X-31
	1. Radioiodine Release from Fuel	X-32
	2. Transport to the Containment Building	X-34
	3. Plateout Inside the Containment Vessel	X-36
H.	Containment Vessel Leakage	X-38
I.	Meteorological Conditions During the Loss of Coolant Test	X-41
J.	Test Termination	X-42
XI.	RADIOLOGICAL ANALYSIS OF THE LOSS OF COOLANT TEST	XI-1
A.	Analysis	XI-1
	1. Assumptions	XI-1
	2. Analytical Techniques	XI-2
B.	Conclusions	XI-7
XII.	SAFETY ANALYSIS OF THE MAXIMUM CREDIBLE ACCIDENT	XII-1
A.	Identification of the Maximum Credible Accident	XII-1
B.	Radiological Analysis	XII-2
C.	Conclusions	XII-4
XIII.	POST-TEST OPERATIONAL SAFETY CONSIDERATIONS	XIII-1
A.	Radiological Safety Analysis	XIII-1
	1. Direct Radiation	XIII-1
	2. Contamination	XIII-2
B.	Secondary Criticality	XIII-3
	1. Removal and Transportation to the Examination Area	XIII-3
	2. Examination Area	XIII-3
XIV.	CONCLUSIONS	XIV-1

FIGURES

III-1	LOFT Schedule	III-13
IV-1	LOFT Dolly Elevation and Plan View	IV-36
IV-2	LOFT Vessel Cross Section	IV-37
IV-3	LOFT Nozzle Summary	IV-38
IV-4	LOFT Vessel Sections	IV-39
IV-5	LOFT Fuel Element Assembly	IV-40
IV-6	Primary System Flow Diagram	IV-41
IV-7	Primary System Instrumentation Diagram	IV-42
IV-8	Reactor Safety Instrumentation Diagram	IV-43
IV-9	Reactor Nuclear and Control Instrumentation Diagram .	IV-44
IV-10	Experimental Test Instrumentation	IV-45
V-1	LOFT Core Quarter Section	V-8
V-2	LOFT Source Ratio versus Peripheral Enrichment	V-9
V-3	LOFT Eigenvalues versus Peripheral Enrichment	V-10
V-4	LOFT Outer Enrichment versus Inner Enrichment	V-11
V-5	LOFT K_{eff} versus Fuel Temperature (Moderator @ 545°F). .	V-12
V-6	LOFT K_{eff} versus Core Temperature (Zero Power)	V-13
V-7	LOFT K_{eff} versus Fuel Temperature (Moderator @ 68°F) .	V-14
V-8	LOFT Thermal Flux versus Distance from Core	V-15
V-9	LOFT Gamma Heating versus Distance from Core	V-16
VI-1	National Reactor Testing Station	VI-10
VI-2	Test Area North, NRTS	VI-11
VI-3	Topographical Cross Section Through NRTS	VI-12
VI-4	NRTS and Southeastern Idaho	VI-13
VI-5	Population Distribution and Wind Frequencies About the Test Area North, NRTS	VI-14
VI-6	TAN 20 foot Level Lapse Wind Roses	VI-15
VI-7	TAN 20 foot Level Inversion Wind Roses	VI-16

VI-8	TAN 20 foot Level Wind Roses	VI-17
VI-9	TAN 150 foot Level Lapse Wind Roses	VI-18
VI-10	TAN 150 foot Level Inversion Wind Roses	VI-19
VI-11	TAN 150 foot Level Wind Roses	VI-20
VI-12	NRTS Vegetation	VI-21
VI-13	Irrigation in Southeastern Idaho	VI-22
VI-14	NRTS Environmental Monitoring	VI-23
VII-1	Containment Building - Outside Elevation	VII-59
VII-2	Containment Building - South Sectional Elevation . . .	VII-60
VII-3	Containment Building - West Sectional Elevation . . .	VII-61
VII-4	Containment Building - Basement Plan	VII-62
VII-5	Railroad Access Door	VII-63
VII-6	Railroad Door Seal System	VII-64
VII-7	Containment Building - Ground Floor Plan	VII-65
VII-8	LOFT-FET Plot Plan	VII-66
VII-9	LOFT Services Building	VII-67
VII-10	LOFT Containment Building Shielding Labyrinth	VII-68
VII-11	Railroad Turntable Modifications	VII-69
VII-12	TAN Shielded Locomotive	VII-70
VII-13	Secondary Coolant System Flow Diagram	VII-71
VII-14	Secondary Coolant System Instrument Diagram	VII-72
VII-15	Spray System Flow Diagram	VII-73
VII-16	Particulate and Halogen Removal System	VII-74
VII-17	Heating and Ventilating System Flow Diagram	VII-75
VII-18	LOFT Radiological Grid System	VII-76
VII-19	Electrical Power Distribution to Inner Radiological Grid	VII-77
VII-20	Electrical Power Distribution to Outer Radiological Grid	VII-78

VII-21	Controlled Leakage System Schematic	VII-79
VII-22	LOFT Service Water Distribution Schematic	VII-80
VII-23	Demineralized Water Flow Diagram	VII-81
VII-24	Fire Protection Water Flow Diagram	VII-82
VII-25	Liquid Waste Disposal System	VII-83
VII-26	Maximum Discharge of Radioisotopes to the Ground . . .	VII-84
VII-27	Plant and Instrument Air System	VII-85
VII-28	Steam System Flow Diagram	VII-86
VII-29	FET Electrical Distribution Modifications	VII-87
VII-30	LOFT Electrical Distribution System	VII-88
IX-1	Direct Radiation from the Containment Vessel During Operation	IX-12
X-1	Pressure in Reactor Vessel During Blowdown	X-43
X-2	Relative Coolant Mass in Reactor Vessel During Blowdown	X-44
X-3	Typical Results from the Subcooled Blowdown Study . .	X-45
X-4	Decay Heat Generation Rate versus Shutdown Time for LOFT (50 Mw Operation for 400 Hours)	X-46
X-5	Mean View Factor versus Row Number for LOFT Core . . .	X-47
X-6	Core Centerline Temperature versus Time after Rupture	X-48
X-7	Temperature Rise versus Time after Rupture (Effect of Power Density)	X-49
X-8	Time for Average Axial Fuel to Reach the Melting Temp- erature of the Clad (Max/Avg Radial Power Density = 1.6)	X-50
X-9	Bottom Surface Temperature of Restraining Grid versus Contact Time	X-51
X-10	Core and Lower Support Plate Temperature versus Contact Time	X-52
X-11	Core and Lower Support Plate Temperature Profiles at Various Times	X-53
X-12	Reactor Vessel Bottom Head Temperature Profiles . . .	X-54
X-13	Maximum Fuel Temperature versus Time for Various Core Thicknesses on the Bottom Head	X-55

X-14	Average Fuel Temperature versus Time for Various UO ₂ Thermal Conductivities	X-56
X-15	Iodine Release and Availability as a Function of Time	X-57
X-16	Iodine Release as a Function of Temperature	X-58
X-17	Leakage Rate from the Containment Vessel	X-59
X-18	Pressure Decay in the Containment Building	X-60
XI-1	Area Surrounding LOFT Showing Segments Used for Radiological Analysis	XI-10
XI-2	Fission Product Activity as a Function of Time after Shutdown	XI-31
XI-3	Particulate Fission Product Activity as a Function of Time after Shutdown	XI-32
XI-4	Halogen Fission Product Activity as a Function of Time after Shutdown	XI-33
XI-5	Integrated Gamma Dose from the Containment Shell . . .	XI-34
XII-1	Pressure Decay in the Containment Building with Six Square Inch Penetration	XII-6
XII-2	Pressure Decay in the Containment Building with Three Square Inch Penetration	XII-7
XII-3	Containment Leakage through Three Square Inch Penetration	XII-8
XII-4	Fission Product Activity as a Function of Time after Shutdown	XII-9
XII-5	Halogen Fission Product Activity as a Function of Time after Shutdown	XII-10
XII-6	Containment Leakage through Six Square Inch Penetration	XII-11
XIII-1	Direct Radiation from the Containment Vessel	XIII-5

TABLES

II-A	Fission Product Inventory Following 100, 400, 800, and 1600 Hours of Operation	II-8
III-A	Summary of Reactor Characteristics	III-3
IV-A	Leakage Flow Requirements for Heat Removal	IV-6
IV-B	LOFT Core Design Features	IV-7
IV-C	Primary System Coolant Volume	IV-9
IV-D	LOFT Operational and Experimental Instrumentation . .	IV-29
V-A	Eigenvalues as a Function of Inner and Peripheral Fuel Enrichment	V-4
V-B	LOFT Eigenvalues and Peak-to-Average Power Density - Clean Core	V-6
VI-A	Population Distribution of NRTS	VI-6
VI-B	Off-Site Population Distribution	VI-7
X-A	Decompression Pressures	X-12
X-B	Latent Heats for Stainless Steel Constituents	X-30
XI-A	Distribution and Relative Locations of Various Towns and Facilities with Respect to LOFT	XI-9
XI-B	Radiation Dose in North-Northeast Segment - Five Day Release Period	XI-11
XI-C	Radiation Dose in Northeast Segment - Five Day Release Period	XI-12
XI-D	Radiation Dose in East-Northeast Segment - Five Day Release Period	XI-13
XI-E	Radiation Dose in East-Southeast Segment - Five Day Release Period	XI-14
XI-F	Radiation Dose in Southeast Segment - Five Day Release Period	XI-15
XI-G	Radiation Dose in South-Southeast Segment - Five Day Release Period	XI-16
XI-H	Radiation Dose in South Segment - Five Day Release Period	XI-17
XI-I	Radiation Dose in South-Southwest Segment - Five Day Release Period	XI-18
XI-J	Radiation Dose in Southwest Segment - Five Day Release Period	XI-19

XI-K	Radiation Dose in West-Southwest Segment - Five Day Release Period	XI-20
XI-L	Radiation Dose in West-Northwest Segment - Five Day Release Period	XI-21
XI-M	Radiation Dose in Northwest Segment - Five Day Release Period	XI-22
XI-N	Radiation Dose in North-Northwest Segment - Five Day Release Period	XI-23
XI-O	Radiation Dose in North Segment - Five Day Release Period	XI-24
XI-P	Meteorological Conditions and Duration of Each for Five Day Test Period	XI-25
XI-Q	Correction Factor to Account for Cloud Meandering Over a $22-1/2^{\circ}$ Sector	XI-25
XI-R	Correction Factor to Account for Wind Variation During Five Day Release Period	XI-27
XI-S	Correction Factor - Product of Wind Persistency and Cloud Meandering Factors	XI-28
XI-T	Fission Products and Organs Considered for Inhalation Dose Calculations	XI-29
XII-A	Radiation Doses for MCA	XII-5

SECTION I

INTRODUCTION

I. INTRODUCTION

A Safety Test Engineering Program (STEP) has been established by the Atomic Energy Commission for the purpose of conducting engineering and safety tests which will demonstrate the behavior of nuclear systems under accident conditions and provide information regarding the consequences of nuclear accidents. This effort is part of the nuclear safety programs sponsored by the Division of Reactor Development. Phillips Petroleum Company has been assigned the responsibility for conducting the experimental program which is to be carried out at Test Area North (TAN) of the National Reactor Testing Station using existing facilities and a new Loss of Fluid Test (LOFT) facility.

Because of prominence of water-cooled and -moderated reactors in the United States power program, the initial experimental investigations in the LOFT facility will be directed toward providing safety information regarding this type of reactor, in particular the consequence of a loss of coolant accident will be demonstrated. The proposed program⁽¹⁾ embodies the testing of a complete nuclear system typical of pressurized water power reactors under conditions which simulate as closely as possible those which might be present in an actual loss of coolant accident.

The purpose of this report is to present the results of a preliminary safety analysis performed at the request of the Atomic Energy Commission. This safety analysis is based on the preliminary LOFT facility design and on the test program presently envisioned. The LOFT reactor and facility is similar in design features and operating characteristics to the general class of pressurized-water, UO_2 -fueled reactors; therefore, the safety considerations associated with normal power operation of the LOFT reactor are not significantly different from those generally associated with other reactors of this type. This safety analysis is therefore concerned principally with the analysis of the hazards attendant to carrying out the planned loss of coolant test and

(1) T. R. Wilson, O. M. Hauge, and G. B. Matheney, "Feasibility and Conceptual Design for the STEP Loss of Coolant Facility", IDO-16833 Revision I (April 1963).

the analysis of potential hazards associated with the postulated maximum credible accident. In addition, the experimental program, operating philosophy, and the analysis of the power operation and post-test phases of the program are summarized herein.

The results of the analysis performed demonstrate that the LOFT experimental program can be conducted at the NRTS using the reactor and facility design as presently conceived without undue risk to site personnel or to the general public.

SECTION II

TEST PROGRAM

II. TEST PROGRAM

A. Introduction

The engineering test program proposed for investigating the consequences of a loss of coolant accident involving a water-cooled and -moderated power reactor is described in this section. The safety tests will be carried out on a complete, large scale pressurized water reactor plant including a containment building and the plant equipment normally associated with a power reactor of this type. The objectives of this program are: (1) to conduct tests which will demonstrate the behavior of the reactor system and plant equipment with respect to safety under simulated accident conditions, (2) to provide quantitative information which will assist in more accurately defining the hazards that may be associated with the loss of coolant accident. Specifically, emphasis will be placed on providing information concerning the following: (1) the fraction of the fission product inventory released from the fuel, (2) the nature and magnitude of the fission products reaching the containment building and the efficiency of the inherent removal mechanisms, and (3) the fraction of fission product inventory which escapes from the containment building.

In order to achieve these objectives, a test program consisting of four separate phases is envisioned. These phases are as follows: Phase I - Containment Vessel Leakage and Pressure Tests; Phase II - Coolant Blowdown Tests in the Containment Vessel with a Dummy Core; Phase III - Power Operation; and Phase IV - Loss of Coolant Test with a Radioactive Core. Following the final phase of the program the fission products will be removed from the containment vessel by recirculation of the containment atmosphere through a filter system and be remotely decontaminating the internal surfaces of the building. The nuclear system will then be removed to the TAN hot shop for post-test examination.

Phase I and II of the experimental program consists of preliminary tests to provide the information needed to interpret and apply the results of the final loss of coolant test and to provide information which will assist in determining the rupture size and location that provides the maximum fission product release from the core and transport

within the containment vessel. To the extent possible, the test conditions for the final test will be selected to maximize the fission product release and transport. Final selection of the test conditions is therefore dependent upon the analytical studies currently underway and the results of preliminary experiments.

A general description of each phase of the test program and the information to be obtained as envisioned at this time are described in this section.

B. Test Outline

1. Phase I - Containment Vessel Leakage and Pressure Tests

This phase of the experimental program is directed at investigating the containment leakage characteristics. A series of tests will be performed to insure that the leak rate from the containment vessel meets the design specifications and to determine the leak rate as a function of pressure. These tests are expected to provide a high degree of assurance that the containment vessel will withstand the pressures associated with primary coolant blowdown as well as providing data on the leak rate as a function of pressure. This latter information is needed for the final assessment of the radiological hazards and for interpretation of the results of the final loss of coolant test.

The first series of leak tests are generally considered as acceptance tests and will be performed by the construction contractor. They will include:

- (1) locating and repairing gross leaks,
- (2) pneumatic pressure tests at 48 psig (120% of design pressure),
and
- (3) leakage rate determination at the design pressure of 40 psig.

These tests will continue until a leakage rate of 0.2% or less of the containment free volume per day has been attained.

The second series of leak tests are designed to investigate the containment leakage characteristics as a function of pressure and to provide some early information on the radioiodine retention characteristics of the containment vessel. These tests include:

- (1) the evaluation of the instrumentation for measuring containment leakage,
- (2) determining the leakage rate as a function of containment pressure, and
- (3) the release of trace quantities of radioiodine to determine the retention characteristics of the containment vessel.

Consideration is being given to collecting and monitoring the leakage through penetrations and to the possibility of installing a controlled leakage path in the vessel. The monitored penetrations would permit a measurement to be made of the leakage rate through each penetration as well as the type and quantity of fission products that escape from the vessel. The fission products leaking through the penetrations would be captured (except for noble gases) either by a separate filter system specifically provided for the purpose or by the building exhaust filter system.

2. Phase II - Coolant Blowdown Tests in the Containment Vessel With a Dummy Core

Following completion of the containment pressure and leak tests, the reactor system containing a dummy core will be installed in the containment vessel and a series of approximately 20 primary coolant blowdown tests will be performed. The purpose of these tests are:

- (1) to determine the coolant blowdown characteristics and fission product transport as a function of rupture size and location,
- (2) to determine the coolant blowdown characteristics and fission product transport as a function of primary coolant temperature and pressure,
- (3) to determine the effect of blowdown on the integrity of the core, primary coolant system, and containment vessel,
- (4) to measure the effectiveness of the containment vessel spray system for reducing the internal vessel pressure,
- (5) to evaluate the equipment to be used for simulation of a rupture,

- (6) to check the reliability and performance of the instrumentation under the atmospheric conditions that will prevail during and following the final loss of coolant test (Phase IV).

These blowdown tests will be performed on the complete nuclear system, except for the core. A dummy core, identical in size and structural rigidity to the actual core will be installed in the system to provide the same hydraulic and mechanical characteristics that will prevail during the Phase IV test in this series, a core support structure which is designed to withstand the hydrodynamic forces associated with rapid depressurization of the system will be used. The last blowdown test will employ a core support structure designed in accordance with the engineering practices currently used in the nuclear industry to determine if a structure so designed can withstand the pressures associated with the blowdown. During all tests the core and core support structures will be instrumented to determine the stresses on fuel pins, control rods, and grid plates and the pressure drop across the core during blowdown.

Phase II of the test program is separated into three parts: (1) coolant blowdown, (2) spray systems evaluation, and (3) fission product transport. The general test procedure and the data to be obtained are discussed below.

a. Coolant Blowdown

The first blowdown test will involve simulating a 4 in. ID break in the outlet coolant pipe with the coolant at 300°F and 1000 psig. This initial test will provide an assessment of the analytical models used to predict the effect of blowdown on the nuclear system and containment vessel and permit estimates to be made of the ability of the reactor and plant to withstand more severe blowdowns.

A series of about six tests will then be performed to investigate the effects of break size and location on the response of all systems. These tests will be conducted by simulating 4 in., 10 in., and 16 in. ID breaks in both the inlet and outlet coolant lines. The device used to simulate a rupture will be as close as possible to the reactor vessel. The primary coolant during these six tests will be at 540°F and 2330 psig (normal operating conditions).

Following these tests, a series of blowdowns through simulated 16 in. ID breaks will be performed to investigate the effects of temperature and pressure on the response of the reactor and plant equipment. The temperatures and pressures to be considered are from 450°F to 600°F and 1700 psig to 2500 psig, respectively.

The quantitative data to be obtained during and following each blowdown test will include the following:

- (1) mechanical stresses on the dummy core, control rods, reactor vessel, primary coolant pipes and components, and containment vessel,
- (2) temperature of reactor components and containment vessel,
- (3) pressure drop across the core,
- (4) pressure history inside the primary coolant system and the containment vessel, and
- (5) coolant blowdown rate.

During several of the blowdown tests an attempt will be made to determine the leakage rate from the containment vessel as a function of pressure. Consideration is being given to methods for achieving stable conditions inside the containment vessel at a number of selected pressures such as the introduction of steam and air into the containment building.

Following each test, the entire nuclear system and containment building will be visually inspected for damage.

b. Spray System Evaluation

Immediately following one of the tests noted above in which a 16 in. ID rupture is simulated, the spray system will be energized and the time to reduce the containment building pressure to atmospheric will be determined. The spray system is designed to reduce the containment pressure from 24 psig to approximately 1 psig in 4 hours in the event of an accident or an excessive fission product release.

c. Fission Product Transport

During several of the preliminary blowdown tests, iodide-129 containing a trace of iodine-130 and a small quantity of radioactive krypton will be released inside the reactor vessel and the transport of these materials will be observed. The isotopes will be contained inside a simulated fuel rod located at the center of the core. They will be released at a time after the rupture that is representative of the theoretical time at which clad melting is predicted to occur. The specific purpose of these experiments is to provide information which will assist in selecting the rupture location and size that provides the maximum fission product transport to the containment vessel. These tests will also provide an opportunity to evaluate the sampling and analytical techniques used to measure the iodine transport and plateout behavior. At least one test employing a fission product release will be performed to provide early information on the effectiveness of the pressure reduction spray system and fission product filtering system for reducing the leakage from the containment vessel.

The measurements to be made during this test include:

- (1) iodine and krypton concentrations as a function of time inside the containment vessel volume,
- (2) iodine and krypton concentrations external to the containment vessel,
- (3) iodine plateout on the containment vessel and machinery surfaces, and
- (4) iodine concentration in the condensed coolant.

3. Phase III - Power Operation

This phase of the program includes all low power and high power tests that are required to provide a thorough understanding of the nuclear, hydraulic, and heat transfer parameters of the reactor. An understanding of these parameters is essential for the interpretation and analysis of the experimental results and for the development of mathematical models for correlating the data with other nuclear safety research programs.

Following the last non-nuclear coolant blowdown test, the dummy core will be removed from the reactor vessel along with the core support structure. The stress and pressure data obtained from the non-nuclear blowdown tests will be used to determine the type of core support structures to be employed in the Phase IV loss of coolant test.

a. Physics and Engineering Measurements

The initial experiments will be aimed at verification of the fuel loading and nuclear characteristics of the reactor. Additional physics and engineering measurements will be made with the nuclear system at various operating conditions to establish temperature, pressure, and density coefficients with and without coolant flow. These tests will consist of:

- (1) determining the control rod worth and excess reactivity available for both hot and cold conditions,
- (2) determining the temperature coefficient as a function of temperature,
- (3) measuring the flux distribution in the core,
- (4) measuring the flow distribution and velocity in the reactor vessel, and
- (5) checking the in-core test and control instrumentation.

These tests are necessary to assure safe operation of the reactor at power and to supply data that will assist in the assessment of the shutdown mechanisms during coolant blowdown and the assessment of the power distribution in the core.

b. Power Operation

The reactor will be operated at 50 Mw(t) for at least 400 hours to produce a fission product inventory and a shutdown decay heat flux capable of melting a significant part of the core cladding. At the end of the power run, all radioiodine isotopes will be at essentially an equilibrium condition except I^{131} , which will be at approximately 70% of equilibrium. The fission product inventories as a function of operating time for some of the important isotopes are presented in Table II-A.

TABLE II-A

Fission Product Inventory (curies)
 Following 100, 400, 800, and 1600 Hours of
 Operation

		Inventory at Shutdown - Curies			
Isotope	Half Life	100 Hrs.	400 Hrs.	800 Hrs.	1600 Hrs.
Krypton					
83	114 sec	2.28×10^5	2.28×10^5	2.28×10^5	2.28×10^5
85 m	4.4 hrs	5.32×10^5	5.32×10^5	5.32×10^5	5.32×10^5
85	10.6 yrs	8.66×10^1	3.64×10^2	7.33×10^2	1.47×10^3
87-95		9.45×10^6	9.45×10^6	9.45×10^6	9.45×10^6
Total Kr		1.02×10^7	1.02×10^7	1.02×10^7	1.02×10^7
Strontium					
89	50.5 days	1.16×10^5	4.30×10^5	7.72×10^5	1.26×10^6
90	28.0 yrs	7.15×10^2	2.20×10^3	5.05×10^3	1.14×10^4
91-97		1.21×10^7	1.21×10^7	1.21×10^7	1.21×10^7
Total Sr		1.22×10^7	1.25×10^7	1.29×10^7	1.34×10^7
Zirconium					
95	65 days	1.18×10^5	4.43×10^5	8.14×10^5	1.38×10^6
Ruthenium					
106	1.0 yrs	1.30×10^3	5.14×10^3	1.01×10^4	1.96×10^4
Tellurium					
132	77.0 hrs.	1.22×10^6	2.01×10^6	2.06×10^6	2.06×10^6

TABLE II-A (Continued)

Isotope	Half Life	Inventory at Shutdown - Curies			
		100 Hrs.	400 Hrs.	800 Hrs.	1600 Hrs.
Iodine					
131	8.0 days	3.54×10^5	9.98×10^5	1.24×10^6	1.32×10^6
132	2.3 hrs	1.20×10^6	2.01×10^6	2.06×10^6	2.06×10^6
133	20.8 hrs	2.92×10^6	3.03×10^6	3.03×10^6	3.03×10^6
134	52.5 min	3.38×10^6	3.38×10^6	2.38×10^6	3.38×10^6
135	6.7 hrs	2.68×10^6	2.68×10^6	2.68×10^6	2.68×10^6
136-138		5.77×10^6	5.80×10^6	5.80×10^6	5.80×10^6
Total I		1.63×10^7	1.78×10^7	1.82×10^7	1.83×10^7
Xenon					
133	5.3 days	9.44×10^5	2.62×10^6	2.98×10^6	3.06×10^6
135 m	15.3 min	8.43×10^5	8.43×10^5	8.43×10^5	8.43×10^5
135	9.2 hrs	2.75×10^6	2.76×10^6	2.76×10^6	2.76×10^6
137-142		8.82×10^6	9.82×10^6	9.82×10^6	9.82×10^6
Total Xe		1.44×10^7	1.60×10^7	1.64×10^7	1.65×10^7
Barium					
140	12.8 days	5.63×10^5	1.66×10^6	2.33×10^6	2.71×10^6
Promethium					
143	13.7 days	2.97×10^5	1.37×10^6	2.09×10^6	2.53×10^6
Cerium					
144	290 days	2.70×10^4	1.06×10^5	2.09×10^5	4.01×10^5
Plutonium					
239	2.40×10^4 yrs		2.31×10^1	4.53×10^1	$\sim 8.6 \times 10^1$
240	6.8×10^3 yrs		8.80×10^1	3.52×10^0	$\sim 8.8 \times 10^0$
Total Fission Product		1.91×10^8	2.04×10^8	2.10×10^8	2.16×10^8
Activity Including Above Isotopes					

c. Power Tests

The magnitude and distribution of the fission product inventory and the resulting decay heat are of importance in evaluating the analytical models for predicting the extent of core melting and for evaluating the fission product release from the core. The magnitude of the average decay heat and fission product inventory can be reasonably predicted from existing analytical techniques, if the power distribution in the core is known. The power distribution will be determined by measuring the radial and axial thermal neutron flux distribution in the core at the beginning of the power run and about every 100 hours throughout the run. A final measurement will be made just prior to the final loss of coolant test. In addition, several items such as the fuel and clad temperature, reactor power, critical rod positions, etc., will be continuously recorded throughout the power run to provide a complete operating history.

4. Phase IV - Loss of Coolant Test with a Radioactive Core

This phase of the program consist of demonstrating the consequences of a loss of coolant accident involving a radioactive core. The major objective of this experiment is to obtain quantitative data pertaining to fission product behavior under actual accident conditions that can be used by the nuclear industry and the AEC for the establishment of siting requirements and safety assessment.

Following the extended power run a primary coolant pipe rupture will be simulated and the core will heat-up from fission product decay heat. In general, the information to be obtained during the experiment are:

- (1) the transient thermal behavior of the core,
- (2) the fraction of the core (fuel and clad) that melts and the physical configuration of the core following the event,
- (3) the identity and fraction of fission products released from the fuel,
- (4) the identity and fraction of fission products that reach the containment vessel,

- (5) the natural or inherent factors that tend to remove fission products from the containment atmosphere and what reduction factors can be obtained, and
- (6) the identity and fraction of fission products that escape from the containment vessel.

This phase of the experiment will be separated into four parts: (1) coolant blowdown, (2) core meltdown and disassembly, (3) fission product behavior inside the containment vessel, and (4) fission product dispersion external to the containment vessel. The measurements to be made, and the general plant conditions for each part are discussed below.

a. Coolant Blowdown

Phase IV of this test program will involve simulating a primary coolant pipe rupture and permitting the primary coolant to be expelled from the system. The test will be identical to that described in Section II-B2 except that only a single test can be performed and the reactor will be operating at power.

The measurements to be made during and following the blowdown will include:

- (1) reactor power,
- (2) primary coolant flow and coolant temperatures and pressures inside the primary system and on each side of the rupture opening,
- (3) coolant expulsion rate at the rupture opening,
- (4) coolant velocity and temperature near the core and at the rupture opening as a function of time,
- (5) temperature gradient across the containment vessel walls as a function of time,
- (6) temperature of the containment atmosphere and the equipment inside the containment vessel as a function of time,
- (7) containment vessel pressure and leakage rate as a function of time,

- (8) strain on fuel rods, control rods, reactor vessel, primary coolant piping and containment vessel,
- (9) temperature of the fuel rods (clad and fuel), core structural materials, control rods, reactor vessel and the remaining coolant as a function of time, and
- (10) relative humidity as a function of time.

The primary coolant blowdown will be initiated while the reactor is operating at 50 MW(t) and the primary coolant is at 2330 psig and 540°F. For this test, the size of the rupture and the location will be determined from theoretical analyses and the results of the preliminary blowdown tests.

Just prior to the blowdown the automatic control rod scrams will be administratively bypassed to avoid a scram during blowdown. During the actual blowdown the rods will be held at the established critical position until the power level has been reduced several decades by the inherent shutdown coefficients. The reactor will then be manually scrammed.

The primary coolant pumps will be turned off at the time cavitation is expected to avoid damage.

b. Core Meltdown

During and following coolant blowdown, many complex and interrelated phenomena occur in the core region which ultimately determine the extent of core meltdown and the quantity of radioactive materials release. It is intended during this part of the experiment and from the post-test examination of the core to obtain data which will assist in evaluating these phenomena as well as evaluating the analytical models used for predicting the extent of core damage.

The data to be obtained will include the following:

- (1) fuel, clad, core support, and reactor vessel temperature as a function of time,
- (2) velocity and temperature of coolant through the core as a function of time, and
- (3) extent of core damage and configuration existing after the test has been terminated (from post-test examination).

c. Fission Product Behavior

Rupturing of the fuel element cladding and core disassembly will permit radioactive fission products to be released from the fuel. Upon release, a fraction of the fission products will be contained in the reactor vessel and others will be transported to the containment building where it is assumed some will be removed from the atmosphere by mechanisms such as fallout, deposition or plateout on material surfaces, and absorption by water particles. The specific purpose of this part of the test is to obtain the data with which to determine what actually happens to the fission products, in particular radioiodine, following a loss of coolant accident. To obtain this information, an extensive sampling system will be employed whereby the following data can be collected as a function of time after coolant blowdown:

- (1) fission product plateout on the material surfaces,
- (2) fission product concentration in the containment atmosphere, and
- (3) fission product concentration in the condensed coolant.

The plateout, gas, and particulate sampling devices will be located at 12 representative locations near the inner surface of the containment vessel. The plateout samples will be constructed of the same material as the walls being monitored and will be located approximately 6 inches from the wall. The gas and particulate samplers will be located approximately 30 inches from the plateout samples or approximately 36 inches from the wall. Each time a plateout sample is taken, a gas and particulate sample will also be taken, thus, data pertaining to the plateout behavior of selected isotopes as a function of concentration in the atmosphere and their release time behavior will also be obtained. The gas and plateout sample holders are designed so they can be removed from the vessel and analyzed immediately after the sample has been collected to provide a history of the fission product plateout and the noble gas and iodine concentrations in the atmosphere throughout the test period.

It is envisioned that the gas, plateout, and particulate samples at each of the 12 locations will be collected every 30 minutes for the first 4 hours, every hour for the next 4 hours, and then every 24 hours until the termination of the test. The condensate samples will be collected from the radioactive waste collection tanks at the same time intervals to provide information on the fission product removal or decontamination characteristics of condensed steam. The particulate samples, although collected as a function of time, will not be removed from the vessel until the test has been terminated and access to the vessel can be obtained.

In addition to the removable plateout samples, samples will also be placed throughout the containment vessel to provide a more thorough determination of the plateout distribution on the walls, floors, and machinery. These samples will be sealed from the atmosphere at the termination of the test and removed whenever access to the building can be obtained. Samples will also be placed inside the reactor vessel during core installation to measure the plateout inside the vessel. They will be removed during the post test examination. All samples will be placed parallel to the surfaces being monitored. They will be opened prior to the reactor power run and closed just prior to removal, or to termination of the test. With this procedure, the samples will be exposed to the same environment as the material surfaces being monitored.

An attempt will also be made during the test to determine the chemical state and particle size of radioiodine in the containment vessel. It is recognized that the existing techniques for collecting samples of this type may not be effective in a steam atmosphere. Therefore, consideration is being given to this problem.

Additional data to be obtained throughout the time that fission product samples are being taken are:

- (1) temperature of the atmosphere both inside and outside the containment vessel,
- (2) temperature gradient across the containment vessel walls,

- (3) pressure and relative humidity in the containment vessel,
- (4) gas leakage rate from vessel,
- (5) temperature of a representative plateout sample at each of the 12 locations, and
- (6) temperature and air-steam currents inside the reactor vessel. }

d. Fission Product Dispersion External to the Containment Vessel

In order to obtain information on fission product dispersion and to provide adequate warning should hazardous conditions develop, the area surrounding the LOFT facility will be extensively monitored. The grid for locating the monitoring instruments will consist of a series of concentric circles with radii extending from 50 to 6400 meters. The general layout of the grid and instrument locations are presented in Figure VI-19. Heavily instrumented portions of the grid are two 60° sectors oriented for the most probable lapse and inversion meteorological conditions. The various types instrumentation to be used on the grid are high-volume air samplers (airborne particulate activity and iodine), particle sizing samplers (airborne activity as a function of particle size), fallout plates (deposition dose), film badges (total beta and gamma doses), ionization chambers (dose rates), telemetering area monitoring stations (airborne activity, dose rate, and integrated dose), fission gas detectors (noble gas activity), and chemical dosimeters (total dose). Some of the larger external arcs will be monitored by mobile surveillance vehicles capable of tracking and recording data from the released effluent. These vehicles will also be used to recover the instrumentation and samples from the grid.

The part of the grid not included in the two 60° sectors will contain instrumentation at 30° intervals to provide information on the radiological doses during changes in the wind direction. These measurements will also be supplemented by measurements taken with the mobile surveillance vehicles.

If it is concluded that monitored penetrations through the containment vessel will be installed, the radiological instrument requirements, number and location, will be reevaluated.

C. Termination of the Experiment

It is anticipated that the test will continue for approximately 5 days after core meltdown. At this time or at the time it is concluded that all the useful data has been obtained, the test will be terminated and the facility cleanup procedure will be initiated. The first step in the procedure will be to reduce the internal pressure to atmospheric pressure and to remove the remaining airborne fission products from the containment atmosphere by use of the filter system. This step will then be followed by remote decontamination to remove fission products from the containment vessel walls and machinery surfaces. After it has been established that the large railroad doors can be opened safely, the shielded locomotive will remove the test package to the hot shop for post-test examination. Access to the building can then be attained and the local hot spots that may have survived the decontamination will be cleaned up manually.

The general procedure for each step is discussed below:

1. Pressure Reduction and Fission Product Removal

The containment building internal pressure will be reduced to atmospheric pressure with the internal pressure reduction sprays. The sprays will be operated for approximately 30 minutes, at which time a plateout sample at 12 sampling stations, an air sample at the 12 sampling stations, and a water sample from the collection tanks will be taken to determine the effectiveness of the spray system for fission product removal. The sprays system will then be re-activated and allowed to operate until the pressure is reduced to atmospheric pressure. Again samples will be taken at each of the 12 stations and the water collection tanks.

Further reduction of the airborne activity will be accomplished by either a "once through" venting or a recirculation of the containment air through the fission product removal filter system. The technique to be used will be dependent upon the magnitude of the airborne activity as well as the meteorological conditions existing at the time filtration is initiated. The "once through" venting technique involves discharging the airborne activity through a series of filtering media directly to the atmosphere through the exhaust stack. This method of fission product reduction will be used only if the airborne activity is sufficiently low to prevent exceeding the allowable on-site and general population exposure criteria and only if the meteorological conditions favor subsequent release.

In the event of excessive airborne activity or unfavorable meteorological conditions, the containment vessel atmosphere will be recirculated through the filter media until such a time as exhaust to the atmosphere is considered feasible. Following the fission product filtration and prior to remote decontamination, plateout samples and air samples will be obtained from the sampling stations.

If it is found that the remote decontamination of the containment walls introduces fission products to the containment atmosphere it may be desirable to again recirculate the air through the filter system prior to opening the railroad doors.

2. Decontamination

Following the discharge of the airborne activity to the atmosphere and prior to gaining access to the building, the internal surfaces of the containment vessel and the external surfaces of the test package and associated test equipment will be remotely decontaminated with chemical solutions. The remote decontamination system to be used in the LOFT facility is presently in the design stage, however, prior to installation of this system the decontamination equipment, chemical solutions, and techniques will be tested, evaluated, and perfected.

Upon completion of the final decontamination, plateout, air and water samples will be obtained from the appropriate sample stations.

Analysis of the various samples taken during the pressure reduction, filtration, and remote decontamination process will indicate the effectiveness of each decontamination step as well as the final airborne and deposited fission product activity remaining at the conclusion of the cleanup operation.

D. Post-Test Examination

The reactor system will be transported to the hot shop for the post-test examination after the building has been remotely decontaminated. The transport of the reactor system is expected to begin approximately ten days after completion of the destructive test and will consist of all components which are located on the railroad dolly, i.e., reactor vessel, primary heat exchanger, water filled shield tank, etc. The reactor system is being designed to allow for its access into the hot shop and to allow for access of the hot shop manipulators into the opened vessel. Because of the weight of the reactor system some modifications will of necessity be made to the turntable outside the hot shop.

At the time of the LOFT experiment, the hot shop is expected to have three wall manipulators and an overhead manipulator in operation. Associated with the manipulators are windows in the hot shop walls and a series of closed TV circuits. The height from the floor to the top of the vessel flange will dictate which manipulators and which viewing method can be used while working the vessel interior. An overhead crane with a designed load limit of 100 tons is another integral part of the hot shop equipment. All operations which must be carried out remotely will be accomplished with the use of the manipulators and/or the overhead crane.

The nature of the LOFT experiment dictates the necessity of a complete post-test examination. In order for the examination to have any meaning, a complete and accurate documentation of all operations will be necessary. This documentation will consist of two main parts: (1) a complete photographic coverage - either in black and white or in color, and (2) a complete written log book.

After arrival in the hot shop, the first major operation will be a complete external investigation of the whole system. During

this investigation, pipes will be checked for breaks, welds will be checked for cracks, and the vessel and all other components will be checked for any signs of mechanical damage. All irregularities will be thoroughly documented. The examination of the core and reactor vessel will in general follow the procedures outlined in the following paragraphs.

1. Core

The two 8 in. inspection nozzles will be remotely opened to allow entry of borescopes and a TV camera for the purpose of examining the under side of the core. Every effort will be made to determine the geometry and condition of the core prior to starting remote disassembly operations. Depending on the extent of core damage a secondary critical experiment may be warranted to obtain experimental information and develop techniques for deactivating a power reactor in the event of a loss of coolant accident should occur.

Following the initial investigations, the overhead manipulator will be used to remove the control drives and the vessel head. Before removal of the internal vessel components begins, a complete inspection of the vessel interior will be made and the results documented. The top and bottom core inspections should indicate the extent of mechanical distortions which have taken place during the loss of coolant accident.

Removal of the core components will be done in a systematic manner. As each component is removed, a complete identifying record will be kept which will include the location and condition of all components. In addition, the particle size and state (molten or unmolten) of all fuel and cladding particles will be recorded.

Examination of the fuel and cladding is expected to provide an indication of the percentage of fuel and cladding which have melted. Knowing the percentages of meltdown and the final and initial configurations, an attempt will be made to arrive at a concept of the meltdown history. The meltdown information will be related to information which is to be derived from mechanical and metallurgical studies of the core components. These studies should provide information concerning thermal and mechanical stresses, temperature gradients, mechanical distortions,

etc. By relating the information obtained from the meltdown and mechanical studies, a better understanding of core disassembly process can be obtained.

2. Reactor Vessel

The determination of fission product plateout characteristics will be obtained by chemical analysis of plateout samples positioned throughout the reactor and primary system. Each plateout sample and where physically possible, each reactor component will be individually decontaminated in chemical solutions to strip the fission products from the metal. From a knowledge of the activity of the solution, the volume of the solution, and the surface area of the decontaminated component, an estimate of the deposited fission products can be obtained. If possible, the reactor system (including heat exchangers, connecting piping, etc.) will be resealed and purged with a decontaminating solution. Once again, from a knowledge of solution activities, solution volumes, and exposed surface area, and estimate of fission product plateout in the reactor system can be obtained.

In order to arrive at some estimate of the relative fission product plateout percentages, some knowledge of the available isotopes is required. Because of the delay time before and during the post-test examination, the short half-life fission products will have decayed out. The following isotopes have sufficiently long half-lives and are produced in sufficient amount to warrant investigation in this experiment: Kr-85, Rb-86, Sr-89, Sr-90, Zr-95, Ru-106, I-131, Te-132, Xe-133, Xe-135, Ba-140, Ce-144.

The objectives of the post-test examination are fairly well defined, however, the methods of fulfilling these objectives are still in the developmental stage. Therefore, in outlining the post-test examination, only basic concepts or methods are presented. Further research may dictate the necessity of altering or amending the proposed experimental methods. In the expansion and development of the post-test experimental program, fulfillment of the experimental objectives will be the controlling factor.

SECTION III

DESIGN CONSIDERATIONS

III. DESIGN CONSIDERATIONS

The test program to investigate the loss of coolant accident is intended to provide the AEC and the nuclear industry with engineering information which will assist in siting power reactors, and in achieving safety in the design of power reactors for the least cost. To achieve this objective, the LOFT facility and experimental concepts developed reflect to the extent possible current reactor design practice in the nuclear industry. However, the assignment of existing facilities at the Test Area North (TAN) of the National Reactor Testing Station (NRTS) to the nuclear safety program and the siting of LOFT near the FET facility complex has influenced the LOFT design. The influence of the site and the general design philosophy is discussed in the following paragraphs.

In view of the existence of extensive hot shop facilities, railroad trackage, a shielded locomotive and other facilities, the design concept selected for LOFT is one in which the reactor and primary coolant system is located on a mobile platform or railroad dolly. This concept offers the following economics and advantages: (1) the reactor assembly or experiment can be fabricated independent of the facility, thus making maximum use of the facility, (2) the facility can be more readily decontaminated and refurbished for reuse, (3) the reactor can be easily moved to the hot shop for post-test examination and disposal, (4) remote handling equipment and biological shielding in the facility can be minimized with considerable savings, since "hot" operations can be performed in the hot shop, and (5) the equipment permanently installed in the containment building can be greatly reduced, thereby minimizing the cost and adding flexibility to the facility.

Use of the dolly mounted reactor concept necessitates certain compromises which are considered acceptable. For example, a permanently installed reactor system with the attendant biological shield would certainly offer a more restricted path for the movement of fission products provided the pipe rupture occurred within the shield. However, because of the difficulty in simulating a typical biological shield the approach has been to design the LOFT experiment such that the

restrictions to fission product transport are minimized and the maximum quantity of fission products will reach the containment building. Elimination of the biological shield makes analysis of experimental results easier and counters the suggestion that a more hazardous condition could have prevailed.

In order to carry out the loss of coolant test, a rupture or break in the coolant circuit must be simulated. The effect of an artificial fully-instrumented orifice on the consequences of a loss of coolant is not well known or easily analyzed. However, numerical analyses have indicated that it is possible with the proposed blowdown nozzle arrangement to discharge the coolant at a rate sufficient to void the core region in approximately 10 to 15 seconds after initiating the rupture. Since this blowdown rate compares favorably with that postulated for most water-cooled reactors (see Table III-A), the rupture configuration is not considered to be of particular significance. It is recognized however, that the location of the rupture and the rupture size may have a significant effect on the transport of fission products. These effects will be investigated during the early phases of the experimental program.

The nuclear core designed to meet the objectives of the initial experimental program consists of slightly enriched uranium dioxide (UO_2) in the form of sintered cylindrical pellets stacked in stainless steel tubes. Although a number of power reactors have recently been proposed with zirconium clad fuel, the stainless steel clad fuel design was based on the predominance of reactors using this type fuel and prognostications in 1963 by leading fuel fabricators and reactor designers that stainless steel fuel would dominate the industry for several years.

The size of the LOFT core and the reactor power has been optimized to reduce costs while achieving experimental objectives. The power has been kept low to minimize heat dissipation equipment costs while at the same time maintaining a core large enough to insure that melting of a significant amount of the fuel will be attained. The power density was chosen consistent with current commercial power designs and is approximately 62 kw.

TABLE III-A

SUMMARY OF REACTOR CHARACTERISTICS

REACTOR PARAMETERS	LOFT	YANKEE	SAVANNAH	RAVENSWOOD	SAXTON	INDIAN POINT	SHIPPING PORT
Thermal Power - Mw	50	392	69	2030	20	585	231
Primary System Volume	650	1470 ⁽³⁾	680 ⁽³⁾	11,930	290	2350 ⁽³⁾	1235 ⁽³⁾
Large Break Size-ft ²	1.43	1.43 ⁽⁴⁾	0.87 ⁽⁴⁾	-----	0.0375	2.05	1.23
Blowdown Time-sec.	10	16	30	-----	80	60	26
Time for Cladding to Reach Melting Temp.-sec (6)	200 ⁽¹⁾	155 ⁽²⁾	500 ⁽²⁾	-----	180 ⁽²⁾	180 ⁽²⁾	Seed 205 ⁽²⁾
System Pressure-psig	2330	2000	1750	2035	2000	1500	2000
Core Power Density-kw/lcore	62	58	21	75	54	76	23
Fuel Power Density-kw/MW ₀₂	256	167	86	198	200	288	79
Average Coolant Temp. °F	540	516	508	569	530	503	523
Core ΔT - °F	31	33	23	50	---	---	---
Reactor Vessel ΔT	---	---	---	-----	22	32	30
Average Fuel Temp.-°F	1500	940	960	1720	1230	881	749
Average Cladding Surface Temp. °F	614	530	534	592	565	570	580
Average Heat Flux Btu/hr-ft ² x10 ⁻⁵	1.67	0.86	0.62	1.46	1.37	1.28	Seed 0.98
Max. Hot Channel Heat Flux Btu/hr-ft ² x10 ⁻⁵ (5)	4.0	2.49	1.87	4.74	4.44	5.37	Seed 4.2
Hot Channel Factor (FAFR)(5)	3.05	2.9	3.0	3.25	3.24	4.2	Seed 4.3
Break Location with Respect to Core	Above/ Below	Above/ Below	Below	-----	Above	Below	Below

- (1) 100 EFPH
 (2) ∞ Operation
 (3) One Half of Total Volume

- (4) One Half of Total Flow Area-Offset Shear Break
 (5) Does Not Include Hot Spot Factors
 (6) At Design Hot Spot

The loss of coolant tests will be conducted with a low-burnup core, since experiment data are available for predicting the fission product release and behavior from UO_2 fuel as a function of fuel burnup. Thus, the overall results of tests with a low-burnup core can be extrapolated with reasonable accuracy to estimate the results obtained with a high-burnup core.

Although the LOFT facility is primarily designed to accommodate a water-cooled and -moderated nuclear system mounted on a mobile dolly, consideration has been given in the design to nuclear safety investigations using other reactor types such as sodium graphite reactors, fast reactors, and gas-cooled reactors. With the addition of supporting utilities and other systems peculiar to a specific class of reactors, sufficient flexibility has been included that it appears feasible to use the LOFT facility for ther safety tests.

A. Design Parameters

The following design parameters for the LOFT facility and experimental systems are those which existed on February 1, 1964.

1. Reactor

Reactor type	Pressurized water
Operating power, thermal	50 Mw
Operating power, thermal	1.7×10^8 Btu/hr
Continuous operation	3,333 Mwd
Core diameter	42 in.
Active core height	36 in.
Number of fuel assemblies	52
Number of pins/assembly	64
Fuel assembly size	4.806 in. x 4.806 in.
Fuel element	
Pellet diameter	0.357 in.
Rod diameter (OD)	0.39 in.
Gap	0.002 in.
Clad thickness	0.015 in.
Rod spacing (center to center)	0.580 in.
Clad material	Stainless steel
Average fuel temperature	1,500°F
Average clad surface temperature	614°F
Fuel enrichment (two zone core)	
Inner zone	2.82%
Outer zone	5.70%
Control rods	12
Type	Cruciform
Material	Boron-Stainless (1.5 w/o B^{10}) or nickel plated
	Ag-In-Cd or hafnium
Follower	Zircaloy

Width	8.898 in.
Thickness	0.404 in.
Poison length	36 in.
Coolant flow	
Total flow	5.6×10^6 lb/hr
Flow past fuel pins	4.5×10^6 lb/hr
Leakage flow	1.1×10^6 lb/hr
Velocity along fuel rods	5.3 ft/sec
Pressure	
Coolant pressure, maximum operating	
Non-nuclear operation	2,500 psig
Nuclear operation	2,330 psig
Reactor vessel pressure drop	20 psi
Core pressure drop	6.5 psi
Heat transfer	
Active surface area	1,022 sq ft
Heat flux, average	1.67×10^7 Btu/(hr)(sq ft)
Heat flux, maximum	4.00×10^7 Btu/(hr)(sq ft)
Power densities	
Core	62 kw/liter
Fuel	256 kw/liter
Average core thermal neutron flux	3.16×10^{13} n/(cm ²)(sec)
Peak-to-average power distribution	2.4
Axial	1.5
Radial	1.6
Core lifetime	1,600 effective full power hours
Reflector	Water
Core composition by volume	
UO ₂	26.7%
Water	66%
Stainless steel	7.3%
Reflector thickness	3.5 in.
Core barrel thickness, stainless steel	1 in.
Thermal shield, carbon steel	2 in.
Water gap between core barrel and thermal shield	1.0 in.
Temperatures	
Average coolant in vessel	540°F
Average coolant rise in vessel	25°F
Average coolant rise in core	31°F
Average film drop	70°F
Average surface	614°F
Inlet to reactor vessel	528°F
Outlet from reactor vessel	553°F
Eigenvalues and average temperature coefficient of reactivity for UO ₂ fuel 2.82/5.70% enriched	
K _{eff} (cold)	1.128
$\rho_c = K_{ex}/K_{eff}$ (cold)	11.3%
K _{eff} (hot, full power)	1.057
$\rho_H = K_{ex}/K_{eff}$ (hot, full power)	5.4%
$\rho_c - \rho_H$	5.9%

Average temperature coefficient of of reactivity/ $^{\circ}\text{F}$	-1.1×10^{-4}
Reactor control	
$K_{\text{eff}} =$ (12 rods in)	0.896
$K_{\text{eff}} =$ (cold, 11 rods in)	0.959
Control rod worth ($\Delta K/K_1 K_2$)	23%
Shutdown reactivity, cold	-11.6%
Reactor pressure vessel	
Internal diameter	57-3/4 in.
Cylindrical shell thickness	3.75 in.
Hemispherical head thickness	3.75 in.
Material	Carbon steel (A-212B or A-302B)
Overall height	24 ft
Total weight	---
Design pressure	2,750 psig
Temperature	600 $^{\circ}\text{F}$
Hydrotest	Per ASME code
Code	ASME code
Vessel insulation	Required
Height of system, limit (including control rod drive extension and railroad dolly)	33 ft
Gamma heating in reactor pressure vessel	0.1 w/gram

2. Reactor Shield Tank

Shield water temperature, maximum	150 $^{\circ}\text{F}$
Mounting	On dolly
Thermal neutron flux from shield during normal operation at full power	Less than $10^5 \text{ n}/(\text{cm}^2)(\text{sec})$
Dose rate from fission products and activation gammas reduced to less than 200 mrem/hr from vessel (after operation for 3,333 Mwd prior to shutdown and after 15 min following normal shutdown)	

3. Reactor Primary System

Material	Carbon steel
Design pressure	2,750 psig
Operating conditions	
Reactor power, thermal	50 Mw
Reactor inlet temperature	528 $^{\circ}\text{F}$
Core outlet temperature	559 $^{\circ}\text{F}$
Vessel outlet temperature	552 $^{\circ}\text{F}$
Average cladding surface temperature	614 $^{\circ}\text{F}$
Reactor inlet pressure, maximum (nuclear operation)	2,330 psig
Core pressure drop	6.5 psi
Vessel pressure drop	20 psi
Average coolant velocity in core	5.3 ft/sec

Core flow	4.5×10^6 lb/hr
Total reactor flow	5.6×10^6 lb/hr
Pressurizer	
Type	Steam dome
Immersion heaters	
(wired for staged operation)	200 kw
Internal surface cladding	Stainless steel
Operating pressure maximum	
Non-nuclear operation	2,500 psig
Nuclear operation	2,330 psig
Operating temperature	668°F
Design pressure	2,750 psig
Design temperature	683°F
Steam generator	
Arrangement	Vertical
Heat transfer surface area	2,000 sq. ft.
Heat load	171×10^6 Btu/hr
Reactor coolant inlet temperature	552°F
Reactor coolant outlet temperature	528°F
Tube layout	Triangular
Tube type	U-tube
Tube size	5/8 or 1/2 in.
Tube passes	2
Pressure drop tubeside	10 psi
Design pressure	2,750 psig
Design temperature	600°F
Shell design pressure	
Shell operating pressure, full load	340 psia
Shell operating temperature, max.	600°F
Shell operating temperature, normal	429°F
Shell diameter	5 ft 6 in.
Overall height	14 ft
Primary pump	
Number	As required
Type	Canned rotor
Flow, total	15,000 gpm
Pumping temperature	528°F
Suction pressure, maximum operating	
(nuclear)	2280 psig
Discharge pressure, maximum operating	
(nuclear)	2,330 psig
Design pressure, casing	2,750 psig
Operating temperature, maximum	600°F
Drive power, total	400 hp
Quench tank	
Flow to tank	High pressure primary steam and vapor
Makeup quench water	Demineralized water
Drains	To warm waste
Vent	To stack
Location	Off dolly
Primary purification system	
Primary makeup pump	
Flow	5 gpm

Suction pressure	1 atm
Design pressure	2,750 psi
Pumping temperature	130°F
Surge makeup tank	
Design pressure	15 psig
Material	Stainless steel
Makeup	Demineralized water
Additive	Hydrazine and/or hydrogen
Ion exchange columns	
Total flow	70 gpm
Flow each	35 gpm
Number of columns	2
Columns operating (one column can operate while the resin is being replaced in the other column)	2
Resin temperature	130°F max.
Type	Mixed bed
Inside diameter	14 in.
Bed depth	30 in.
Operating pressure, maximum	
Non-nuclear	2,500 psig
Nuclear	2,330 psig
Primary purification heat exchanger	
Primary coolant flow	70 gpm
Primary coolant inlet	528°F
Primary coolant outlet	160°F
Primary coolant return	130°F
Primary coolant design pressure	2,750 psig
Primary coolant operating temperature, maximum	600°F
Primary purification cooler	
Primary coolant flow	70 gpm
Primary coolant inlet	160°F
Primary coolant outlet	130°F
Primary coolant design pressure	2,750 psig
Service water	85°F
Shutdown pump	
Flow	3.0×10^5 lb/hr
Pump type	Canned rotor
Motor power	Emergency

4. Secondary Coolant System

Design pressure	1,100 psig
Design temperature	600°F
Steam generator, shellside	
Operating pressure, full load	340 psia
Operating temperature	429°F
Operating pressure, maximum	950 psig
Material	Carbon steel
Steam generation	214,000 lb/hr
Heat load	171×10^6 Btu/hr
Air-cooled condenser	
Type	Multi-cell, fan cooled

Operating pressure	330 psia
Pressure drop	5 psi
Operating temperature	426°F
Tube material	Carbon steel
Fin material	Aluminum or copper
Control	Adjustable shutters and fans
Provisions will be made to prevent freezing of the fin-fan air condenser at low power or during shutdown.	
Condensate receiver	
Capacity	500 gal
Material	Carbon steel
Condensate pumps	
Number	1
Flow each	550 gpm
TDH	150 ft
Normal suction pressure	325 psia
Maximum suction pressure	1,000 psig
Maximum suction temperature	600°F
Material	Carbon steel
Secondary auxiliary pump	
Number	1
Flow	15 gpm
TDH	90 ft
Maximum suction pressure	1,000 psig
Maximum suction temperature	600°F
Secondary demineralized water makeup pump	
Flow	5 gpm
TDH	3,600 ft

5. Seismology

Earthquake zone	2
-----------------	---

6. Test Building

Containment shell	
Free volume	$2.6 \times 10^5 \text{ ft}^3$
Pressure after blowdown of primary system	24 psig
Maximum pressure after blowdown of primary and secondary system	30.5 psig
Atmospheric pressure	12.5 psia
Reactor primary system pressure	2,500 psia
Reactor primary system temperature	600°F
Volume of primary water	650 cu ft
Secondary system pressure	340 psia
Secondary condensate	270 cu ft
Secondary system temperature	429°F
Secondary quantity of steam	250 lb
Code	ASME
Test pressure	Per ASME code
Thickness of concrete lining from operating floor to overhead crane	1 ft

Dome lining thickness	6 in.
Uncontrolled leakage to atmosphere, maximum	0.2% of free volume/ day at 24 psig
Venting	
Free volume per day	0 to 5%
Pressure conditions	5 psig to maximum
Largest opening (railroad door)	
Width	22 ft
Height	33 ft
Personnel air lock	2 required, 7 ft dia.

The operating floor of the test area within the containment shell will be designed for the live load resulting from the containment shell internal pressure, plus an additional live load of $2,000 \text{ lb/ft}^2$ (in the floor area occupied by the dolly trackage the live load imposed by the dolly will be used if this exceeds $2,000 \text{ lb/ft}^2$).

A pressure reduction spray system shall be provided inside the containment shell, capable of reducing the pressure from 24 psig to less than one psig within four hours following a loss of coolant test. In addition, the spray system shall be capable, if the spray is initiated immediately after the accident, of reducing the containment shell differential pressure and hence shell leakage within a period sufficient to provide less than 300 rem dose to the thyroid at the NRTS boundary under the following meteorological conditions:

Average wind velocity	$\bar{U} = 2.0 \text{ m/sec}$
Stability parameter	$n = 0.5$
Virtual diffusion coefficient in the horizontal plane	$C_y = 0.280 (\text{meters})^{n/2}$
Virtual diffusion coefficient in the vertical plane	$C_z = 0.035 (\text{meters})^{n/2}$
Down wind distance to nearest inhabited site boundary	$D = 7 \text{ miles}$
Leak rate at 24 psig internal pressure	0.2%/day

These parameters enter the Sutton equation for calculation of the down-wind concentration of radioactive material at a distance D. A core melt-down which releases 15% of the fission products to the containment shell will be assumed.

A deluge spray system will be provided and arranged to assure a complete wash down of all expressed containment building and equipment surfaces. The system will contain high pressure jet sprays with rotating head nozzles. The following criteria apply to this system:

Capacity, each	60 gpm
Pressure	115 psig
Nozzle distance from permanent surfaces	20 ft
Localized decontamination	
Steam outlets	4
Steam pressure	125 psig
Portable solution tank	1
Decontamination solution mix tank (with motorized propeller mixer)	
Size	15,000 gal
pH	1.5 to 13
Mix tank steam heater	
Decontamination pump	
Flow	200 gpm
Power	20 hp

B. Design Responsibilities

As a part of the nuclear safety research program sponsored by the AEC, Phillips Petroleum Company was assigned the responsibility to conduct a series of engineering scale experiments aimed at investigating the consequences of nuclear accidents. The often postulated loss of coolant accident for pressurized water systems was selected as the initial experiment and investigations were begun to define a program which would provide the nuclear industry with experimental information and data for use by safety evaluators and reactor designers. Subsequently, report IDO-16833, Revision 1, entitled "Feasibility and Conceptual Design of the STEP Loss of Coolant Facility" was issued by Phillips Petroleum Company early in 1963. The report presented an experimental program aimed at demonstrating a loss of coolant accident on a pressurized water reactor and a conceptual design for a dry containment test facility to accomplish the loss of coolant experimental safety program.

Preliminary invitations for architect-engineer services were then solicited by the Engineering and Construction Division of AEC-ID using report IDO-16833 as the design criteria. In June of 1963, Kaiser Engineers of Oakland, California was awarded the A-E contract on the LOFT facility and the complete experimental system excluding the design of the reactor core and vessel internals which was assigned to Phillips Petroleum Company. The Babcock and Wilcox Company of Lynchburg, West Virginia was awarded a nuclear sub-contract to Kaiser Engineers for review of the experimental concepts for compliance with current pressurized water reactor technology.

As shown in the LOFT lapse-time schedule, Figure III-I, the experimental facility Title I design is approximately 70% complete (as of February 1, 1964) with Title II design presently scheduled to commence approximately May 1, 1964. Assuming project authorization is granted to permit construction of the facility and fabrication of the experimental assembly to begin in the first quarter of fiscal year 1965, it is estimated that the experimental program can begin in the second quarter of fiscal year 1967 to culminate in the final loss of coolant test approximately one year later.

LOFT SCHEDULE
(DURATION IN MONTHS)

Fig. III-1 - LOFT Schedule

SECTION IV

EXPERIMENTAL ASSEMBLY DESCRIPTION

IV. EXPERIMENTAL ASSEMBLY DESCRIPTION

A. General

The nuclear system for LOFT is a 50 Mw (t) pressurized water reactor containing a low enrichment-oxide core. The core design reflects, as far as possible, contemporary reactor physics and engineering technology. No attempt has been made to simulate or duplicate the features of a particular operating or planned system. Instead, the system incorporates features common to most commercial pressurized water reactors.

The nuclear system is mounted on a dolly and designed for remote assembly, disassembly and maintenance of the experimental equipment. The dolly mounted system will be moved from the test building to TAN hot shop following the final loss of coolant test for a post test examination. The dolly, which is approximately 48 ft. long by 20 ft. wide, moves over two parallel sets of railroad track using commercially available running gear. The primary coolant system and associated equipment on the dolly are mounted on four load cells designed to provide weight loss information as a function of time during the system blowdown. The dolly and experimental equipment are shown in Figure IV-1.

The dolly floor and the supporting framework have smooth surfaces to facilitate decontamination. Instrumentation and control leads are armored and located for maximum protection from flying missiles and are shielded from stray electrical fields. The leads connect through containment vessel penetrations to instrumentation and controls located in the control and equipment building.

B. Reactor Description

1. Pressure Vessel

The 57 3/4 inch ID x 24 ft. high reactor pressure vessel shown in Figures IV-2, IV-3, and IV-4 consists of a cylindrical shell with a flanged and bolted top head. The vessel contains an inlet flow baffle and supports for the core and the thermal shields. The inlet flow baffle distributes the coolant flow uniformly across the core and prevents direct coolant impingement on control rod shrouds. The baffle

will withstand the full force of the coolant blowdown. The top head studs are designed for remote tightening and loosening. The vessel is designed in accordance with ASME Code but does not contain an ASME Code Stamp.

A catch pan below the bottom head of the pressure vessel will confine any fuel that might melt through the vessel and prevents fission products from escaping through the bottom of the containment vessel. The catch pan is designed to retain 75% of the core (6.25 cu ft) at 3,000 F. The catch pan is also designed so that the convective heat loss from the bottom head, prior to possible melt-through, is maintained at approximately the level expected if there were no catch pan.

The materials used in the pressure vessel and its important design features are as follows:

Materials

Plate	- SA212B or SA302B - firebox quality
Forgings	- SA105 grade 2
Pipe and tubing	- SA106 grade B
Bolting	- SA193 grade B14 (studs) SA194 grade 4 (nuts and washers)
Flow baffle	- SA212B - firebox quality

Dimensions

Vessel ID	- 57 3/4 in.
Vessel OD	- 67 in.
Height	- ~ 24 ft. (1)
Cylindrical shell thickness	- 3 3/4 in.

<u>Weight</u>	100,000 lb (less internals)
---------------	-----------------------------

Nozzles

Two, 20 in. O.D. coolant nozzles

(1) The overall height of the dolly-mounted experiment is limited to the vertical clearance of the TAN hot shop door, 33 feet.

Four, 6 in. ID for sample (top head)
Two, 8 in. for inspection (bottom head)
Five, 8 in. for instrumentation (shell above core)
Six, 4 in. for instrumentation (shell below core)
One, 3 in. for instrumentation (shell above core)
Twelve, actuator nozzles (top head) and nozzles as
required for operational instrumentation
Four, 2 in. for instrumentation (shell above core)

Components Designed for Remote Handling (2)

Vessel head, including bolting and seals.

Internal core support barrels.

Flanges on two, 8 in. inspection nozzles in the bottom head and four, 4 in. sample nozzles in the top head.

Control rod actuators.

Thermal shields.

The vessel is supported by a skirt attached to the upper vessel wall resting on the experiment structure. To control possible vibration during blowdown, the lower end of the vessel has radial supports to restrain lateral movement without impeding radial growth due to thermal expansion.

The exterior surface of the vessel is insulated to maintain wall temperature during the loss of coolant test similar to a commercial reactor. The insulation also minimizes thermal stresses in the vessel wall during tests. The insulating material is radiation resistant, fire-proof, will not break down to produce toxic or corrosive products, and is lagged with sheet metal. Overlaps in the lagging reduce moisture penetration and contamination.

A platform is provided around the reactor pressure vessel level top flange and includes safety chains and access ladders to permit handling of in-tank components.

(2) All remote handling methods are compatible with the equipment and procedures used at the TAN hot shop.

2. Vessel Internals

The general layout of the core support structure, the thermal shield and the core are shown in Figure IV-2. The core support structure consists of an upper and lower plate, an upper and lower support barrel, and twelve control rod shrouds. This structure supports the core weight, aligns the core assemblies and control rods, and provides for the proper channeling of the upward coolant flow. The entire core support structure rests as an integral unit on the support ledge of the reactor pressure vessel and is restrained by a compressive load applied by the top head. A compression flange and guide pin positioning holes restrain and orient the support structure.

An instrument lead and shock absorber support plate is located above the upper core support plate but within the upper support barrel. This plate is also attached to a support barrel which rests on the support ledge. The function of this structure is to support the shock absorber weight and deceleration forces of the control rods. It also positions the fuel element sub-assembly instrument lead tubes.

A 2 inch thick thermal shield surrounds the reactor core in the annulus between the lower core support barrel and the vessel wall to reduce the gamma heating rate of the reactor vessel wall. The shield is supported and oriented by brackets which are an integral part of the vessel. The support brackets are located so that they do not obstruct the horizontal access of the lower instrument nozzles.

3. Reactor Core

The reactor is fueled with slightly enriched uranium dioxide in the form of sintered cylindrical pellets stacked in stainless steel tubes. The individual fuel pins are grouped into 52 fuel elements of fuel pins each as shown in Figure IV-5. The pins are maintained in proper position by spring clip spacer grids. The grids are welded to perforated stainless steel side plates at three elevations. The core has an active length of 36 inches. The fuel element is equipped with top and bottom restraining grids. These grids distribute the coolant flow across the assembly and restrain the fuel pins in their vertical position. All 52 fuel elements contain a removable sub-assembly of

four fuel pins which can be removed with the top core support plate in position.

The fuel elements are positioned in the core by top and bottom end boxes which mate with the upper and lower core support plates. The core is surrounded by a flow skirt which confines the upward coolant flow to the active core. The flow skirt is safety-lock-bolted and doweled to the flange of the lower core support barrel.

The total primary coolant flow is 5.6×10^6 lb/hr of which 4.5×10^6 lb/hr flows past the fuel pins and 1.1×10^6 lb/hr is the leakage flow. The required leakage flow tabulated in Table IV-A is determined by heat removal requirements of the flow skirt, the lower core support barrel, the thermal shield, and the vessel wall. Part of the leakage flow is required to cool the control rods.

4. Control Rods and Drives

Twelve top-entry control rod drive mechanisms move the cruciform control rods. The drive mechanisms, capable of sealing against 2,750 psig, 600 F water are mounted on access nozzles located on the vessel top head on 10 inch centers. The drive mechanisms can be operated individual or as a group and can simultaneously scram all rods.

The control rods are located between the fuel elements as shown in Figure IV-4. Each rod is composed of an active poison section of boron stainless steel and a zirconium follower section to minimize flux peaking when the absorber section is withdrawn.

5. Shield Tank

A reactor shield tank surrounds the reactor pressure vessel to decrease induced activity in the components inside the containment building from the gamma rays emanating from the reactor pressure vessel. The shield tank is capable of:

- (1) Reducing the thermal neutron flux emanating from the shield to 10^5 neutrons per cm^2 -sec during full power operation.

- (2) Reducing the dose rate from fission products and activation gammas to less than 200 mrem/hr at the surface of the shield at 15 min after a normal shutdown.

TABLE IV-A

Leakage Flow Requirements for Heat Removal

Component	$\times 10^{-4}$ Heat Flux Btu/hr-ft ²	Flow gpm	Inside Temp. °F	Outside Temp. °F	Max. Temp. °F
Flow Skirt (inside)	0.856	597	543	552	553
(outside)	0.298				
Support Barrel					
(inside)	0.444	666	558	558	561
(outside)	1.13				
Thermal Shield					
(inside)	1.13	237	558	558	598
(outside)	0.456				
Pressure Vessel					
(inside)	0.456		558	600	600

6. Summary of Design Features

A summary of the important design features of the core is presented in Table IV-B. Even though some changes may be made in the reactor design as a result of additional information which will be obtained later, present calculations indicate that the basic design presented here can meet the requirements of the LOFT program.

C. Heat Removal Systems

1. Primary Coolant System

a. System Description

The LOFT primary coolant system which is shown in Figures IV-6 and IV-7 is a single closed loop of pressurized water under forced circulation. The two functions of the system are: (1) to extract 50 Mw

TABLE IV-B

LOFT Core Design Features

Active length (in.)	36
Number of fuel assemblies	52
Number of pins/assembly	64
Fuel element	
Fuel pellet diameter (in.)	0.357
Gap between clad and fuel (in.)	0.002
Clad thickness (in.)	0.015
Fuel pin diameter (in.)	0.391
Rod spacing (in.)	0.580 (center to center)
Clad material	Stainless steel
Average fuel temperature, (°F)	1,500
Average clad temperature, (°F)	625
Fuel enrichment	
Outer zone, %	5.70
Inner zone, %	2.82
Control rods	
Number	12
Type	Cruciform
Material - poison section	Boron-stainless (1.5% B ¹⁰)
Follower section	Zirconium
Normal Operating power Mw(t)	50
Average heat flux, Btu/(hr)(sq. ft)	1.67×10^5
Maximum heat flux, Btu/(hr)(sq. ft.)	4.00×10^5
Physics	
K excess (cold), (%)	12.8
K excess (hot), (%)	5.7
Shutdown reactivity (cold) (%)	11.6
End of life peak-to-average power distribution	
Axial	1.5 (approximately)
Radial	1.6 (approximately)
Core lifetime	1,600 efph
Core area, ft ²	9.96
Reflector	H ₂ O
Average coolant temperature, (°F)	544
Average coolant rise in core, (°F)	31
Primary coolant system pressure (psig)	2,330

of heat from the reactor core and to transfer this heat to a secondary coolant in the primary heat exchanger, and (2) to develop data useful for blowdown evaluation in an experimental program. The entire system, with the exception of the chemical addition and water make-up auxiliary system, and the resin hopper and eductor, is mounted on the dolly.

The primary coolant loop consists of a reactor pressure vessel containing the nuclear core, the tube side of the steam generator, primary coolant pump, and connecting piping. An emergency decay heat removal pump is arranged in parallel with the primary coolant pump and is used for decay heat removal in case the primary coolant is inoperative.

Under normal operating conditions, high purity water enters the reactor pressure vessel at 528 °F. Most of the primary water passes upward through the core, while a portion cools the flow skirt, the thermal shield, and the pressure vessel wall. In passing through the pressure vessel, the bulk water temperature is raised to 552 °F. The primary water then leaves the pressure vessel, passes through the tube side of the steam generator and enters the primary coolant pump. The primary coolant pump circulates water through the loop at approximately 15,000 gpm (5.6×10^6 lb/hr). From the pump discharge, water flows back to the reactor for reheating to complete the closed cycle.

Certain auxiliary systems are provided within the primary coolant system. These include (1) pressurizer system, (2) emergency decay heat removal pump, (3) coolant purification system, (4) chemical addition and water make-up system, and (5) experimental blowdown nozzles.

The pressurizer system maintains the primary loop at operating pressure and reduces pressure transients due to changes in water volume during load transients. Too low a pressure would permit boiling in the core and too high a pressure would exceed design limits. The water in the pressurizer is maintained at a higher temperature than the bulk circulating coolant. Electrical immersion heaters keep the water in the pressurizer at the saturation temperature corresponding to operating pressure; e.g., 669 °F for 2,500 psig assuming negligible non-condensables. Rapid load changes momentarily tend to change the bulk circulating water temperature, resulting in a change in water volume. However, the steam volume in the pressurizer expands or contracts to maintain a relatively constant pressure in the primary loop.

A portion of the primary coolant (approximately 70 gpm) from the primary coolant pump discharge passes through the coolant purification system. This coolant stream passes in series through the tube side of regenerative and non-regenerative heat exchangers, ion exchange demineralizers (to remove both corrosion and fission products), the shell side of the regenerative heat exchanger, then back to the primary pump suction.

High liquid level in the pressurizer opens a valve permitting some of the coolant to flow into a quench tank. A high pressure makeup pump injects make-up water into the primary system in which additives required to maintain low oxygen levels or adjust pH in the coolant are present. The make-up water is pumped through the shell side of the regenerative heat exchanger in the coolant purification system in order to warm the cold make-up before adding it to the hot primary coolant loop.

Make-up to the primary coolant system is demineralized water with specific resistance of 500,000 ohm-cm. Hydrazine is added through the make-up pump for oxygen removal when the reactor is not operating. When the reactor is in operation, hydrogen is added to the system for oxygen removal. The pH in the primary coolant system is controlled from approximately 9 to 10.5

The nitrogen-16 activity in the primary coolant is expected to be 2×10^7 Mev/(cm³) (sec) at the reactor outlet and 4.5×10^6 Mev/(cm³) (sec) at the reactor inlet during nuclear operation.

The volume of primary coolant in the system and in the individual systems' components at reactor operating conditions is listed in Table IV-C.

TABLE IV-C

Primary System Coolant Volume

<u>Component</u>	<u>Primary Coolant Volume, ft³</u>
Reactor pressure vessel	320
Steam generator	20
Pressurizer	70
Piping	290
Miscellaneous	<u>50</u>
Total	750

The design service life for the primary coolant system is two years. All material in the primary coolant system is carbon steel, except for the pressurizer which is stainless steel clad. The design pressure for the system is 2,750 psig at 650°F. Maximum operating pressure is 2,500 psig for non-nuclear operation and 2,330 psig for nuclear operation.

The permissible rate of pressurization and depressurization of the primary system is 1%/sec during normal operation. During startup, and shutdown the maximum rate of heating and cooling is 100°F/hr.

The system is insulated on the outside with calcium-silicate type insulation designed for an outside insulation temperature of 140°F when the primary coolant is 600°F. A moisture protecting jacket covers the insulation to allow decontamination of its exterior surface.

b. System Components

(1) Primary Loop Piping. Primary loop piping is fabricated from fire-box quality, carbon steel, seamless pipe. The 20-in. OD (16.5 in ID) pipe was selected to limit water velocity to 25 to 30 ft/sec and is designed in accordance with the latest applicable revisions to Section III of the Code for Pressure Piping (ASA-13-31-3). All pipe and fittings are fabricated from ASTM-A-106 grade C seamless material with all welded construction used throughout the system.

(2) Steam Generator. The steam generator is a vertical U-tube heat exchanger 15 ft high and 6 ft in diameter. The primary coolant passes through the tubes transferring its heat to a secondary coolant on the shell side. Differential thermal expansion between the tubes and between the tubes and the shell is allowed by the U-tube design. The secondary coolant is demineralized water which boils at 340 psia at full load. The water level is controlled on the secondary side during operation. In case of tube failure, the pressure in the secondary side will be limited by relief valves. Small leaks in tube joints or tubes are detected by an increase in make-up to the primary coolant system or by a fall in the pressurizer water level.

(3) Pressurizer. The pressurizer vessel contains replaceable immersion-type heaters rated at 200 kw and a water spray system. A standpipe connects the pressurizer to the primary coolant loop at the reactor outlet piping. A bypass line connects the primary pump discharge to the spray nozzle in the top of the pressurizer. Safety valves and vent valves for the pressurizer discharge into a quench tank if excessive pressure develops in the primary coolant system. Enough water is maintained in the pressurizer to keep the heaters fully immersed at all times.

Upon expansion of the primary coolant during heatup or transient load changes, water flows from the primary loop, through the standpipe and into the lower end of the pressurizer. This causes the steam in the top to be compressed. The compression causes some condensation tending to reduce the pressure surge. Upon contraction of the water in the primary coolant system, water flows out of the pressurizer into the primary loop. Simultaneously a portion of the remaining water evaporates thus tending to maintain pressure. The heaters are energized automatically to maintain pressure at its normal value.

Variations in pressurizer water level control the liquid flow into or out of the primary coolant system. High level causes the blowdown valve to admit primary coolant to the quench tank. Low level activates the make-up water pump to keep the heaters submerged.

The spray nozzle in the pressurizer performs the following functions:

(1) Provides a means by which a small stream of primary coolant can be passed through the pressurizer so that the immersion heaters can assist in raising the primary coolant temperature during non-nuclear startup and operation.

(2) By introducing a small spray system of relatively cold water into the pressurizer during normal operation, non-condensable gases can be dissolved in the water in the pressurizer. When the non-condensables build up a high pressure they are vented to the quench tank. The non-condensables in the quench tank are vented to the stack.

(3) The sprays act to reduce the magnitude and duration of a rising pressure transient by promoting both mixing action and condensation.

Since the pressurizer operates at temperatures exceeding the rest of the primary system and contains both liquid and vapor phases the internal surfaces are clad with stainless steel. The pressurizer is designed for the saturation temperature (738 F) at 2,750 psig.

(4) Primary Coolant Pump. Primary coolant is circulated by a centrifugal, single-staged, canned, electric motor-driven pump with a capability of 5.6×10^6 lb/hr of water. The total discharge head equals the sum of the pressure drops through the reactor pressure vessel (20 psi), primary piping, primary heat exchanger and the flow trimming valve. The design conditions for the pump casing are 2,750 psig at 600 F.

The pump operates at constant speed so that coolant flow is essentially steady. Changes in load are attained by varying the pressure at which steam is generated on the secondary side of the steam generator.

A check valve is located at the primary pump discharge to prevent bypassing flow from the emergency decay heat removal pump when the primary pump is inoperative.

(5) Valve for Trimming Primary Coolant Flow. A 16-in. rotary gate valve is located between the venturi flow meter and the reactor vessel inlet. It is used to trim the flow through the reactor since the primary pump speed can not be varied. The valve is designed to fail in the open position. The pressure drop across the valve does not exceed 25 psi at a 5.6×10^6 lb/hr flow rate.

(6) Relief Valves. The relief valve which prevents overpressuring the primary coolant system is connected to the pressurizer above the water level. It is sized for 5% of full load steam generation in order to remove reactor decay heat after scram. The steam is blown down into the quench tank.

An electromatic relief valve is also connected to the pressurizer. Vapor may be blown down remotely through this valve at any time during operation.

(7) Coolant Purification System. The coolant purification system removes corrosion and fission product impurities from the primary coolant system. As shown in Figure IV-6, a portion of the primary coolant (709 gpm) leaving the primary coolant pump is cooled from 528 F to 130 F by passing in series through the tubes of a regenerative and a non-regenerative heat exchanger, after which the coolant passes through one or both of two ion exchange demineralizer columns. The effluent from the ion exchange columns passes through the shell of the regenerative heat exchanger and then enters the suction of the primary pump.

The ion exchange resin in the demineralizer columns is of the mixed bed type with the anion resin in the lithium form. Lithium form cation resin is selected to provide high pH (9.0 to 10.5) effluent. No provision is made for regeneration of the spent resin. However, means are provided for remotely removing spent resin by sluicing with demineralized water and replacing with fresh resin. Two columns are provided so spent resin from one column can be removed and replaced with fresh resin, while the other column is in operation. The ion exchange columns are each 14 in. in diameter with a 30-in. deep resin bed and are rated at 30 gpm and 130 F (max).

The activity level in the primary coolant system following a fission break in the fuel cladding can be reduced by a factor of 100 in approximately six hours.

The non-regenerative heat exchanger is designed for demineralized water at the primary coolant system design pressure on the tube side and service water on the shell side. The regenerative heat exchanger is designed for demineralized water at full primary system design pressure on both sides. Both the regenerative and non-regenerative exchangers are constructed of carbon steel.

(8) Chemical Addition and Water Make-Up System. This system adds chemicals for oxygen removal and pH control and demineralized water to the primary system. The demineralized water and chemicals

are added through a make-up pump. It is contemplated that hydrazine will be added for oxygen removal when the reactor is not operating at significant power levels. When the reactor is in operation, hydrogen will be added to the system for oxygen removal.

The make-up pump is capable of delivering 5 gpm at full primary coolant system design pressure, with suction at one atmosphere and 60°F, and is actuated by low liquid level in the pressurizer. The make-up pump is located outside the containment building.

(9) Experimental Blowdown Nozzles. Nozzles for conducting the loss of coolant tests are located on the inlet and outlet primary coolant piping near the reactor pressure vessel exterior wall. These nozzles have an area equivalent to the cross sectional area of the primary coolant piping. Space is provided on the dolly for the rupture mechanism and expulsion flow measurements.

An energy dissipator is mounted near the blowdown nozzles. No water or other liquid is used as part of the energy dissipation system because such an arrangement would tend to remove halogen and particulate matter and act as a pressure suppression mechanism.

(10) Sample and Instrumentation Nozzles. Sample nozzles are provided for insertion and removal of samples from the primary coolant piping in the TAN hot shop after completion of the blowdown experiment. Instrumentation nozzles are also located throughout the primary system to obtain experimental information on coolant expansion processes during the blowdown by using magneto-hydrodynamic and anemometry techniques.

2. Emergency Coolant System

The emergency coolant system is designed to provide decay heat removal from the reactor core when the primary coolant pump is inoperative. The decay heat pump circulates the primary coolant through the reactor core and primary heat exchanger to remove decay heat under the following conditions:

- a. Normal electric power failure
- b. Mechanical failure of the primary coolant pump

- c. Differential pressure across the primary pump is less than the discharge pressure of the emergency decay heat removal pump.

Any of the above conditions will result in a reactor scram when operating in the power range. The emergency decay heat removal pump is a single stage centrifugal pump of the canned-rotor type, with a capacity of 3.0×10^5 lb/hr at suction conditions of 2,280 psig and 528 F. Total discharge head of the emergency decay heat removal pump is to be the sum of the pressure drops at (3.0×10^5 lb/hr) through the reactor pressure vessel, primary and shutdown heat removal piping and the primary heat exchanger.

To ensure the availability during an emergency of the decay heat removal pump it is connected to the emergency electric power, and will operate continuously when the primary pump is operating. Since this pump will be arranged in parallel with the primary coolant pump and the head developed will be less than that of the primary coolant pump, a bypass line with a restricting orifice is provided to permit a circulation flow of 10 gpm to prevent overheating.

Demineralized water circulated through the secondary coolant system by an auxiliary secondary pump removes decay heat from the steam generator.

D. Instrumentation

1. Nuclear Instrumentation and Control

a. Reactor Safety System

The reactor safety system will prevent the occurrence of any plant condition which can lead to core damage or a radiological hazard. It functions independently of the plant control system within practical limitations. Nuclear instrumentation consists of seven independent channels of neutron monitoring equipment for use from source level through full power operation. The system contains two startup range channels, two intermediate range channels, and three power range channels with associated indicating, recording and trip circuitry. The reactor safety system also includes control rod

interlock protective circuitry and instrumentation channels which can initiate a scram (automatic reactor shutdown). Scram action occurs when plant conditions approach allowable limits.

The instrument block diagram for the reactor safety system is shown on Figures IV-8 and IV-9. The system has redundant and coincident design features so that:

- (1) No single component failure shall prevent automatic scram when it is needed.
- (2) Inadvertent operator action cannot prevent automatic scram when needed.
- (3) Minimum reliance on the plant operators shall be necessary to assure plant safety.
- (4) Coincident reactor safety system design minimizes the number of false scrams due to component or power supply failure.
- (5) Safety system channels shall be designed for "fail safe" operation (failures initiate channel scram trip).
- (6) Any single scram channel shall be capable of being tested for operability and of being removed for repair during power range operation.
- (7) The reactor safety system shall maintain simplicity and economical design consistent with meeting the system's functional requirement.
- (8) Automatic rod withdraw interlocks shall prevent all rod movement until normal operating temperature and flow exists in the primary coolant loop, until a minimum source range count rate is established and until acceptable overlap between startup and intermediate ranges is achieved.

Neutron detectors for the nuclear instrumentation channels are installed in guide tubes located in the neutron shield tank. Detector locations are chosen to minimize control rod shielding of detectors and to provide a adequate neutron flux level measurement over the appropriate ranges.

All nuclear instrumentation and safety channels are powered from independent electrical sources and a failure in any one channel does not affect the operation of the remaining channels. In general, only static devices are used in nuclear instrumentation and logic circuits.

The overall response time from detector to output relay trip of any channel does not exceed 250 milliseconds (excluding the detector response time). Response times are measured at 10,000 cps in the startup range, at 10^{-6} amp in the intermediate range, and at 20% power in the power range.

Each reactor safety channel has provisions for insertion of test signals which serve to calibrate and test the entire channel. Interlocks prevent the possibility of simultaneously testing more than one safety channel for each scram parameter.

The circuit logic for the rod withdraw interlocks is shown on Figure IV-8. The low count rate rod interlock is caused by low scale alarm contacts on the count rate recorders. Startup range detector power supplies are automatically de-energized when the intermediate range recorders begin to indicate on scale (with startup range overlap). If the intermediate range flux recorders have not begun to indicate by the time the startup range recorders reach full scale, a rod withdrawal interlock occurs. Also, the short period scram and alarm functions are automatically disabled during power range operation.

Startup and intermediate range channel accuracies do not exceed $\pm$ five percent for trip actions and $\pm$ six percent for indicator recording accuracy. Power range channels do not exceed $\pm$ three percent for trip actions and $\pm$ five percent for indication or recording accuracy.

Process safety channels consist of: (1) low outlet loop pressure and (2) high outlet loop pressure.

b. Reactor Control

The reactor control system utilizes both inherent reactor characteristics and external controls to regulate reactor power.

Manual rod withdrawal is used to bring the reactor into the power range of operation (10 to 100% of full power). During power range operation, the reactor is maintained at an average primary coolant temperature (T_{avg}) either manually, or automatically by a T_{avg} controller. Reactor power is controlled by the heat removal control system.

Pressurized water reactors with a negative temperature coefficient are inherently stable because of the degenerative effect of the temperature coefficient feedback loop. Simulator studies and operating experience for pressurized water reactors show that reactor control reduces to maintaining a desired average coolant temperature. The negative temperature coefficient helps to maintain the average coolant temperature in the core at a constant value. When steam flow is suddenly increased, the primary coolant temperature returning to the core decreases resulting in a lower coolant T_{avg} . This increases the core reactivity which leads to increased power generation until T_{avg} returns to its initial value. Thus, the reactor responds to steam flow changes using only temperature coefficient control. Decreasing steam flow transients cause the core to stabilize at a new and lower power level in a similar manner.

Control systems stability, instrument ranges and control set point values for both primary and secondary coolant systems will be analyzed in detail by means of analog simulation of the reactor processes and instrumentation systems. Design will be optimized for transient conditions which may be encountered during emergency conditions resulting from equipment failure or operator error.

c. Radiation Monitoring

The radiation monitoring system alerts plant personnel any time contamination or direct radiation reaches an unsafe level during normal plant operation. It will also be used to indicate the level of gamma activity inside the containment shell during and after the planned blowdown experiments. These functions are accomplished with the detection systems listed below:

- (1) Stack monitoring system
- (2) Area radiation monitoring system

- (3) Air particulate monitoring system
- (4) Extended range gamma monitoring system
- (5) Fission products water monitoring system
- (6) Secondary coolant monitoring system
- (7) Waste disposal monitoring system
- (8) Stationary personnel monitors
- (9) Wind speed, direction, temperature, and humidity monitoring system.

The equipment is located mainly in the main control room with detectors and auxiliary equipment distributed generally throughout the facility.

All equipment is considered as permanently installed except for the air sample monitor and area monitors located in the containment building. These units are removed from the containment building during blowdown tests and replaced when conditions have returned to normal.

Each instrument trip point and alarm, except stationary personnel monitors, activates an audible alarm and an indicating light in the main control room. All local detector installations are provided with audible and visual trip point and alarm indicators as well as actual count rate indication.

Trip points are determined after installation considering normal background count rate at each detector location. In general, after each system is initially calibrated the normal average background is determined as well as the normal statistical variation about this average reading. Each trip point is then set at a value equal to 2.5 times the normal average background.

(1) Stack Monitoring System. Two separate stack monitoring systems are used to improve reliability and accuracy. Both systems are identical, except that one has a linear scale readout. Both linear and log equipment trip points activate alarm circuits. The stack monitoring system initiates automatic containment isolation whenever the gaseous activity concentration reaches a level which would result in excessive radiological consequences to the public.

The linear scale instrument has a range from 0 to 200,000 cpm while the log scale instrument measures from 20 to 200,000 cpm on a four decade scale. Accuracy is $\pm 20\%$ of the numerical reading for the log scale instrument and $\pm 5\%$ of full scale deflection for the linear.

Air is drawn through a constant flow nozzle in the lower portion of the stack, passes through the monitor and back into the lower portion of the stack. As the gas enters the monitor, it passes first through a continuous particulate monitor, next through a radioactive gas monitor, and then out through the pumping system. Prior to entering the pump, a small portion of the gas is diverted through an iodine monitor. Scintillation detectors are used to monitor the gas streams.

The main equipment is located in a separate room or section of a room adjacent to the stack. It is as high in elevation as possible to allow minimum piping length from the input nozzle. The room is oriented to minimize electrical interference from high current and high voltage sources. Radiation shielding is provided such that access to the equipment is possible during reactor operation and shortly after the planned nuclear blowdown.

(2) Area Radiation Monitoring System. The area radiation monitoring system indicates, at the panel in the main control room, the ambient radiation dose rates from approximately 20 remote stations located strategically throughout the plant. Halogen quenched GM tubes are used in the remote units. Each remote unit is provided with an indicating meter, alarm light and calibrating gamma source.

(3) Air Particulate Monitoring. Five air particulate monitors are located at strategic points determined from heating and ventilation air flow. Scintillation detection techniques are used. Units use either a fixed or moving filter, depending upon the dust content of the air anticipated at each location, and activate alarms in the main control room.

(4) Extended Range Gamma Monitoring System. Six extended range gamma monitors with associated pre-amplifiers are located in the containment building. Five of the units are wall

mounted and the sixth is mounted approximately 5 ft. above the head of the reactor pressure vessel.

Detecting units are of a special design to insure proper operation in ambient conditions up to 275°F, 40 psig and 100% relative humidity. The unit consists of a ion chamber and pre-amplifier combined to form a cylindrical assembly approximately 3 in. in diameter by 10 in. long inside a thin-walled plastic canopy.

(5) Primary Coolant Monitoring System. Primary coolant activity is monitored by using ion-exchange resins to separate the short-lived iodine isotopes. The radioactivity of the resulting sample, measured by a scintillation detector with properly set discrimination, indicates the presence of fission products in the primary coolant stream. The water sampler contains two ion exchange columns for separation of cations and anions. It also contains a filter, flow controls, pressure regulator, check sources, shielding and sample taps.

The water sampler and detector are placed in the containment building at the ground floor level, as far removed and shielded from the reactor as possible. Stainless steel tubing is used for connecting inlet and outlet to the primary coolant loop. Flow through the system is throttled to 100 to 200 cm³/min. The piping system is of welded construction with a 3000 psi design pressure. The system design is 160°F to allow a safety margin over the 130°F water supplied to the coolant purification system. The inlet sample line length introduces a one or two-minute lag to allow the 7.3 sec nitrogen-16 activity to decay through several half lives.

(6) Secondary Coolant Monitor. A gamma scintillation detector is located so that a canned crystal is submerged directly in the secondary coolant. The detecting unit is installed in a continuous bleed line which takes approximately 3 gpm from the secondary coolant line after it leaves the air-cooled condensers. The bleed water is cooled to below 120°F before it enters the detecting unit. The sample water can be routed to other water analyzing equipment before being drained to liquid waste.

The detector is equipped with a pre-amplifier that feeds a log count rate amplifier and meter in the radiation monitoring panel in the main control room. A check source which is operated remotely from the radiation monitoring panel, is installed with the detector. Audible and visual alarms are activated in the main control room when coolant activity exceeds a pre-determined trip level.

(7) Waste Disposal Monitor. This system consists of a sampler assembly and an amplifier power supply unit.

The detector assembly is located in the piping at the entrance to the disposal well. It is arranged in a trap configuration so that flows keep the detector full and flows pass through readily.

Working in conjunction with the alarm circuit is an interlock circuit which stops the hot waste storage tank pump, the coupling room sump pump, the liquid waste pump and the cold water sump pump. The interlock circuit also closes the line which allows containment building drains to flow directly to the disposal well.

A measurement is obtained which is proportional to the concentration of radioactivity and rate of fluid flow. This is accomplished by direct addition of the log count rate signal and a single equivalent to the log of fluid flow. This signal is read out on a strip chart recorder using three-decade log paper. The value of accumulated activity for any period of time can be found by graphical integration of the strip chart record.

(8) Stationary Personnel Monitors. Stationary personnel monitors are of two types; portal monitors, and hand and shoe monitors. Two portal monitors are installed, one at each personnel exit, to provide a quick and efficient method of head-to-toe monitoring of personnel as they leave the area.

Two hand and shoe monitors are provided. One monitor is located in the access tunnel between the change room and the containment building while the other is located at the clean exit from the change room. They provide simultaneous automatic detection and measurement of beta-gamma contamination on the palm and back of each hand and on the bottom surface of both shoes.

(9) Wind Speed, Direction, Temperature and Humidity Monitoring System. A meteorological tower is

used to obtain measurements of wind speed, direction, temperature and humidity following the controlled release of activity from the LOFT test building.

d. Main Control Room

All instrumentation and control equipment pertinent or vital to normal operation of the reactor is contained in the main control room. The main control room contains components for the reactor safety system, reactor control and radiation monitoring.

The main control console is located in the approximate center of the control room with auxiliary rack-mounted equipment along both sides and at the back of the room. Both the main console and the auxiliary rack-mounted equipment have rear-access through-hinged back panels to facilitate installation and maintenance.

A plant annunciation system provides centralized monitoring of all important safety and control parameters on the main control room panels. Highly reliable visual and audible annunciation of plant conditions provides immediate operator acquaintance with plant performance. Indication, annunciation and overriding controls for critical utilities are provided in the main control room.

2. Process Instrumentation

a. General

The primary coolant system instrumentation performs the following functions:

- (1) Indication of primary coolant system parameters
- (2) Manual control
- (3) Automatic control
- (4) Audible and visual annunciation of abnormal system parameters.

- (5) Interlock and permissive features depending on system parameters and the operator's discretion
- (6) Spare auxiliary trip contacts
- (7) A semi-graphic reactor console with supporting panels
- (8) Switching circuits for rod control and indication

The primary coolant system instrumentation is powered from normal 120 v AC plant power with non-synchronous transfer to the emergency diesel bus on loss of normal plant power. All devices are designed to annunciate on loss of power or component failure and fail-safe.

All transducers and final control elements are located with the reactor equipment on a railroad-type dolly within the containment building and are subject to the containment shell ambient conditions. All other devices are located in a main control room on one of the four following panels:

- (1) Reactor control console
- (2) Primary coolant system instrument panel
- (3) Primary coolant system instrument auxiliary panel
- (4) Reactor control console auxiliary panel

b. Instrumentation Systems

The primary coolant instrumentation includes the systems tabulated below. All components within each system are provided with individual branch circuit fuses.

(1) T_{avg} Controller. The primary function of this system is to provide automatic rod operation for correction of long term core reactivity effects within the power range of operation (10 to 100% power). It also provides indication of T_{avg} and narrow range reactor inlet (C) and reactor outlet (H) temperatures with alarms.

(2) Reactor Pressure Drop. The main function of this system is the recording and annunciation of an abnormal pressure drop across the reactor vessel.

(3) Wide Range Temperature Instrument. This instrument system records reactor inlet and outlet temperatures and annunciates abnormal values.

(4) Pressurizer Temperature Controller. This system records and annunciates abnormal pressurizer temperature and provides automatic pressurizer heater cutoff with manual override at high temperature.

(5) Pressurizer Pressure Controller. This system records and annunciates high and low pressure, controls the pressurizer cycling and backup heater elements, controls the pressurizer spray valve with manual override, and controls the electromatic relief valve.

(6) Pressurizer Level Controller. This system provides level indication with abnormal high and low level alarm. It also provides low level heater cutoff with manual override, make-up pump control with manual override, and discharge valve control with manual override.

(7) Indicating and Control Circuits. These circuits provide remote manual control and indication of the operating condition of all primary system valves, pumps, and equipment. A multipoint recorder (calibrated in percent of design range) is used to record trends of selected parameters.

3. Experimental Instrumentation

a. General

The objectives of a complex experimental effort such as the LOFT program requires the proper selection of experimental measurements and the location of measurement instrumentation. Typical measurement requirements include:

- (1) Amplitude accuracy
- (2) Event-time accuracy

- (3) Time-history accuracy
- (4) Dynamic range
- (5) Spatial gradient coverage

For optimum selection of detectors to accommodate the measurement requirements the following detector capabilities are considered:

- (1) Transducing accuracy
- (2) Response time
- (3) Dynamic or useful range
- (4) Environmental sensitivities

The measurements have been classified into two groups which relate to accuracy, dynamic range, and location. The first group, "operational", refers to all measurements and instrumentation required prior to the programmed loss of coolant. The second group "experimental", refers to all measurements and instrumentation required after the loss of coolant. Instruments from both groups sometimes are used to measure the same parameter. For example, two temperature measurements are made at the same point by a detector from both groups. This results from the requirement of high accuracy during normal operation and the requirement for extremely high temperature measurements during the meltdown period. The detector from the operational group is a chromel-alumel thermocouple and from the experimental group a platinum-platinum-rhodium thermocouple. The measurements are divided into five major categories according to the location of the sensing device. These categories are: (1) in-core, (2) in-vessel, (3) primary coolant system, (4) inside the containment shell, and (5) exterior to the containment shell.

To provide flexibility in the final allocation and installation of all measurement detectors, termination stations are located at various locations on the dolly and in the test area. From these stations the instrument cables run to a master "pressure barrier termination", and from there to the Experimental Instrumentation in the Control and Equipment Building.

The termination stations contain electronic equipment such as pre-amplifiers, thermocouple reference junctions, cathode followers etc. The presence of the electronic equipment and electrical connections makes it necessary for these stations to be waterproofed and pressurized slightly above the containment design pressure with dry air. Since these stations contain raw detector signals in the microvolt region, no alternating current power is contained within the station other than that required for the electronic equipment. Most transformers and power lines are located remotely to these stations and to the conduit returning to the pressure barrier termination.

A description of the detector types, number of detectors, and the location of the operational and experimental measurement instrumentation is presented in the following sections. A summary of the operational and experimental measurements and instrumentation is presented in Table IV-D and the terminating stations in Figure IV-10.

b. In-Core Measurements

The various in-core measurements to be made in LOFT are tabulated below:

(1) Nuclear. In-core nuclear measurements are needed during startup, operation, and during the loss of coolant accident. Power, period, and flux measurements are taken. Miniature ionization chambers in the removable sub-assemblies traverse the length of the core vertically to give the vertical flux profile. The detectors are radially positioned to completely map one octant, one diameter, and one diagonal. Detectors are located at selected positions in the remaining sections of the core to verify symmetry, and to give a flux measurement in each quadrant of the core. The cross-arms of the cruciform control rods are slightly shorter than the width of the adjacent fuel assemblies. Small nuclear chambers are also inserted in these spaces.

(2) Temperature. Operational and experimental fuel, clad, coolant, and structural temperature measurements are made throughout

TABLE IV-D LOFT Operational and Experimental Instrumentation

Measurement	Detector	Location	Number of Measuring Points	Termination Station(s)
<u>In-Core</u>				
Nuclear	Neutron - Teleflex-Type	Assembly Centers	20	B,C
Temperature	Pt-Pt-Rh Tc's	Fuel Material (Experimental)	30	B,C
	Chromel-Alumel Tc's	Fuel Material (Operational)	18	B,C
	Chromel-Alumel Tc's	Fuel Clad (Operational)	18	B,C
	Pt-Pt-Rh Tc's	Fuel Clad (Experimental)	66	B,C
	Pt-Pt-Rh Tc's	Assemblies Centers	19	B,C
	Pt-Pt-Rh Tc's	Between Assemblies	12	B,C
	Pt-Pt-Rh Tc's	Shroud	8	B,C
	Pt-Pt-Rh Tc's	Control Rods	4	B,C
	Chromel-Alumel or Resistance Tc's	Coolant Inlet and Outlets	75	B,C
Pressure	Bellows Type	In-pins - Diameter Diagonal	10	B,C
	Tubes	Coolant in Assemblies	30	B,C
	Tubes	Between Assemblies	18	B,C
Velocity	Probes	Inlet	8	B,C
	Probes	Outlet	8	A
	Probes	Between Assemblies	5	B,C
Strain	Strain Gauges	Pins	24	B,C
	Structural	Grids and Assembly Shrouds	21	A,B,C
		Total In-Core Measuring Points	413	
<u>In-Vessel</u>				
Nuclear	Neutron and Gamma Chambers	Reactor Shield Tank	13	A
Temperature	Pt-Pt-Rh and Chromel-Alumel Tc's	Core Flow Skirt	12	A,B,C
		Inner and Outer Thermal Shield	24	A,B,C
		Upper and Lower Grid Support Barrel	16	A,B,C
		Upper and Lower Support Grid	18	A,B,C
		Instrument Grid	8	B,C
		Water Baffles	6	A
		Vessel Walls, Head and Bottom	62	A,B,C
		Axial Positioning	4	A,B,C
		Skirt	12	A,B,C
		Instrument Grid	4	A,B,C
Pressure	Tubes - Transducer	Upper and Lower Support Barrel	18	A,B,C
		Inner and Outer Thermal Shield	6	A,B,C
		Vessel Walls, Head and Bottom	24	A,B,C
Strain	Strain Gauges	Vessel Walls, Head and Bottom	24	A,B,C
		Top and Bottom	3	A,B,C
Vapor	Calorimeter	Top and Bottom	3	A,B,C
Coolant Density		Top and Bottom	3	A,B,C
Liquid Level	Conductivity Probes	Vessel Bottom	2	A
Condensation		Vessel Bottom	2	B,C
		Total In-Vessel Measuring Points	237	

TABLE IV-D LOFT Operational and Experimental Instrumentation (Cont'd)

Measurement	Detector	Location	Number of Measuring Points	Termination Station(s)
<u>In-Primary Coolant System</u>				
Nuclear Temperature	Gamma and Neutron Chambers Chromel-Alumel Tc's	Reactor Shield Tank	3	A
		Water	12	A
		Shield Wall	12	A
		Reactor Vessel Outer Wall	90	A
		Insulation	12	A,B,C,D,E,F,G
		Coolant Piping	12	A,B,C,D,E,F,G
		Vessel Outer Surface	90	A
		Coolant Piping	20	A
		In-Coolant Piping	30	A
		In-Pipe Space	2	A
Strain Pressure Velocity Mass Flow	Strain Gauges Transducers in Tubes Hot Wire Anemometer Orifice Load Cell	At Rupture	4	A
		Water Tank - Intr. Outlet Nozzle	36	A
		Along Pipe Length		
		Outlet to Vessel	10	A
		At Rupture - Inlet and Outlet	3	A
		At Rupture - Inlet and Outlet	3	A
		In-Pipe Space	4	A
		In-Pipe Space	4	A
		Total In-Primary Coolant System Measuring Points	347	
		<u>Inside Containment Shell</u>		
Nuclear Temperature Pressure Strain	Neutron and Gamma Chambers Chromel-Alumel Tc's Microphones Strain Gauges	X, Y and Z Axes	20	D,E,F,G
		Walls - Dome - Concrete	13	D,E,F,G
		X, Y and Z Axes	26	D,E,F,G
		Vessel Walls and Dome - Door	137	D,E,F,G
		Flatcar	14	A
		Penetrations of Walls	4	D,E,F,G
		Total Inside Containment Shell Measuring Points	214	
<u>External to Containment Shell</u>				
Pressure Strain Temperature	Microphones Strain Gauges Chromel-Alumel Tc's	Radial to 300 ft	6	K,L,M
		Vessel Wall and Dome - Door	37	K,L,M
		Walls - Dome	29	K,L,M
		Total External Containment Shell	72	

the core. Operational clad temperatures are determined at vertical positions diametrically across the core. Experimental temperature measurements are made in the fuel at vertical positions diagonally across the core and on the cladding at vertical positions along one diameter, one-half diagonal, and at selected points in each quadrant. Coolant temperatures are measured at inlets and outlets of the assemblies. Thermocouples located in assemblies along one diameter are used to determine the coolant temperature in the center of the fuel pin array. Structural temperature measurements are made on the fuel assemblies.

Operational temperature measurements require the use of chromel-alumel thermocouples which give a high degree of accuracy. The experimental temperature measurements are made by platinum-platinum-rhodium thermocouples which do not have the high degree of accuracy of chromel-alumel thermocouples, but are usable over a higher temperature range.

The determination of experimental fuel, clad, and structural temperature measurements in the molten region, ($\sim 5000^{\circ}\text{F}$), will be attempted by developmental tungsten-tungsten-rhenium thermocouples. These measurements are made at the above experimental locations and in the region of the bottom head.

(3) Pressure. A pressure transducer is positioned in the end of one fuel pin in each assembly along one diameter and along one-half diagonal. Coolant pressures are determined at inlet and outlet positions in several assemblies.

(4) Strain. In-core strain measurements are made by strain gauges positioned on pins located in assemblies along one diameter at the center position and on at least one position below and above the center of the pin. Several gauges are attached to the assembly shrouds. Structural strain measurements are made on the upper and lower grids to determine the vibrational and thermal stresses due to blowdown and heatup.

(5) Velocity. Coolant velocities due to blowdown and to transient heatup are determined at inlet and outlet positions along one diameter and at positions along coolant channels between fuel assemblies. Operational measurements are made in the inlet and outlet of the core to determine the normal coolant flows before the loss of coolant accident.

c. In-Vessel Measurements

The various in-vessel measurements to be made in LOFT are tabulated below:

(1) Nuclear. Startup, operational, power, period, and flux measurements are made at increasing distances from the core. Miniature nuclear chambers are positioned along the axis running vertically through the core center.

Gamma radiation measurements are made by gamma chambers located at the top and bottom of the core at positions where gamma heatup may significantly affect the vessel and core structure or instrumentation.

(2) Temperature. Temperature measurements are made in the coolant, one the flow skirt, thermal shields, upper and lower support barrels, upper and lower core support plates, instrument lead and shock absorber support plate and barrel, diffuser baffle, vessel walls and on the top and bottom heads. These measurements are made to determine the temperatures occurring on support structures, the heat absorption of the thermal shields, the cooldown temperatures from blowdown, and the heatup temperatures from decay heat.

(3) Pressure. Pressures resulting from blowdown and the subsequent steam evolution are measured by pressure transducers positioned axially above the instrument lead and shock absorber support plate. They are also located between the instrument lead and shock absorber support plate at the upper core support plate, and below the lower core support plate.

(4) Strain. The stresses exerted on the vessel and internal structures resulting from vibrations during blowdown, thermal expansion resulting from the transient heatup, steam pressure, and the weight of supported components of the core are measured by strain gauges. Longitudinal, radial, circumferential, and diagonal stresses are determined.

Strain gauges are positioned on the flow skirt and on the instrument lead and shock absorber support barrel. They are also located on the upper and lower support barrels, and inner and outer thermal shields, vessel walls, and at the top and bottom head.

(5) Mass Flow. Mass flow measurements are made between the instrument lead and shock absorber support plate. Mass flow measurements are also obtained at the upper support plate, near the center of the vessel, and near the coolant inlet-outlet nozzles.

(6) Coolant Property. Coolant density measurements are made to determine coolant quality.

(7) Miscellaneous. Conductivity probes are used to determine the liquid level remaining in the bottom of the vessel following the blowdown phase. An electrical contact method is used to measure the condensation and plateout on the walls of the reactor pressure vessel.

d. Primary Coolant System Measurements

The various primary coolant system measurements to be made in LOFT are tabulated below:

(1) Nuclear. Gamma and neutron chambers are located at various heights in the reactor shield tank surrounding the reactor pressure vessel. These chambers provide information concerning the flux profile with respect to distance from the core.

(2) Temperature. Thermocouples are attached to the outer vessel surface, vessel insulation, in the water shielding, and at the center, top and bottom of the reactor shield tank. The

coolant piping is instrumented with thermocouples along its entire length on inner and outer surfaces, and on system components including valves and pumps. Other temperatures are taken at the inlet and outlets to the reactor pressure vessel and on both sides of the rupture.

(3) Pressure. Blowdown and heatup pressures are determined by transducers at the inlet nozzle, outlet nozzle, along the primary coolant pipe, and at the point of rupture.

(4) Strain. Vibrational strain during blowdown and thermal stresses during heatup are measured by strain gauges on the reactor pressure vessel outer surface, reactor shield tank surface, outlet and inlet coolant nozzles, and along the coolant piping particularly near the point of rupture.

(5) Velocity. Flow velocity measurements are made by hot wire anemometers on a grid network in the primary coolant loop and at near the rupture point.

(6) Mass Flow. The routine operational mass flow measurements are made by orifices in the in-pipe space. Expulsion mass flows following the rupture are measured by load cells in the primary coolant piping.

(7) Miscellaneous. Vapor state, coolant density, and condensation measurements will be made during blowdown and heatup. Thrust measurements using load cell detectors will be made to determine thrust resulting from the rupture in the coolant system.

e. Inside Containment Vessel Measurements

The various inside containment vessel measurements to be made in LOFT are tabulated below:

(1) Nuclear. An X, Y, Z array of nuclear chambers gives the flux distribution at increasing distances from the reactor pressure vessel. Gamma, health physics monitoring, and operational chambers are positioned inside the containment shell during the loss of coolant accident.

(2) Temperature. Thermocouples are located on the walls and dome of the vessel and in the concrete shield outside the vessel.

(3) Pressure. Air pressure measurements will be made at radial positions along the X and Y axis out along the centerline

(4) Strain. Circumferential, longitudinal, and diagonal strain measurements will be made on the vessel walls and dome. An extensive detector array is placed around and on the railroad door and all shell penetrations. Strain gauges on the bed and supports of the dolly yield information on the mechanical energy release from the blowdown forces.

(5) Visual. Television, movie, and pin-hole camera positioned in sealed enclosures are used for visual coverage of the point of rupture and pertinent portions of the system. Both standard and infrared film are used in the movie cameras. Camera and film protection against radiation, excessive temperature, moisture, and other adverse environmental conditions is provided by the shielded stations. Special sequence type cameras are used to detect "hot spots" inside the containment shell.

f. Exterior to the Containment Vessel Measurements

The various exterior to the containment vessel measurements to be made in LOFT are presented below:

(1) Pressure. Pressure transducers are located at positions up to 300 ft. from the containment vessel.

(2) Strain. Strain gauges are located in the concrete shield and at positions on the walls and dome. They are also located around the door and at all other vessel penetrations. These gauges provide longitudinal, circumferential, and diagonal strain measurements.

(3) Temperature. Thermocouples are placed on the dome and walls at distances of one-fourth, one-half, and three-fourths of the distance up the wall and also at the apex of the dome.

4. Containment Vessel Isolation

An instrumentation system for initiating the isolation of the containment shell is included in the overall plant design. This system is devised so that it can be activated either manually or automatically. Automatic activation of the system results from signals such as high stack activity, low primary coolant pressure, high pressure within the containment shell, etc.

Fig. IV-1 - LOFT Dolly Elevation and Plan View

Fig. IV-2 - LOFT Vessel Cross Section

Fig. IV-3 - LOFT Nozzle Summary

SECTION **1313**

SECTION **CC**

Fig. IV-4 - LOFT Vessel Sections

Fig. IV-5 - LOFT Fuel Element Assembly

LINE NO.	FULL LOAD OPERATING		
	PRESSURE PSIA	TEMPERATURE °F	FLOW GPM
1	2330	520	5,600,000
2	2330	520	5,600,000
3	2300	520	5,600,000
4	2280	520	5,600,000
5	2280	520	5,600,000
6	2280	520	5,600,000
7	2280	520	300,000
8	2300	520	300,000
9	2295	520	5,000
10	2330	520	31,000
11	2330	520	30,000
12	2320	308	30,000
13	2310	130	30,000
14	2310	130	15,000
15	2295	130	30,000
16	2280	350	30,000

Fig. IV-7 - Primary System Instrumentation Diagram

FLUX-FLOW SAFETY CHANNELS

OUTLET LOOP LOW PRESSURE SAFETY CHANNELS & SUMMER STAGES

OUTLET LOOP HIGH PRESSURE SAFETY CHANNELS

Fig. IV-8 - Reactor Safety Instrumentation Diagram

SAFETY SYSTEM START UP RANGE NUCLEAR INSTRUMENTATION

SAFETY SYSTEM INTERMEDIATE RANGE NUCLEAR INSTRUMENTATION

SAFETY SYSTEM POWER RANGE NUCLEAR INSTRUMENTATION

NOTES

1 - ALL COMPONENTS ARE LOCATED ON THE MAIN CONTROL ROOM SAFETY & NUCLEAR SYSTEM PANELS, UNLESS OTHERWISE NOTED.

2 - [1] - DENOTES MOUNTED ON DOLLY.

3 - [2] - DENOTES MOUNTED WITHIN CONTAINMENT SHELL.

4 - [3] - DENOTES MOUNTED IN MAIN CONTROL ROOM.

5 - [4] - DENOTES MOUNTED ON REACTOR CONSOLE.

6 - S - DENOTES SCRAM.

7 - 2/3 - DENOTES 2 OUT OF 3 COINCIDENCE LOGIC.

8 - 8/5 - DENOTES B1 STABLE TRIP UNIT.

9 - (A) - DENOTES ALARM.

10 - FF - DENOTES FAILURE FREE POWER SOURCE.

11 - (PL) - DENOTES PILOT LIGHT.

12 - 10 - DENOTES FLUX IN % REACTOR POWER.

13 - K - DENOTES RELAY, SEE ELEMENTARY DWGS. FOR CIRCUIT DEVELOPMENT.

TABLE #1

TERMINAL STATION	NO. OF COAX	TC EXT. WIRE PAIRS	4 CONDUCTOR SHIELDED	SPECIAL MULTI COND.	REMARKS
A	20	150	200	30	Terminal Station for Detectors Located In-Core, In-Vessel and On Dolly
D	20	150	100	20	Terminal Station for Detectors Located In-Core and In-Vessel
C	20	150	100	20	SAME AS ABOVE
D	20	10	30	5	Terminal Station for Detectors Located In-Containment Bldg. and Test Cell Area
E	10	10	100	5	SAME AS ABOVE
F	10	20	30	5	Terminal Station for Detectors Located in Containment Bldg. Above Main Floor Level
G	10	20	30	5	SAME AS ABOVE
H					PLATINUM-RIIODIUM Thermocouple Temperature Reference Box
J					CHROMEL-ALUMEL Thermocouple Temperature Reference Box
K	10	30	50	5	Terminal Station for Detectors Located Outside Containment Bldg.
L	10	30	50	5	SAME AS ABOVE
M	10	30	50	5	SAME AS ABOVE

SECTION V

NUCLEAR DESIGN

V. NUCLEAR DESIGN

A. Reactor Physics

Reactor physics calculations for the LOFT reactor used two dimensional, four-group diffusion theory. Experience with light-water-moderated cores employing low enrichment UO_2 fuel elements has shown that this approach is satisfactory for conceptual design purposes. The results of similar calculations previously carried out for the MARTY critical experiment and the SPERT-I UO_2 core⁽¹⁾ were found to be in agreement with experimental data.

Thermal constants for neutrons with energies from 0 to 0.625 e.v. are based on the Wigner-Wilkins energy distributions and were calculated using the 650 SOFOCATE program⁽²⁾. An initial neutron energy spectrum is computed with the atom densities homogenized on a volume fraction basis. Constants for the pure materials within a fuel pin unit cell are then obtained by averaging their energy dependent cross sections over the previously calculated energy spectrum. With these constants, a P-3 spherical harmonics program⁽³⁾ is used to calculate a fine flux profile over the unit cell. These calculations determine flux disadvantage factors for the UO_2 and stainless steel cladding. A second Wigner-Wilkins spectrum is then computed with atom densities weighted by the integrated regional fluxes in the unit cell divided by the integrated cell flux. The constants obtained from this procedure allowed the fueled core region to be treated as homogeneous when considering thermal neutrons in the diffusion calculations.

-
- (1) IDO-16879, "Power Burst Facility (PBF) Conceptual Design", A. A. Wasserman, S. O. Johnson, R. E. Heffner and A. H. Spano (1963).
 - (2) IDO-16578, "650 SOFOCATE - A Code to Determine Thermal Constants on the IBM-650", R. J. Wagner (March 1962).
 - (3) "P₃ Flux Distribution", Phillips Petroleum Company 650 Program Library, File No. 8.2.014, J. Weil and P. Cabral.

The fast group constants are determined from the GAM-1⁽¹⁾. In calculating effective resonance integrals for U^{238} , the code takes into account geometric effects of fuel lumping and Doppler broadening. Except for this heterogeneity correction, the fueled core is assumed to be homogeneous with respect to fast neutrons.

Except for the control rod poison section, both fast and thermal group constants for all non-fuel interior regions are obtained by averaging the cross sections over the GAM-1 and the SOFOCATE fuel core spectrums respectively. Thermal constants for the control rod poison sections are calculated by assuming these sections "black" to thermal neutrons and using a logarithmic boundary condition. Fast group constants for the poison section are obtained from the MUFTR-D Blackness Coefficients Edit⁽²⁾. All constants for regions external to the core are calculated by averaging the cross sections over a water spectrum.

PDQ-3 computer code⁽³⁾ calculations in X-Y geometry with both quarter-core and full-core representations are used to establish the reactivity characteristics and neutron flux distributions for the LOFT core design. The quarter-core representation used for the PDQ calculations is shown in Figure V-1. To arrive at the required core reactivity, and hence the fuel enrichment, TURBO⁽⁴⁾ lifetime calculations are made. Preliminary calculations indicate that the reactivity requirement for fission product poisoning and fuel burnup is approximately $0.055 \Delta\rho$. Therefore, the neutron multiplication (k) at the beginning of core life must be about 1.058. To determine the fuel enrichment necessary, PDQ-3 calculations are made for various fuel enrichments in two distinct core

-
- (1) "GAM-1, A Consistent P_1 Multigroup Code for the Calculation of Fast Neutron Spectral and Multigroup Constants", GA-1850, G. D. Joanou and J. S. Dudek (June 1961).
 - (2) "MUFTR-D Blackness Coefficient Edit", Phillips Petroleum Company 650 Program Library File No. 00-111.
 - (3) "WAPD-TH-70 A Code to Solve the Two-Dimensional Few Group Neutron-Diffusion Equations", G. G. Bilodeau, W. R. Cadwell, J. P. Dorsey, J. G. Faricy, R. S. Varga (1957).
 - (4) WAPD-TM-221, "Nuclear Reactor Depletion Programs for the Philco 2000 Computer" (January 1961) O. J. Marlowe, et al.

regions. The outer ring of 20 fuel elements have a higher enrichment than the 32 inner fuel elements. The purpose of a two-zone fuel loading is to reduce peak power densities and flatten radial power distribution. Fuel melting in the core following coolant blowdown is thereby increased. Therefore, optimum fuel loading is considered as the enrichments which result in a minimum peak-to-average power ratio. Peak-to-average power behavior is shown as a function of outer fuel enrichment in Figure V-2. The crossover points of the upper and lower curves represent the minimum peak-to-average power attainable for a given set of enrichments.

The eigenvalue results for these problems are tabulated in Table V-A. In Figure V-3 these eigenvalues are plotted as a function of outer fuel enrichment. The intersections of the curves with $k = 1.058$, the desired eigenvalue, determine three pairs of enrichments which will give the necessary hot core reactivity. A normalizing factor of $1/1.02$ for k is used as similar calculations for the MARTY experiments resulted in calculated eigenvalues that were approximately 2% high. In Figure V-4 the outer fuel enrichment is plotted as a function of inner enrichment for the crossover points of Figures V-2 and V-3. The intersection of these two curves then determined the optimum fuel loading for the LOFT core. This loading is 2.82% for the inner zone and 5.71% for the outer zone.

The above calculations assumed a metal-water reflector. Recent gamma heating calculations indicate that metal filler pieces in the reflector region are unnecessary. Calculations now assume an all water reflector. Radial power peaking and core reactivity does not change significantly as illustrated by the following PDQ-40⁽¹⁾ calculations:

Reflector Composition	k	ρ	Power Density Ratio (Peak Pin/Average Pin)
Metal (63%)-Water (37%)	1.0568	0.0537	1.54
Water 100%	1.0531	0.0504	1.55

(1) "PDQ-40, A Program for the Solution of the Neutron Diffusion Equations in 2 Dimensions on the IBM-7040", Phillips Petroleum Company Program No. 40.0018.

TABLE V-A

EIGENVALUES AS A FUNCTION OF INNER AND

PERIPHERAL FUEL ENRICHMENTS*

<u>Fuel Region Enrichments</u>	<u>k</u>	<u>$\rho = \frac{k - 1}{k}$</u>
Inner Zone 2.7	1.0494	0.0471
Outer Zone 6.0		
Inner Zone 2.7	1.0324	0.0314
Outer Zone 4.0		
Inner Zone 2.8	1.0585	0.0553
Outer Zone 6.0		
Inner Zone 2.8	1.0509	0.0484
Outer Zone 5.0		
Inner Zone 2.8	1.0423	0.0406
Outer Zone 4.0		
Inner Zone 2.9	1.0497	0.0473
Outer Zone 4.0		
Inner Zone 2.9	1.0655	0.0615
Outer Zone 6.0		

* Moderator Temperature - 544°F
 Fuel Temperature - 1500°F

The above data indicate that a water reflector decreases the hot core reactivity by 0.0033 $\Delta\rho$. Axial calculations are not completed for the water case. However, if the assumption is made that the rod worth is unchanged by the water reflector, the critical rod bank position for the reactor at initial full power will be about 45 cm from the top of the active core. This decrease of approximately 2 cm in rod insertion depth will result in a correspondingly small decrease in axial peak power density. From calculations for the metal-water reflector, an axial peak-to-average power ratio of approximately 2.10 might be expected.

Temperature coefficients are determined from the results of four-group, one-dimensional diffusion theory calculations in cylindrical geometry. The normalized problem eigenvalues in Figures V-5, -6, and -7 are plotted as a function of temperature. The temperature coefficients are then obtained by differentiating the curves at the pertinent points. The eigenvalues of Figures V-6 and -7 are normalized such that the curves agree at 68°F with the cold core eigenvalue (1.1225) determined from PDQ-40. The eigenvalues of Figure V-5 were normalized to agree at 1500°F with the hot (50 Mw) core eigenvalue (1.053) obtained from PDQ-40. The results of those calculations are as follows:

Reactor Condition	Moderator Temp.	Fuel Temp.	Prompt Doppler Coefficient of Reactivity	Overall Temp. Coefficient of Reactivity (Includes Doppler)
Cold	68°F	68°F	$-1.5 \times 10^{-5} \Delta\rho/^\circ\text{F}$	$-0.57 \times 10^{-4} \Delta\rho/^\circ\text{F}$
Hot (zero power)	545°F	545°F	$-1.3 \times 10^{-5} \Delta\rho/^\circ\text{F}$	$-2.6 \times 10^{-4} \Delta\rho/^\circ\text{F}$
Hot (50 Mw)	545°F	1500°F	$-0.0 \times 10^{-5} \Delta\rho/^\circ\text{F}$	-----

The diffusion theory fast group constants are obtained with the GAM-1 Code. With fuel temperature as an input parameter, this code accounts for the Doppler broadening of the U-238 resonance in arriving at a few group cross sections. The thermal constants are determined using TEMPEST⁽¹⁾ after having calculated flux disadvantage factors for the unit cell materials with the TOPIC⁽²⁾ program.

(1) Phillips Petroleum Company Program No. 40.0010.

(2) IDO-16968, "TOPIC, A FORTRAN Program for Calculating Transport of Particles in Cylinders", G. E. Putnam. (To be published).

B. Reactor Control

The LOFT core employs twelve cruciform control rods of boron stainless steel, enriched to 1.5% by weight in B^{10} , with zirconium followers. The design is based on the requirements that sufficient control rod material be available to render the cold clean reactor at least 5% subcritical under normal operating conditions and to shut the reactor down in the event of a stuck rod in the full-out position. The reactivity worth of all twelve control rods is 23% with 36 dummy fuel pins containing a nuclear poison in the core. When the poison pins are replaced by fuel, rod worth is decreased to 21%. The worth of 11 rods (one rod stuck out) is 15.6% with poison pins in the core. On the basis of these preliminary studies, it is estimated that the worth of 11 rods is decreased to 14.6% when the poison pins are replaced with fuel pins. Cold core reactivity is increased by 0.0190 $\Delta\rho$ when all poison pins are replaced by fuel.

In general, it is found that zirconium filler plugs have practically no effect on rod worth or core reactivity, but reduce local power densities approximately 12% below those occurring with water slots. Eigenvalues and peak-to-average power densities calculated for a core containing 36 poison pins and the optimum fuel loading are listed in Table V-B.

TABLE V-B

LOFT EIGENVALUES AND PEAK-TO-AVERAGE POWER DENSITY - CLEAN CORE

k (cold)	1.128
k (hot, full power)	1.057
k (cold, 12 rods in)	0.896
k (cold, 11 rods in)	0.959
Radial Peak/Average Power	1.60 (Bank Rods)

C. Gamma Heating

Gamma heating calculations are made to determine the cooling flow required in regions external to the core and to analyze thermal stresses in the support barrel and pressure vessel. The calculations consider two sources: (1) primary gammas computed using the GRACE-1 computer

code⁽¹⁾, and (2) secondary gammas computed using the GRACE-II computer code⁽²⁾. The primary gamma sources considered are those appearing in the core from thermal and epithermal neutron capture, prompt fission, and fission product decay. The secondary gammas considered are those resulting from thermal capture in regions external to the core. Input information to these codes concerning energy group structures, prompt fission and fission product decay, gamma spectra, buildup factors, and attenuation coefficients were obtained from the work of Avery, et al⁽³⁾. One dimensional diffusion theory calculations are made to determine the primary gamma sources. The thermal flux exterior to the core is also determined with this calculational technique using as input the fast removal flux from previous GRACE-II results. GRACE-I is then used to calculate gamma heating in the external regions using the resultant thermal fluxes.

Figure V-8 presents a radial plot of thermal flux from the core out into the reactor shield tank and Figure V-9 gives the gamma heating plot from the core out into the reactor shield tank. The values of gamma heating are axial maximums, as they are based on the core midpoint values assuming an axial cosine distribution.

-
- (1) NAA-SR-3719, "GRACE-I, An IBM 704-709 Program Designed for Computing Gamma-Ray Attenuation and Heating in Reactor Shields", D. S. Duncan and A. B. Spier (June 30, 1959).
 - (2) NAA-SR-4649, "Grace-II, An IBM 709 Program for Computing Gamma-Ray Attenuation and Heating in Cylindrical and Spherical Geometries", D. S. Duncan and S. B. Spier.
 - (3) AERE-A3216, "Methods of Calculation for Use in the Design of Shield for Power Reactors", A. F. Avery, P. E. Bendall, J. Butler and K. T. Spinnery.

1 Thru 8. Fueled Core	14. Dummy Fuel Pin Unit Cell
9. Control Rods	15. Zirconium Plugs
10. Water Slots	16. Core Barrel, Thermal Shield And Water
11. Water And Cans	17. Reactor Vessel
12. Flow Skirt	18. Non-Diffusion Region (Flux = 0 at Extrapolated Vessel Boundary)
13. Reflector (Water And Stainless Steel)	

Fig. V-1 - LOFT Core Quarter Section

Fig. V-2 - LOFT Source Ratio Versus Peripheral Enrichment

Fig. V-3 - LOFT Eigenvalues Versus Pheripheral Enrichment

Fig. V-4 - LOFT Outer Enrichment Versus Inner Enrichment

Fig. V-5 - LOFT K_{eff} Versus Fuel Temperature (Moderator @ 545°F)

Fig. V-6 - LOFT K_{eff} Versus Core Temperature (Zero Power)

Fig. V-7 - LOFT K_{eff} Versus Fuel Temperature (Moderator @ 68°F)

Fig. V-8 - LOFT Thermal Flux Versus Distance from Core

Fig. V-9 - LOFT Gamma Heating Versus Distance from Core

SECTION VI

SITE DESCRIPTION

VI. SITE DESCRIPTION

A. General

The Test Area North (TAN) facilities are located within the National Reactor Test Station (NRTS) boundary and are approximately 27 miles north-northeast of the Central Facilities Area (See Figures VI-1 and -2). The Flight Engine Test (FET) facility is located approximately one mile west northwest of the Technical Support Facilities (TSF). The Loss of Fluid Test (LOFT) containment and services building adjoin the FET control and equipment building at the latter's south-east wall.

B. Environmental Conditions

1. Geography⁽¹⁾

The NRTS is located in southeastern Idaho on the Snake River Plain at the foot of the Lemhi, Lost River, and Beaverhead Centennial mountain ranges. From north to south the station extends 34 miles, and from east to west it extends 29 miles.

The general topography of the NRTS is essentially flat with a 1% slope from southwest to northeast at an average elevation of 5000 ft above sea level. Along the western boundary the mountains rise to an elevation of 10,000 ft. The elevation is lower along the Snake River to the east and southeast except for a gentle 600 ft rise midway between the site and the river. This rise runs parallel to the Snake River (See Figure VI-3).

From the valleys between the mountain ranges to the northwest of the area, the courses of the Big Lost River, Little Lost River, and Birch Creek run into playas or sinks, located in the north central section of the NRTS. The mountains encircle the Snake River Plain, and

(1) R. L. Nace, Morris Deutsch, P. T. Voegeli, "Geography, Geology, and Water Resources of the National Reactor Testing Station, Idaho", IDO-22033 - USGS, USAEC (1956).

rise as much as 10,000 to 11,000 ft above mean sea level. While the Centennial Mountains form an unbroken barrier at the northern end of the Snake River Plain, the ranges to the northwest and southeast, which form the side of the plain, are penetrated by deep valleys oriented northwest-southeast. (See Figure VI-4).

2. Meteorology^(1,2,3)

The meteorological conditions at NRTS are described in detail in previous reports. These reports are used as a basis for the following data:

a. Temperature

The maximum recorded temperature at the Weather Bureau Office (WBO) at the NRTS was 101°F in July, 1960. A minimum temperature of -40°F was recorded in January, 1962. For the TAN area, a maximum temperature of 103°F was recorded in July, 1960. A minimum temperature at TAN was recorded at -43°F in January, 1960.

b. Precipitation

Precipitation for the NRTS in general averages 7.69 in. and appears in the form of rain, snow, sleet, and hail. As to the rain-snow distribution, snow has been experienced in every month except July, and rain can be observed at any time during the year. The greatest 24-hour fall of rain was 1.78 in. in May, 1963, and that of snow, 8.5 in. in January, 1957. The annual amount of moisture at TAN is 7.35 in., the maximum rainfall recorded for any 24-hour period was 1.33 in. in July, 1963.

-
- (1) G. A. DeMarras, "The Engineering Climatology of the National Reactor Testing Station", IDO-12004, U.S. Weather Bureau (November 1958).
 - (2) G. A. DeMarras, Norman F. Isplitzer, "Diffusion Climatology of the National Reactor Testing Station", IDO-12015, U.S. Weather Bureau (April 1960).
 - (3) Private Communication with Weather Bureau Concerning Temperature Variation at NRTS.

c. Lapse and Inversion Conditions

Normal weather conditions at the NRTS develop lapse conditions during daylight hours with inversion conditions forming around sunset and continuing until after sunrise. During the day, especially clear summer days, thermal convection and the accompanying turbulence mix the surface layers of air with those above so as to bring both to a nearly common speed. If the surface winds maintain a speed greater than 15 mph through the night they will frequently prevent the formation of an inversion.

From Figure 30 of IDO-12015⁽¹⁾, inversions may be expected on 92% to 98% of the nights and an inversion of at least 10-hours duration may be expected on more than 61% of the nights of the year. Inversion conditions occurring during the winter months have a 50% probability of lasting for at least 15 hours, however, when longer durations are considered, approximately 20 hours, the probability decreases to 6%.

d. Winds

The NRTS is in the belt of prevailing westerly winds which are channeled upon entering the valley. A southwest wind predominates at the south end of the site while south-southwest winds occur most frequently at the north end (See Figure VI-5). The strongest channeled winds at the north end of the site generally come from the northwest out of the Birch Creek Valley. On occasion, these winds have been observed for more than 60 consecutive hours.

The winds at NRTS show a seasonal variation with the principal contrast being between winter and summer (See Figures VI-6 through VI-11). Particularly noticeable in the winter is the absence of the southwest wind at the north end of the site in the vicinity of TAN. The prevailing wind during this time is from the northeast.

(1) G. A. DeMarras, Norman F. Islitzer, "Diffusion Climatology of the National Reactor Testing Station", IDO-12015, U.S. Weather Bureau (April 1960).

3. Geology⁽¹⁾

The NRTS is at the central northern edge of the semi-arid Snake River Plain in southern Idaho. This plain extends in a great arc about 350 miles across Idaho, from the Oregon boundary west of Boise to near Ashton in eastern Idaho. The surface of much of the plain is covered by waterborne and windborne topsoil, under which there is a considerable depth of gravel ranging in size from fine sand to 3 in. aggregate.

The NRTS has no well-defined, integrated, surface-water drainage system, and it is not crossed by perennial streams. However, the NRTS overlays a natural underground reservoir of water, having an estimated lateral flow of not less than 500 cu ft/sec, or about 323 million gal/day. The main sources of water for this reservoir are the streams that originate in the mountains to the north and disappear into the porous soils of the NRTS area. These sources of water include the Big Lost River, Little Lost River, Birch Creek, and also an additional source from Mud Lake Basin.

The altitude of the water table ranges from 4,580 ft above sea level in the northern part of the station to about 4,400 ft near the southwest corner. The water table in the TAN area is at a depth of 200 ft and is seemingly very flat and at places may slope less than one foot per mile. Immediately beneath the central TAN area, the general direction of underflow appears to be south and southwest.

4. Seismology⁽²⁾

The NRTS site is located in a region which "The Pacific Coast Uniform Building Code", (1949), designated as a zone 2 area. Although many recorded earthquakes have been felt in Idaho, none were of sufficient intensity to cause more than minor damage to buildings. Of the 14 recorded earthquakes with an epicenter within the state, seven had their epicenter

(1) N. L. Nace, J. W. Stewart, W. C. Walton, "Geography, Geology, and Water Resources of the NRTS, Idaho", IDO-22034, USGS, USAEC (1959).

(2) "Seismic Probability Map of the United States", United States Coast and Geodetic Survey (1949).

within 100 miles of the TAN site, four to the southeast and three to the west. One of these, of unrecorded intensity, was at Arco, Idaho. In spite of the fact that a zone 3 area exists both north and south of the Arco area, the distances are so great that a zone 2 designation has been considered safe.

Although the lava plain of the Snake River is geologically young, the surrounding mountains are mostly of great age. Some recent geological faults appear to cross the plain beneath the lava beds, although their traces are not evident on the surface. None of these show indication of recent historical movement outside the lava plain. It may be expected that earthquake shocks will continue to be felt in the site area, but a prediction as to their intensity cannot be made with assurance.

C. Population Distribution⁽¹⁾

1. On-Site

The working force at each of the NRTS facilities as shown in Table VI-A is variable, depending on the construction work in progress. The closest major facility to TAN is NRF which is 17 miles south-southwest and has a daytime population of 1533 with 200 nighttime personnel. The daytime population of TAN is approximately 365 with 50 nighttime personnel.

2. Off-Site⁽²⁾

Population of outlying communities are shown in Table VI-B. Towns not appearing in the table are unincorporated and no census figures are available.

(1) "Population Distribution of NRTS", 1964 Census, USAEC Site Survey Office, Central Facilities, NRTS, Idaho.

(2) "Off Site Population Distribution", Life Atlas, 1960 Census Survey Rand and McNally (1961).

TABLE VI-A
POPULATION DISTRIBUTION OF NRTS

Location	Day	Night	Distance from TAN in Miles	Direction from TAN
TAN Area	365	50		
NRF	1533	200	17	SSW
TREAT & EBR-II	304	5	18	S
TRA	922	150	22	SSW
CPP	315	20	22	SSW
SPERT	114	5	22	SSW
Army Reactor Area	87	5	23	SSW
EBR-I & BORAX V	75	2	27	SSW
Central Facilities	946	100	25	SSW
OMRE-EOCR	<u>0</u>	<u>0</u>	25	SSW
Total	4661	537		

TABLE VI-B

OFF-SITE POPULATION DISTRIBUTION

Location	Distance from TAN (miles)	Population	Direction from TAN
Mud Lake	10-15	187	East
Monteview		*	NE
Bernice			West
Terreton		*	East
Howe	15-20	25	WSW
Hamer	20-30	144	East
Winsper			NNE
Roberts		422	ESE
Small		10	NNE
Dubois	30-40	447	NE
Menan		496	ESE
Lewisville		385	ESE
Idaho Falls		33,161	SE
Arco		1,562	WSW
Moore		385	WSW
Darlington		10	West
Atomic City		141	South
Humphery	40-50	25	NNE
Spencer		100	NE
St. Anthony		2,700	ENE
Parker		284	ENE
Plano			East
Salem			East
Sugar City		584	East
Rexburg		4,767	East
Lorenzo		100	ESE
Sunny Dell			ESE
Rigby		2,281	ESE
Ucon		532	ESE
Iona		702	SE
Ammon		1,882	SE
Shelley		612	SE
Basalt		275	SSE
Firth		322	SSE
Goshen			SSE
Blackfoot		7,378	SSE
Moreland		320	SSE
Riverside			SSE
Rockford			SSE
Martin			WSW
Grouse		58	West
Mackay		560	West

*The rural population in the area surrounding Mud Lake, Monteview, and Terreton is approximately 1000.

D. Land Usage

The NRTS encompasses an area of approximately 572,000 acres of which only 443 acres are utilized for the operational areas. The land is semi-arid, covered with desert shrubs, scattered grass, and frequent lava rock outcroppings (See Figure VI-12). Controlled grazing is permitted within the station's boundary, however, due to the limited vegetation coverage and seasonal conditions, grazing is normally restricted to the spring and fall seasons. In 1962, approximately 6500 sheep and 285 cattle permits were issued over the five month grazing season⁽¹⁾.

In the area near the northeast boundary of the station, there are about 600 dairy cows on 91 farms. Approximately 40 of the farms use their milk for private consumption, while the rest sell their milk primarily for cheese production. About 40 of the farms have pasture land which is used for cattle grazing from April to October. For the remainder of the year, the cattle are normally on feed lots⁽¹⁾.

Irrigated farm land is found in the valleys and along the perimeter of the Snake River Plain, with most of the water for irrigation coming from the Snake River and smaller streams. The irrigated farm land in the vicinity of the northern end of the NRTS as well as the general pattern of irrigated land in Southeastern Idaho is shown in Figure VI-13.

E. Radiological Surveillance

Continuous radiological surveillance of air, water, and food is conducted at the NRTS and surrounding communities.

The locations of the fixed stations around the NRTS, where routine samples of air, water, and milk are collected, are shown in Figure VI-14. Off-site underground water samples are collected at regular intervals from an area southwest of the site since this is the most prevalent direction of underground water flow. Two surface water sampling locations at the Snake River near Idaho Falls, Idaho and Twin Falls, Idaho, are used for comparison purposes.

(1) J. R. Horan, G. W. Wehmann, and B. L. Schmalz, "Experience in Site Selection at the National Reactor Testing Station, USA". IDO-12023 (1963).

On-site samples are taken from production wells in and near the plant sites in order to monitor water used for NRTS personnel consumption.

Fig. VI-1 - National Reactor Testing Station

Fig. VI-2 - Test Area North, NRTS

Fig. VI-3 - Topographical Cross Section Through NRTS

Fig. VI-5 - Population Distribution and Wind Frequencies About the Test Area North, NRTS

WINTER

SPRING

SUMMER

FALL

TOTAL

Fig. VI-6 - TAN 20 Foot Level Lapse Wind Roses

Fig. VI-7 - TAN 20 Foot Level Inversion Wind Roses

Fig. VI-8 - TAN 20 Foot Level Wind Roses

Fig. VI-9 - TAN 150 Foot Level Lapse Wind Roses

Fig. VI-10 - TAN 150 Foot Level Inversion Wind Roses

Fig. VI-11 - TAN 150 Foot Level Wind Roses

Fig. VI-12 - NRTS Vegetation

Fig. VI-13 - Irrigation in Southeastern Idaho

NATIONAL REACTOR TESTING STATION
ENVIRONMENTAL MONITORING PROGRAM
1963

Fig. VI-14 - NRTS Environmental Monitoring

SECTION VII

FACILITY DESCRIPTION

VII. FACILITY DESCRIPTION

A. General

The Flight Engine Test (FET) facility located approximately one mile west-northwest of the Technical Services Facilities (TSF) (see Figure VI-2) was selected by the Commission as the area in which to conduct the Loss of Fluid Test (LOFT) program. The LOFT containment building is integrated into the FET complex to utilize buildings, services and utilities previously constructed for the Aircraft Nuclear Propulsion (ANP) program.

The operational philosophy used in the design and construction of the original FET was based on remote testing of mobile, relatively unshielded, nuclear systems capable of releasing large quantities of fission products to the atmosphere. The LOFT design requirements to assemble and disassemble the experiment at facilities adequately equipped to provide these services and the requirement to control and operate the experiment from shielded facilities adjacent to the containment building have made the FET ideally suited for siting of the LOFT facility. Similar operating philosophies between the former GE-ANP programs and the STEP programs will permit maximum utilization of the services and equipment available in the FET.

B. Buildings and Facilities

1. Containment Building

The LOFT experiment will be conducted in a containment building which is designed to house the experiment and to contain the radioactive materials released as a result of the experiment. The containment shell is a steel cylinder with a hemispherical top head and ellipsoidal bottom head. The shell is 67 ft in diameter and 122 ft 3 in. high, with 95 ft 6 in. above ground (see Figures VII-1, -2, and -3). The portion of the shell below ground level is isolated from the main test chamber by a horizontal steel membrane. An 8 ft 6 in. diameter sump, 17 ft 6 in. deep is located in the center of the test chamber floor. The horizontal membrane is continuous around this sump which opens into the main test

chamber. The ground floor is designed for 2000 lb/ft^2 in addition to the pressure load.

The basement area is divided into rooms around the center sump (see Figure VII-4). These rooms are the decontamination sump, the decontamination equipment room, and the equipment room. Adequate concrete shielding is provided around the central sump, the decontamination sump and equipment and above and below the horizontal membrane to permit access to the equipment room at all times.

A six foot square hatch penetrates the floor of the test chamber into the equipment room below to facilitate the installation and maintenance of bulky equipment. Similarly a three foot square hatch gives access to the decontamination sump. Both hatches are closed with flush stepped plugs and are sealed to preserve the integrity and leak-tightness of the test chamber. An airlock is provided for personnel entrance into the basement areas.

The containment vessel volume and pressure design must simultaneously accommodate the loss of coolant experiment and the contents of the secondary coolant system. This insures that the containment design pressure is sufficient to withstand the pressure associated with the coolant blowdown which simultaneously ruptures the steam generator tube sheet.

In order to provide leakage to the atmosphere commensurate with postulated leakages from typical reactor containment vessels, the vessel design includes means for variable controlled venting to the atmosphere from 0% to a maximum of 5% of the free volume per day under all pressure conditions. The maximum allowable uncontrolled leakage from the building is 0.2% of the vessel free volume per day.

The containment shell is designed to maintain its leak tightness when subjected to 40 psig internal pressure and 260°F internal ambient temperature. The containment shell is also designed for an external pressure of 0.75 psi. A relief valve is provided to prevent excessive external pressures which may occur due to the sudden condensation of steam inside the containment shell. The walls of the containment shell are fabricated of SA-212-Grade B steel plate of firebox quality. The

portion of the shell below ground is embedded in concrete. The portion of the containment shell enclosing the test chamber is protected from missile damage by a concrete lining suspended from the shell. This missile shield is 2 ft thick on the floor, 1 ft thick on the cylinder, and 6 in. thick on the dome.

a. Access Openings

Each access opening into the containment building is designed to maintain the leak tightness and structural integrity of the shell.

(1) Railroad Access Door (Figure VII-5). The largest opening is that provided for passage of the railroad dolly and reactor system. This opening, which is 22 ft wide by 33 ft high, is covered by a specially constructed door which also serves as a continuation of the exterior biological shield. The door is a 5 ft thick steel structure filled with reinforced concrete.

The opening is sealed by two concentric inflatable seals, (see Figure VII-6) either of which is capable of maintaining the building leak tightness. Inflatable seals are used to assure that leak tightness is maintained even if small distortions occur in the door due to changes in temperature and pressure. This type of seal also permits movement of the door parallel to the seals without rubbing when the seals are deflated. The sealing effect can be verified at any time by monitoring the void between the two seals. Dependability is further insured by inflating each seal from an independent air source.

The door is mounted on four, four-wheel trucks which operate on two parallel rails installed in the bottom of a 6 ft deep trench. Special resilient supports are provided between the door and the trucks to permit lateral movement of the door during sealing. As the door is supported on trucks, the containment shell is not required to support the door weight. The containment shell does, however, support the pressure loading on the door and is reinforced around the door frame for this loading.

The door opens and closes at speeds not greater than 5 ft/min. It is pulled open and closed by means of a wire rope winch which is powered

by a variable speed drive. Shock absorbers stop the door at the end of travel in either direction. The drive equipment is located in a protected enclosure below ground. A protective enclosure is also provided to house the door in the open position for operational maintenance.

The dolly track system entering the containment building is made continuous across the trench by a bridge which is towed by the railroad access door as it opens. Four parallel railroad rails mounted on the bridge are spaced to match the track system on either side of the trench when the door is fully open and the bridge is in position. The bridge is accurately positioned by two locking pins that are operated by air cylinders located in the paving outside the containment shell where they are accessible at all times.

(2) Personnel Airlock. A personnel airlock into the test chamber is provided at ground level (see Figure VII-7). This airlock opens into the personnel shielding room which is accessible from the services building. The airlock is standard construction, designed for the 40 psig internal pressure. It is equipped with local controls as well as with interlocks and signal lights at the main control room.

(3) Basement Airlock. The basement airlock provides access to the equipment room in the basement from the remote control room tower (see Figure VII-4). The airlock is capable of containing the internal pressure in the event a rupture should occur in the membrane which isolates the basement area. This door is equipped with the necessary interlocks and signals at the main control room.

b. Penetrations

In addition to the access openings, the containment shell is penetrated by the viewing window, heating and ventilating ducts, secondary cooling system piping, service piping, electrical leads and instrumentation. Each penetration is designed to maintain the leak-tightness of the building.

The viewing window is located in the containment shell between the test chamber and the remote control room and provides visual verification of equipment operation. The viewing window is equipped with double seals

and provisions are made to verify the leak tightness by monitoring the void between the seals. The viewing window is protected from missile damage and from excessive pressures and temperatures by a sealed, steel door which closes over the entire window during blowdown. The protective door is opened and closed remotely from the remote control room.

Three 42 in. heating and ventilating ducts penetrate the containment shell into the test chamber; two additional ducts enter the basement area. Each duct is equipped with two remotely-operated butterfly valves in series which are supplied by separate electric circuits and are designed to close on signal or power failure. These ducts penetrate the containment shell through sleeves which are designed for the necessary flexibility and strength.

The containment shell also is penetrated by a multiplicity of piping, instrumentation, and electrical leads. Each penetration is designed to maintain the integrity and leak-tightness of the containment shell. Isolation valves are provided in the piping to assure that the contamination is confined to the test chamber.

c. Shielding

(1) General. Shielding for the LOFT facility is provided for three different radiation source conditions. The first condition is normal reactor operation. The water-filled reactor shield tank surrounding the reactor pressure vessel reduces the thermal neutron leakage to a level sufficiently low so that neutron activation of components in the containment shell does not limit access to the containment shell following reactor shutdown. In addition, the concrete shield surrounding the containment shell reduces the radiation level outside of the shell so that limited access to the radiation monitoring grid is permissible during reactor operation. All areas which require continuous access during reactor operation either are located underground or are shielded locally.

The second source condition is that following reactor shutdown. The shutdown shielding is provided by the water-filled reactor shield tank. The tank, which also contains lead, attenuates core and activation gammas so that personnel can enter the containment shell, if necessary, shortly after the reactor has been shutdown but before the nuclear blowdown.

The third source condition is that following nuclear blowdown and release of fission products to the containment shell. The required reduction in radiation level is accomplished by providing local shielding. For most of the areas requiring access both during reactor operation and following nuclear blowdown, the shielding required by the nuclear blowdown condition is more than that required by the reactor operating condition.

(2) Calculational Methods. The major portion of the shielding calculations have been performed on the IBM-7090 using the Grace I and Grace II programs (ref. 1 and 2). The programs determine the radiation level and nuclear heating values using a multi-group, multi-region calculation for slab, cylindrical or spherical geometry. The source distribution may be uniform, exponential or have a polynomial variation. The radiation levels exterior to the containment shell due to gamma ray scattering (or skyshine) were determined by use of the G-33 program developed by Los Alamos Scientific Laboratory for the IBM-7090. This program assumes point source geometry and includes exponential attenuation through zones to the scatter point. Exponential attenuation from the scatter point to the receiver point is also included plus a buildup factor to account for secondary scattering.

(3) Shielding for Reactor Operating and Shutdown Conditions

(a) Reactor Shield Tank. The reactor pressure vessel is surrounded by a reactor shield tank 4 ft thick, which contains lead and water. The 3.5 in. thick lead shield located adjacent to the reactor pressure vessel is contained in aluminum to preclude contact with the water in the reactor shield tank. The shield tank extends sufficiently far above and below the reactor core so that the radiation levels established as design criteria are not exceeded.

-
- (1) Duncan, D. S., Spier, A. B., "GRACE I - An IBM 704-709 Program Designed for Computing Gamma-Ray Attenuation and Heating in Reactor Shields", NAA-SR-3719, Atomics International Division of North American Aviation, Inc., Canoga Park, California, June 30, 1959.
 - (2) Duncan, D. S., Spier, A. B., "GRACE II - An IBM 709 Program for Computing Gamma-Ray Attenuation and Heating in Cylindrical and Spherical Geometries", NAA-SR-4669, Atomics International Division of North American Aviation, Inc., Canoga Park, California, November 12, 1959.

Design criteria for the shield tank are established for both operating and shutdown conditions. During operation at 50 Mw(t), the reactor shield tank reduces the thermal neutron flux at the outside surface of the tank to $10^5 \text{ n/(cm}^2\text{)(sec)}$. This flux is sufficiently low so that activation of components in the containment shell is no problem. After the reactor has operated for 1600 effective full power hours (efph) at a power level of 50 Mw(t), and has been shut down for 15 minutes, the lead and water in the shield tank reduce the radiation level from fission product decay gammas, the activated thermal shields, and from the pressure vessel to 200 mrem/hr. This level is sufficiently low to allow personnel to approach the shield tank and effect minor repairs to instruments, etc. in the event of component failure before reactor blowdown.

(b) Shielding for Areas Outside the Containment Shell. The one-foot thick concrete missile shield on the inside of the containment shell and the two feet of concrete on the outside of the containment shell provide shielding for areas outside the containment shell. The one-foot missile shield extends to the spring line of the containment shell 62 ft above the floor level. The two-foot thick concrete shield extends approximately 40 ft above the floor level and is only placed at locations around the containment shell which do not require access during reactor operation or following nuclear blowdown. More shielding is provided for areas which require access. This thickness reduces the radiation level at one meter above grade and 100 m from the center of the containment shell to 20 mrem/hr. This level allows personnel to service the radiation and fallout monitoring instrument grid during reactor operation. The radiation includes both direct radiation through the shield and skyshine. The skyshine originates from the gamma rays which penetrate the 6 in. thick concrete missile shield located on the interior surface of the containment shell dome.

(4) Shielding Following Nuclear Blowdown. Shielding for the blowdown condition is provided by a combination of concrete and earth fill. The source of radiation is assumed to be mixed fission products equivalent to 15% of the total fission product inventory following 1600 efph operation at 50 Mw(t). Assumed distribution of these fission products varies depending upon the location that is being shielded.

(a) Services Building. When determining the shielding required for the buried services building to meet the design criterion of 2.5 mrem/hr 100 sec after reactor blowdown, it was assumed that the fission products were all in the dome of the containment shell. The mass thickness of concrete and earth fill required over the building is 560 lbs/ft². To reduce the radiation level during operation to less than 1.0 mrem/hr during reactor operation and 100 sec after the reactor blowdown, a concrete and fill mass thickness of 600 lb/ft² is provided.

(b) Remote Control. The concrete thickness between the containment building interior and the remote control is 4-3/4 ft. This reduces the radiation dose in this area to 100 mrem/week when 15% of the core fission product inventory is released into the containment shell following a nuclear blowdown. The weekly dose is based upon an eight-hour day, five-day week with the first eight-hour period starting 100 sec after reactor shutdown. Two source conditions were examined for this case: (1) the fission products were distributed uniformly throughout the volume of the containment shell, and (2) the fission products were distributed uniformly on the surface of the containment shell. It was found that the volume distributed source was controlling and that the 4-3/4 ft of concrete would reduce the radiation level in the rooms to less than 10 mrem/hr during reactor operation.

The viewing window mounted in the concrete wall has a mass thickness equivalent to 4-3/4 ft of concrete. The fast neutron flux incident upon the radioactive side of the window during reactor operation is reduced to approximately 0.1 mrem/hr by the window glass; consequently it is not necessary to add hydrogenous material for neutron moderation.

(c) Covered Roadway. The length of the covered roadway used for personnel access is based on the maximum direct gamma dose rate from the containment shell following nuclear blowdown. The dose rate at the tunnel exit is such that personnel walking (3 mph) from the tunnel to an area outside the high radiation field will not receive a total dose in excess of 5 rem whole body. The existing tunnel exit is at a distance of 710 ft from the centerline of the containment shell which adequately satisfies this criterion. Earth fill over the tunnel varies from 6 ft immediately adjacent to the containment shell to 1.5 ft at the far

end of the tunnel. This is sufficient to reduce the radiation level inside the tunnel to 7.5 mrem/hr 30 minutes after the nuclear blowdown.

(d) Containment Building Basement Area. The allowable radiation level in the basement area is 100 mrem/hr 30 minutes after the nuclear blowdown, and 10 mrem/hr during reactor operation. The six-foot thick concrete floor required for structural purposes reduces the radiation level to less than the allowable values for the operating and post-blowdown conditions.

(e) Containment Penetrations. Radiation streaming through penetrations provided for electrical and instrumentation conduits and coolant piping is reduced to an acceptable value by providing three 90-deg bends before exiting to areas which require occupancy during reactor operation or following nuclear blowdown. These penetrations are located below the remote control room. Heating and ventilating ducts enter and leave the containment shell through shielded boxes to reduce the radiation level outside of the shell to allowable values.

(5) Miscellaneous Shielding

(a) Condensate Sump. The shielding around the 5000-gallon condensate sump, located directly below the reactor pressure vessel, is 4.5 ft of ordinary concrete. This reduces the radiation level in the basement area to 100 mrem/hr 30 minutes after the nuclear blowdown. The source of radiation is assumed to be 15% of the core fission product inventory distributed uniformly throughout the volume of the condensate sump.

(b) Decontamination Sump. The decontamination sump is shielded by four feet of concrete. This reduces the radiation level in the basement to 10 mrem/hr one week after nuclear blowdown. It is assumed that 15% of the fission products are distributed throughout the volume of the decontamination sump.

(c) Hot Waste Tank. The two 25,000 gallon hot waste storage tanks are buried and shielded by seven feet of fill. This reduces the radiation level at grade to less than 10 mrem/hr one week

after nuclear blowdown. Source of radiation is 15% of the fission products distributed in the volume of one hot waste storage tank.

(d) Primary Coolant Purification System. The primary coolant demineralizer and spent resin cask are shielded by five inches of lead to reduce the radiation level three feet from the container surface to 200 mrem/hr. Source of radiation is assumed to be 1% of the fission products in one fuel pin which have decayed for 15 minutes.

(e) Halogen and Particulate Removal System. The demisters, roughing, absolute and carbon filters are located in a pit immediately adjacent to the containment shell. A concrete shield 4-1/2 ft thick reduces the radiation level over the filters to 20 mrem/hr one day after the nuclear blowdown. The exhaust fans are located in a subgrade concrete room which reduces the radiation level to 20 mrem/hr following the nuclear blowdown.

d. Building Equipment

Several pieces of equipment are installed inside the containment shell area which could affect the safe operation of the test. These include the overhead crane, TV viewing system, railroad coupler and bumper, coupling station and the blowdown jet deflectors.

(1) Overhead Crane. A 30/5-ton circular overhead crane is provided inside the test chamber. Signals and power supply to the crane are collected at a slip-ring unit which is mounted on a structure above the crane bridge. This eliminates flexing of cables or runway collectors and permits unlimited rotation of the crane bridge. Controls, wiring, motors and slip-ring units are protected from missiles and are designed for the conditions which occur inside the containment shell during blowdown including the pressures and temperatures resulting from an MCA, the steam atmosphere and the use of depressurizing sprays and decontamination solutions. It is controlled remotely from the remote control room.

(2) Railroad Coupler and Bumper. A railroad coupler and two shock absorbers are located on a concrete structure inside the

test chamber. The shock absorbers are designed to absorb the energy of the railroad dolly and locomotive at one mile per hour. The stopping of the dolly is accomplished within a distance of 12 inches. The shock absorbers are designed to travel an additional 8 in. to permit the locomotive to push the dolly into the coupler mounted on the same concrete structure.

The coupler is a standard type F railroad coupler modified to permit mounting on a vertical surface without the usual draft gear. An air cylinder is mounted above the coupler to provide remote uncoupling. Control of the air cylinder is from the remote control room.

The concrete structure is designed to hold the dolly in place during an accidental rupture in the primary coolant system.

(3) Coupling Station. A station is provided inside the test chamber for the remote uncoupling of all piping, instrumentation and electrical leads from the dolly. All low-pressure pipes are equipped with quick disconnects which are gang-operated by air cylinders. Each of these lines is equipped with a second quick-disconnect which is operated by a line attached to the dolly. In the event the remotely-operated air cylinders fail to uncouple the lines, the backup disconnects are uncoupled as the dolly is backed away from the coupling station. The high pressure secondary coolant lines are equipped with motor-operated disconnects. The electrical and instrumentation leads are equipped with spring loaded connectors released by lanyards attached to the dolly.

Operation of the disconnects is controlled from the remote control room.

(4) Blowdown Jet Deflectors. Two deflectors are provided to prevent damage to the containment shell walls by the force of the water-steam jets during blowdown. These deflectors, designed to break up and scatter the energy of the jets, are mounted on each side of the dolly, directly in line with the blowdown nozzles.

(5) TV Viewing Facilities. TV cameras are provided in the containment building to facilitate operations inside the test chamber after blowdown. They are located to permit remote viewing of all critical

mechanisms and areas. The controls and TV screens are located in the main underground shielded control room.

(6) Fuel Storage Cells. Ten fuel storage cells are located in the floor of the test chamber (see Figure VII-7). These are equipped with water supply and drain lines to permit storing the spent fuel under water. Each storage cell is closed with a shield plug and is sealed to prevent gas leakage. The cells are positioned 5 ft from the wall where they are accessible to the overhead crane.

2. Services Building

The service building is a structure adjacent to the LOFT containment building and joining it to the FET control and equipment building (see Figures VII-8 and -9). The service building is a one level structure constructed below ground at the level of the containment building basement. The services building houses the decontamination equipment and utility room, the electrical equipment room, the remote control room structure, the sample room, and the shielded personnel entrances.

a. The Remote Control Room Structure

The remote control room structure is adjacent to the north side of the containment building. It is constructed with four levels, three of which are above ground level. This concrete structure contains the shielded labyrinths through which the piping, instrumentation and electrical leads are routed after leaving the containment shell. It also contains the remote control room which is located on the fourth level. A stairway provides access to all levels. Entrances located on the basement floor provide access from the decontamination equipment and utility room into the basement of the containment building.

Secondary coolant and service piping from the containment shell enter the west labyrinth below the 4808 ft level. Power leads leaving the containment shell enter the east labyrinth below the 4808 ft level (see Figure VII-10). The power leads and piping are then routed through a shielded passage to the cable tray tunnel and then to the cable room

below the control room in the control and equipment building. Piping, electrical and instrumentation to the remote control room are routed through the stairwell.

The remote control room is located on the fourth level of the structure. The controls and instrumentation necessary for the remote operation of the equipment in the containment building are located in this room. The equipment to be operated from this room includes the overhead circular crane, the mobot, the railroad dolly coupler and jacks, the piping and electrical disconnects at the coupling station, the railroad access door and bridge, the protective closure for the viewing window, the pressure reduction spray system, the decontamination spray system and the containment building lighting.

A window is provided between the remote control room and the test chamber to facilitate the remote operation of equipment. This window is designed to permit viewing of the coupler system, the spray system, sample collecting stations, circular crane and dolly-mounted equipment. A protective pressure-tight steel closure is provided on the test chamber side of the viewing window. This closure protects the window from damage by missiles as well as excessive temperatures and pressures. An air flow is provided to the test chamber side of the window to prevent condensate from collecting on the glass. The window is designed to maintain the necessary biological shielding and the leak tightness of the containment shell. Double seals are provided with taps for monitoring the voids between seals.

b. Personnel Entrance

A shielded personnel entrance is located on the south side of the service building above ground level. This entrance from the parking area opens into a shielded corridor which provides access to the control and equipment building, the test chamber in the containment building, and to the service building below. The personnel entrance is equipped with shield doors which open to the outside and into the control and equipment building. A labyrinth is provided outside the airlock into the test chamber to protect personnel in the corridor from air activity inside the containment building.

c. Decontamination Equipment and Utility Room

The decontamination equipment and utility room is in the service building. This room houses the heating and ventilating units and air intakes, the decontamination equipment, the secondary coolant pumps and the makeup equipment for both the primary and secondary coolant systems. The heat exchanger and condensate receiver for the secondary coolant system are located on the roof of the service building.

d. Electrical Equipment Room

A 36 ft x 36 ft area at the west end of the service building is walled off for the electrical equipment room. This room houses LOFT electrical equipment not provided in the control and equipment building. This includes the 480 and 2400 volt switchgear and the 480 volt motor control centers.

e. Sample Room

The equipment for the gas and plateout sampling system is in the service building under the shielded corridor. The sample containers are transferred from the test chamber to this room through an airlock. Provisions are made in the airlock to depressurize the plateout sample containers and to return the released radioactive gases to the test chamber. The containers are loaded directly into shielded casks as they are taken from the airlock.

The room is surrounded by concrete shield walls to isolate it from the rest of the service building.

Sample removal equipment is located in the sample room. Interlocks and signal lights are provided in the control room for supervisory control.

An earth cover is provided over the service building to shield personnel. The earth shield also covers the shielded corridor and airlock. Shielding of the remote control room is provided by the concrete walls of the structure. Air intake and exhaust openings into the building are equipped with protective closures which seal off the opening if outside activity is detected.

3. Support Facilities

a. Examination Area

(1) General. TAN Buildings 607, 657, and 633 are designated as the examination area and comprise an integrated facility for performance of all phases of assembly, disassembly and post-operative analytical examination of nuclear systems and subsystems following sustained power operation.

The examination area consists primarily of the hot shop, radioactive materials laboratory (RML), hot cells, storage pool area, assembly area, warm shop, and machine shop. Other support facilities housed within the buildings forming the examination area include the decontamination room, sand blast room, chemical cleaning room, storage area, components test laboratory, instrument shop, inspection and control area, and miscellaneous auxiliary facilities (locker room, offices, shower area, etc.).

(2) Hot Shop. The hot shop is an area 160 ft long by 51 ft wide by 67 ft 6 in. high. It is surrounded by reinforced concrete shielding walls to provide a maximum dose rate of 6.25 mrem/hr in the operating galleries located on both sides of the hot shop while containing an experimental assembly measuring 5.7×10^6 R/hr gamma. Located in both the upper and lower operating galleries are windows which provide visual observation of the reactor assemblies during the remote handling operations.

The east end of the hot shop is partitioned to form the special equipment services cubicle. This cubicle is used for repairing the overhead manipulator and overhead crane when the hot shop contains radioactive material.

The west wall contains the locomotive and rolling stock entrance door. It is a sliding, biparting, concrete door with a staggered joint. The door is operated by electric motors located outside the hot shop and controlled from the control console.

A personnel labyrinth is provided in the southwest corner for access to the hot shop. A monitoring and change room is located just outside this labyrinth.

Two turntables which are flush with the floor have been provided to rotate radioactive devices so that work may be viewed from the hot shop windows. Remote control allows rotation and automatic selective indexing of the turntables.

Each turntable has an overall diameter 17 ft 6 in. and is capable of supporting a 60-ton load. Two control stations are provided for each turntable and are located at windows adjacent to the turntable in the upper and lower operating galleries. Controls for directional rotation, speed and indexing are located at each control station.

The hot shop heavy crane is a 100-ton overhead, single dolly, double-hoist crane especially equipped and adapted for operation by remote control.

The control stations for the heavy crane are located at the hot shop windows. The stations are interlocked so that only one may have control of the crane at a given time and control cannot be "stolen" from the operating station by a standby station. Each station contains all control necessary for the remote operation of the crane.

The remote handling equipment in the hot cell is of the general purpose type. It was engineered and supplied by the General Electric Laboratory for the General Electric Company, Idaho Test Station. This equipment was designed to service a variety of "hot" mechanisms with great versatility, and to handle future mechanisms with a minimum of modification.

The hot shop contains two manipulators on each long wall and a heavy-duty overhead manipulator, all of which can be coordinated to work together. The overhead manipulator also serves as a crane follower and can be remotely controlled.

A pair of Argonne master-slave type manipulators are also installed in the hot shop. These manipulators are mounted in 10 in. tubes passing through the hot shop wall above one of the viewing windows.

A dispatcher's control console located in the south lower operating gallery provides a central unit for remote operation of the outdoor turntable, railroad signal, hot shop door, outdoor viewing, communications control for the locomotive, and control of emergency power.

(3) Special Services Equipment Area. The remote handling equipment in the hot shop has been so designed that maintenance of this equipment can be performed with a minimum of personnel contact. A special equipment service cubicle has been provided at the rear of the hot shop for maintenance of the overhead manipulator, overhead crane, and the General Mills manipulators. This service cubicle is provided with sliding shielding doors so that it may be radiologically isolated from the hot shop.

(4) Radioactive Materials Laboratory. The radioactive materials laboratory (RML) is located adjacent to the southeast corner of the hot shop. It is equipped and used for remote inspection, cutting, and other operations of a more delicate nature.

The RML periscope is provided for the close inspection of objects in the inspection cubicle. This area is serviced by four Model 8 master-slave manipulators, two Argonne #6 manipulators, and two General Mills-type manipulators.

(5) Service Facilities. The self-sustaining TSF area furnishes other services which are used in support of the examination area and the remote test facilities. These include a warehouse and receiving building (TAN-628), health physics building (TAN-606), a medical dispensary building (TAN-618), fire station building (TAN-603), and equipment repair shop (TAN-604). Other services which support the TSF buildings include the conventional utility systems such as steam, fuel, water, power, waste, etc.

(6) Storage Pool Area. A storage pool area located adjacent to the hot shop is used for transfer of material to and from the hot shop. The storage pool area has a standard 15-ton, single trolley, single hoist bridge crane for transfer of material within the storage pool.

(7) Hot Cells. The hot cell facility, consisting of four hot cells and miscellaneous work area, was designed and built to satisfy the need for space in which post-irradiation examination of reactor fuel and mechanical components could be performed. The cell

working space is divided into four equal areas, the type and degree of contamination that can be handled in each being slightly different.

b. Transportation System

(1) Trackage. Four-rail trackage connecting the original FET area and the hot shop is extended to the LOFT facility by adding a new turn out into the LOFT containment building. In adding the new section, all fill and subgrade were compacted to 95% maximum density at optimum moisture content, and the track was designed in accordance with the American Railway Engineering Associations Standards. The minimum radius of curvature of this trackage is 750 ft and the maximum grade is 1%.

(2) Turntable. The 90 ft diameter turntable located just west of the main TAN hot shop is used to switch the railroad dolly mounted nuclear systems from the examination area to the outlying test facilities. This turntable is equipped with television camera, indicators and controls for remote operation by a dispatcher stationed on the south gallery of the hot shop.

The turntable is a pit-mounted bridge type with full circular deck, approximately 90 ft in diameter with four rails spaced to provide three standard gauge tracks. The builtup plate main girder spans 42 ft from the spindle to the carriage rail.

Calculations indicated that the turntable was not capable of supporting a total load of 800 tons for the LOFT test components and the locomotive concurrently; therefore, the turntable was reinforced to handle the loads required in the LOFT tests by constructing an additional 42 ft diameter circumferential track complete with carriage assemblies (see Figure VII-11).

(3) Rolling Stock. The experimental equipment is moved from the test building to the TAN hot shop on a railroad dolly designed to utilize the existing four-rail, three-track standard gauge railroad presently installed at TAN.

The dolly was designed for movement by the shielded locomotive (Figure VII-12) located at the TAN complex. Locomotive specifications are as follows:

Weight

Total locomotive	430,000 lb
On drivers	224,800 lb
Per driving axle, front truck	57,400 lb
Per driving axle, rear truck	55,000 lb
Per idle axle, front truck	144,700 lb
Per idle axle, rear truck	90,500 lb

Ratings

Tractive effort at 30% adhesion	67,400 lb
Tractive effort at continuous rating of motors	22,700 lb
Maximum permissible speed	15 mph

Shieldings

Front:	36-1/2 in. water 9-1/2 in. lead
Rear:	22-1/2 in. water 2-1/8 in. lead
Sides:	26 in. water 3-1/8 in. lead
Bottom:	26 in. water 2-1/8 in. lead

Dimensions

Overall length (over coupling)	44 ft 7 in.
Height	19 ft 4 in.
Width	11 ft 10 in.

4. Experimental Facilities Systems

a. Secondary Coolant System

(1) System Description. The secondary coolant system, shown in Figures VII-13 and -14, is an intermediate heat transfer link between the primary coolant system and the atmosphere.

The system is a closed loop comprised of the shell-side of the steam generator, the air-cooled condensers, the condensate receiver, the secondary coolant pump and the auxiliary secondary pump (operating in parallel with the secondary coolant pump). High purity condensate is forced by the secondary coolant pump into the steam generator where it absorbs heat and evaporates. The steam then flows through air-cooled condensers where it gives up its latent heat to the air. The condensate is collected and recirculated.

The steam is condensed at approximately the same pressure at which it is generated. All components and piping are designed for 1100 psig.

Makeup to the system is supplied from the demineralized water system by means of a high pressure makeup pump. Corrosion in the system is minimized in a manner similar to that utilized for the primary coolant system; i.e., low conductivity, low oxygen content, high pH.

At part loads, steam is generated at higher temperatures (and pressures) in order to decrease the LMTD across the steam generator. In order to prevent subcooling of the condensate in the air-cooled condenser or excessive differential pressure at part loads, the following steps can be taken:

- (1) Valving off sections of the air-cooled condenser.
- (2) Controlling air flow by variable-pitch and/or variable-speed fans.
- (3) Recirculating air by means of shutters.

(2) Components

(a) Piping. The piping is carbon steel, ASTM A-106, Grade B, seamless, and butt-welded. It is designed for 1100 psig and 600°F. Exterior surfaces are insulated to reduce surface temperatures to 140°F maximum and are protected by a moisture-impervious jacket.

(b) Secondary Coolant Pump. The condensate pump is a hermetically sealed (canned) motor type, rated at 550 gpm at 100 to 150 ft TDH and is designed for 1100 psig suction pressure at 600°F. The material of construction is carbon steel.

(c) Air-Cooled Condenser. The condenser is a multi-cell forced draft unit with finned tubes designed to condense 214,000 lb/hr of saturated steam at 340 psia. It is equipped with variable shutters and variable (and fixed) pitch fans. Materials of construction are carbon steel except for the fins on the outside of the tubes which are aluminum or copper. Condenser design pressure and temperature are 1100 psig and 600°F respectively. The air-cooled condenser has approximately 100,000 ft² of extended finned surface area. Inlet air temperature is maintained at 100°F (by partial recirculation). The air-cooled condenser is permanently mounted outside of the containment building and provisions are made to keep the condenser from freezing in the winter during shutdown. Air vents are provided to remove noncondensables and drains to remove water.

(d) Auxiliary Air-Cooled Condenser. This unit is an integral part of the main condenser. Its fan, however, is connected to the emergency power supply and the unit is used whenever the required cooling capacity is less than 5 Mw thermal (which is the cooling capacity of the auxiliary air-cooled condenser).

(e) Condensate Receiver. The receiver is a carbon steel pressure vessel (ASME Code) having a capacity of 500 gallons and is designed for 1100 psig pressure at 600°F. An electrical immersion heater rated at 200 kw is provided to assist in startup of the secondary system.

(f) Makeup Pump. The demineralized water makeup pump is a positive-displacement type, having a rating of 5 gpm at 1100 psi head. The material of construction is carbon steel.

(3) Process Instrumentation and Control. Automatic and manual controls with indications of normal operating functions are located in the main control room. Off-normal operations approaching hazardous conditions are visually and audibly annunciated.

All controls and signal transmissions are electrical for items located inside the containment shell, with power supplied from normal 120 vac plant power with standby emergency power available. Pneumatic instrument systems are utilized in those areas of the plant exterior to the containment shell where response time is not a governing factor.

(a) Steam Generator

- (1) A valve located in the blowdown line to the quench tank is used to maintain the water level in the steam generator during startup periods when heat transfer from the primary coolant increases the volume of the water in the generator. The valve is remotely actuated manually by the operator in the main control room.
- (2) During normal operation after sufficient water has evaporated from the generator, a level controller in the generator operates a valve in the feedwater line and admits feedwater into the generator to maintain a constant water level.
- (3) Feedwater flow to the generator and steam in the main line leaving the generator are recorded in the main control room and abnormal conditions annunciated.

(b) Air-Cooled Condensers

- (1) The condensers are air-cooled and consist of five cells. One cell belonging to the auxiliary condenser is equipped with a variable pitch fan; the remaining four cells are fitted with three fixed blade fans and one variable pitch fan respectively. The variable pitch fans are controlled automatically. The selection of the number of fixed blade fans required to maintain adequate cooling is manually selected by the operator.

Steam passing through the condenser is condensed at basically the pressure at which it was generated, less any pressure drop inherent in the piping system.
- (2) A pressure controller in the main steam line leaving the generator develops a signal which is used to change the setting of the variable pitch fans to regulate the amount of air passing over the condenser. The setpoint of the pressure controller determines the reactor power level (when the latter is larger than 1 to 2% of full power) and is positioned manually by the operator in the control room.

The controller is interlocked electrically with the reactor safety system. In the event of a scram, or a normal plant power failure, the four-cell condenser is isolated from the steam loop and the controller output is switched to the auxiliary condenser, which is maintained on automatic control to handle decay heat removal.

- (3) Air inlet dampers on each cell of the condensers are positioned locally to compensate for seasonal temperature changes. A temperature controller located in the air inlet to each condenser cell regulates a small bypass damper which recirculates heated air back to the inlet to maintain 100°F inlet temperature to the condenser. Actuation of the bypass damper and the outlet damper for each section of the condenser are ratioed. Condensate temperature, fan bearing temperature, air inlet temperature and steam flow are indicated in the main control room.

(c) Condensate

- (1) A level controller on the condensate receiver provides an on-off signal over a narrow band to start the demineralized water makeup pump on low level in the receiver. The condensate level in the receiver is indicated and annunciated in the control room.
- (2) Recycled condensate returning to the steam generator is monitored for impurities. Condensate conductivity, per cent oxygen content and radioactivity are indicated in the main control room.

The secondary coolant pump, the auxiliary secondary coolant pump, the electric heater in the condensate receiver and the system air vents are actuated manually by the plant operator. The feedwater control valve is operated manually during startup operation only.

(4) System Operation

(a) Startup

- (1) The initial conditions for startup are: normal water level in steam generator and condensate receiver; steam flow valves closed; feedwater valve closed; condensers and steam piping full of air.
- (2) Primary coolant and the secondary coolant are brought up to temperature and pressure simultaneously by operation of the primary pumps, pressurizer heaters, and the condensate heaters.
- (3) The secondary coolant system (including the auxiliary condenser bank only) is brought up to temperature and pressure by energizing the heaters in the condensate receiver. Air is vented from steam piping and the auxiliary condenser. The steam flow valve in the auxiliary condenser line is open, the fans are not operating, and the louvers are closed. The auxiliary secondary coolant pump is turned on.

Since thermal expansion increases the volume of both primary and secondary coolant, water levels in pressurizer and steam generator are maintained by blowing down to the quench tank.

- (4) When temperatures have reached the desired level, the reactor may be brought to criticality. The temperature of the primary coolant system can be maintained at a constant value by adding heat through operation of the primary pumps and the pressurizer heaters, or by removing it through the steam generator.
- (5) When the primary and the secondary coolant systems are at the proper temperature, the reactor is brought manually up to 1 to 2% power. The pressure controller, which actuates the variable pitch fans is adjusted to maintain the desired condensate temperature. When sufficient water has evaporated from the steam generator to lower the liquid level, the feedwater control valve is manually opened to admit feedwater to the steam generator. When temperatures and pressures stabilize, at 1 to 2% flow, the main secondary coolant pump can be started.

- (6) The plant can gradually be brought to full load by adjusting the pressure controller and manually cutting in additional condenser units. The reactor is now on automatic control and the reactor power follows automatically, maintaining constant average primary coolant system temperature. It is necessary to vent and drain each condenser simultaneously as it is brought into service. The water level sensing device in the steam generator automatically controls feedwater flow to maintain the level in the steam generator. As the load is increased from 0 to full load, the secondary coolant pressure will drop; e.g., at 540°F primary coolant system temperature, the steam pressure will drop from approximately 1000 psia at zero load to 340 psia at full load.

(b) Steady-State Operation. The plant can be maintained at constant load by the pressure controller which maintains the appropriate condensate temperature by varying fan pitch. The inlet louvers of the condenser cells are manually adjusted for seasonal temperature variations. Outlet and recirculation louvers are adjusted automatically to recirculate the air and maintain its temperature at the fan intake at 100°F. Secondary coolant purity is maintained by blowdown from the steam generator and demineralized water makeup. Additives for corrosion control and oxygen removal are injected with the feedwater.

(c) Normal Shutdown. The steam pressure setting is increased which results in a reduced steam flow, a rise in the primary coolant temperature and, as a consequence, a reduction of the reactor power. ~~Maintaining the steam flow at a few per cent of full load, the~~ reactor is brought below criticality. The secondary coolant system can then be used to reduce the primary coolant system temperature at a controlled rate.

(d) Shutdown After Failure of Normal Power. Failure of commercial power result in automatic scram of the reactor. Decay heat rapidly drops to 2 to 3% of full load power. The heat is dissipated by generating steam in the steam generator and condensing the steam in the auxiliary condenser. The variable pitch fan of the

auxiliary condenser is connected to the emergency power supply. The liquid level in the steam generator is maintained by the auxiliary secondary coolant pump which is also connected to the emergency power supply.

b. Sampling Systems

The techniques of sampling and the number and sizes of samples, are based on the conceptual design. As a result, some modification may be required as experiments are completed.

(1) Plateout Sampling. The quantity and type of fission products deposited inside the primary coolant system and containment vessel are determined by periodic sampling and analyzing samples of various materials located throughout the system and building. These plateout samples are approximately $3/4$ in. in diameter by $1/2$ in. thick and are identical in composition to the surface area being monitored. For example, the samples used to measure the plateout on the concrete walls are made of concrete and coated with the same material used on the walls.

The samples for measuring plateout external to the primary coolant system are housed in $4-1/2$ in. diameter spherical aluminum bombs. Each sample bomb contains two samples and is constructed such that, when open, a single face of each sample is exposed to the containment atmosphere within six inches of the reference surface. The average plateout on the two samples is considered as the representative value. The total plateout divided by the combined exposed surface area will be considered as the average plateout per unit of surface area. The bombs, when closed, are water tight so that excessive moisture from decontamination solutions or primary coolant condensate will not affect the plateout on the samples.

To determine the time-space behavior of fission products such as iodine and the saturation characteristics of the exposed containment vessel and equipment surfaces, plateout samples are required at representative positions throughout the containment volume.

Some of the samples are removed periodically during the tests. These samples are identified as "removable samples". Other samples remain inside the containment vessel until the termination of the test. These are identified as "nonremovable samples".

(a) Removable Samples. Removable samples are located at 12 stations on the inner surface of the test area within the containment vessel. Each station contains a sufficient number of samples (approximately 24) to provide a complete coverage of the fission product buildup and decay for a period of approximately five days. One sample from each station is removed every one-half hour for the first four hours, then every hour for the next six hours or until the iodine concentration inside the building has reached a maximum. Thereafter samples are removed from each station every 24 hours until the termination of the test. The general locations of the sample stations inside the containment vessel are as follows:

- (1) One station at the center of each quadrant on the surface of the vessel wall, approximately 38 ft above the containment vessel area operating floor.
- (2) One station on the operating floor of the containment vessel in each quadrant approximately midway between the center and the periphery of the test area.
- (3) One station in each quadrant on the surface of the dome approximately midway between the center of the containment vessel and the vertical walls.

The containment vessel wall temperature is recorded each time a sample is removed.

(b) Nonremovable Samples. The nonremovable samples are positioned at the following general locations:

- (1) At four foot intervals along the vertical centerline in each quadrant extending from the containment vessel floor to the top of the dome. All samples are attached to the vessel walls.
- (2) At four foot intervals on the periphery of the containment vessel wall at approximately 38 ft above the floor.

- (3) At four foot intervals on the periphery of the dome at the same elevation as the removable samples.
- (4) At four foot intervals in each quadrant on the vessel floor extending from the wall to the center of the vessel.
- (5) On each major piece of machinery inside the containment vessel: reactor vessel (4 samples), pressurizer (1 sample), primary heat exchanger (1 sample), primary coolant piping (4 samples), top of railroad dolly (4 samples), underside of railroad dolly (4 samples), overhead crane bridge (4 samples), secondary system piping (4 samples), and other major pieces of equipment (~ 20 samples).
- (6) Inside the reactor pressure vessel. These samples are not housed in containers.

In addition to the above, representative samples of materials such as carbon steel, stainless steel, coated concrete, uncoated concrete, etc. are placed inside the containment vessel to investigate the plate-out characteristics of various materials. These samples are also contained in bombs as described above and are removed after decontamination of the containment vessel. The containers will be closed at the termination of the test.

A number of material samples are located inside the reactor pressure vessel and primary coolant piping. These samples are fixed and will not be removed until the experimental package is transferred to the examination area for post-operative analysis.

(2) Gas Sampling

(a) General. Gas sampling encompasses the collection of representative samples of the containment vessel atmosphere following core meltdown. These samples are used to determine the volatile and noble gas (Xe, Kr, I, Cs, Tc, Ba) concentration in the air. By obtaining samples as a function of time, information can be obtained pertaining to the time-space behavior of the isotopes and the plateout characteristics as a function of air concentration. In addition, these data assist in evaluating the total quantity of fission

products released from the fuel and the concentration available for leakage from the containment vessel as a function of time. Other samples are obtained at the time of maximum concentration of iodine inside the containment vessel. These data are used primarily to determine the plateout behavior as a function of concentration for the various materials inside the containment vessel.

The gas samples are obtained by an evacuated "gas bomb" technique. These evacuated bombs, or bottles, are installed prior to the test and remotely opened at the time a sample is desired. Each aluminum bomb measures 4-1/2 in. in diameter, and contains an activated charcoal cartridge at the inlet for iodine collection.

(b) Removable Samples. The removable gas samples are located at the 12 stations identified for the removable plateout samples, except that they are positioned approximately three feet from the vessel wall. Each station contains 24 sample containers and a gas sample is removed simultaneously with a plateout sample at the same station.

(c) Nonremovable Samples. Nonremovable gas samples are located approximately three feet from surfaces of representative equipment where plateout samples are being taken.

(d) Chemical State Iodine. The chemical state of iodine released to the containment atmosphere, as a result of a loss of coolant, is of paramount importance in the study of various iodine retention mechanisms and for the design of devices for iodine removal from the containment atmosphere.

Four iodine collection devices, capable of quantitatively distinguishing the two basic forms of iodine released, are used for atmospheric sampling of iodine during the test. The two basic forms of iodine are elemental iodine and iodine compounds (iodides and iodates). Each collection device is composed of several sampling trains, thus allowing several samples to be taken at various times during the duration of the tests at each of the four locations.

Solenoid valves at each sampling station are remotely operated from the remote control room. The four stations are located on the containment vessel operating floor, one in each quadrant.

The iodine sampling train consists of a carbon-tetra-chloride (CCl_4) trap for elemental iodine collection, sodium bisulfate (NaHSO_3) for the collection of iodine compounds, a charcoal filter for removal of the remainder of the iodines not being collected by the first two traps, and a gas sampling bomb for air sampling downstream of the charcoal trap. Each collection component in the train is removable for analysis at the conclusion of the test.

(3) Particle Sampling

(a) Particulate Concentration and Location. The particulate sampling device is designed to collect samples of the particulate activity in the atmosphere as a function of time after core melt-down. These data are necessary to evaluate the time-space behavior of the particulate activity and to determine the concentration of various isotopes in particulate form that are released from the core. In addition, these data determine the particulate activity that is available for leakage from the containment vessel.

It is desirable to collect samples as a function of time. However, it is not necessary to remove them from the vessel until the test has been terminated.

Each sampling device consists of 24 collecting tubes connected to a central manifold which is attached to a suction pump. The device has the capability of drawing a sample of the containment atmosphere through a single tube, one inch in diameter containing a filter cartridge, at a rate of one cubic foot per minute. The filter cartridges are located at the intake of the collection tubes. This means that, when air is being drawn through one tube, the other 23 must be closed. In addition to the control for selecting the sample tube to be used, the air flow through the sampling tube is recorded in the remote control room.

Twelve sampling devices are required, one at each location of the removable plateout samples. However, the sampling tube intakes and

exhausts are located at a sufficient distance from the plateout and gas samplers to preclude disturbance of the normal air currents in the building.

(b) Particle Size and Location. The particle size of the fission products is identified as a function of position inside the containment vessel. These data are of value in evaluating the biological effects and filter efficiency following a loss of coolant accident.

Because of the environmental conditions inside the containment vessel, the most common techniques for determining particle size such as the sticky paper and impactor techniques produce questionable results. However, this technique is being employed until a better one is developed.

Sticky paper samples are housed in a box type enclosure approximately 12 in. by 12 in. by 2 in. with a spring loaded lid. After a sample has been collected, the lid is closed so that decontamination can be performed prior to removal of the samples. Two such samplers are located at each position. One is in a vertical position and the other in the horizontal. The vertical position proves more selective for small particles while the horizontal position is more selective for larger particles. Twenty-four samplers (2 at each position) are required.

Six impactors are installed inside the containment vessel and samples are collected after most of the condensed moisture has settled.

Control for remotely closing the sticky paper boxes and turning the impactor motors on and off are located in the remote control room.

(4) Radiation Monitors. Remote area gamma radiation monitors (RAM) are required in the containment building to measure the fission product distribution during core meltdown. These monitors will be capable of continuous operation in the environmental conditions expected in the building following the loss of coolant. Since the plateout of fission products on the detector head will result in erroneous readings, it is necessary to remotely decontaminate the head before taking a reading or to house the head in an enclosure to preclude plateout.

The RAM units have two consecutive ranges of three logarithmic decades each; thus requiring the appropriate remote range switching mechanisms and logarithmic recorders.

The recording instruments are located in the control room of the LOFT facility with slave instruments located in the remote control room and the health physics monitoring area.

The general location of each detector is as follows:

- (1) One above the locomotive entrance door,
- (2) one near the shielded sample window,
- (3) one in the dome,
- (4) one near the personnel entrance to the building,
- (5) one approximately 180° from position 4, and
- (6) one approximately five feet above the reactor vessel head.

The detectors at positions 1 through 5 are located approximately five feet from the containment building wall.

(5) Condensate Sampling. Fission product concentrations in the condensate are obtained as a function of time following the loss of coolant test. These data provide information concerning the ability of condensate to trap and hold fission products. In addition, this information is needed to obtain an approximate balance on the total quantity of fission products released. The technique for obtaining these data is to obtain samples of water from the test area sump as a function of time.

(6) Fission Product Behavior Sample Recovery System. The removable gaseous and plateout sample containers are removed from the containment building as a function of time for analysis in the radio-chemistry laboratories at TAN-607 and the Chemical Processing Plant (CPP). The mechanisms for accomplishing this removal consist of motorized wheels with 24 sample containers attached to the periphery of each wheel. A wheel of fallout samples and a wheel of gaseous samples are located at each of the 12 sample stations within the containment building. At a selected time, an automatic programmer actuates each

wheel. This closes the sample container immediately prior to release into a gravity tube system which leads to an isolation tube penetrating the containment vessel. The isolation tube consists of the tube itself and two valves in series spaced to accommodate several sample containers. The containers are subjected to high velocity water washing in the isolation tube prior to release into shielded casks located in the sample room. The casks are then moved by monorail hoists to the loading area of the shielded roadway for transport to the TAN-607 area. The casks attenuate the direct radiation to acceptable tolerances as outlined in the latest ICC regulations for transport of fissionable materials and consistent with the operating contractor's health and safety standards.

c. Documentary System - Cameras

Motion picture cameras are employed, during the non-nuclear testing only, to document the blowdown phenomena and the response effects on the experiment and containment systems. Two cameras are installed in each of two shielded containers mounted high on the containment building vertical walls. Pressure and temperature are controlled within the shielded container to maintain ambient conditions. The cameras are loaded and unloaded manually and are equipped with remote controls to facilitate the required high speed photography.

Pin hole cameras are used during the actual loss of coolant test to determine the fission product transport and distribution within the containment vessel and to determine the time history of fuel migration to the lower regions of the reactor vessel following the core meltdown. Because of film sensitivity to gamma radiation, the cameras are remotely installed and removed from the test area or shielded prior to and following the loss of coolant test. A gamma pin hole camera aids in locating areas or objects emitting high levels of gamma radiation. The camera records optical and gamma images on successive frames of highly sensitive 35 mm panchromatic film. The assembly consists of three pieces; camera, electronic flash unit, and a remote control panel. The camera resembles a 35 mm camera of conventional design. A barrel-type lens assembly protrudes from the body and contains a bakelite slide shutter operated by a solenoid and spring return.

The cameras are housed in a shielded container to preclude contamination from the atmosphere to which it is exposed within the containment building. Pressure and temperature are controlled within the enclosure to maintain ambient conditions existing outside the containment building. The pin hole cameras are located adjacent to the containment vessel wall with one oriented to view the blowdown nozzle area and the other observing the catch basin of the reactor vessel.

d. Pressure Reduction Sprays

An independent spray system of carbon steel piping and piping appurtenances is installed in the containment shell which is capable of reducing the 24 psig pressure within the containment shell to one psig within four hours following the rupture of both the primary and secondary coolant systems (see Figure VII-15). In addition, the spray system is capable, if the spray is initiated 20 minutes after an accident, of reducing the containment shell pressure and hence uncontrolled shell leakage within a period of time sufficient to provide less than 300 rem dose to the thyroid at the NRTS boundary under the following meteorological conditions:

Average wind velocity	U - 2.0 meters per sec
Stability parameter	n - 0.5 (dimensionless)
Virtual diffusion coefficient) in the vertical plane)	$C_y - 0.280 \text{ (meters } n/2)$
Uncontrolled leakage rate at) 24 psig internal pressure)	0.2%/day

A core meltdown which releases 50% of the halogens to the containment shell is assumed for calculation of the thyroid dose. In meeting the above requirements, the initial containment shell pressure will be taken as the maximum design pressure.

The pressure reduction spray system is designed to deliver 200 gpm to multiport fog spray nozzles located in the containment shell dome to promote natural circulation.

Part of the water expelled from the primary and secondary coolant systems remains liquid and collects in the condensate sump. The pressure reduction spray system pump, rated at 200 gpm, 370 ft TDH, circulates the condensate through a cooler and to the spray nozzles. Service water at a flow of 350 gpm circulates through the cooler to remove heat from the spray water. The service water flows directly to the disposal well from the cooler.

The electric motor driven spray system pump is supplied from emergency power. The decontamination system pump serves as a standby to the pressure reduction spray system pump. Service water can be supplied to the condensate sump in the event that primary and secondary system liquids are not adequate for the spray system inventory. The spray system flow diagram is shown on Figure VII-15.

e. Decontamination System

Permanent facilities are provided for decontaminating all exposed surfaces inside the containment building (see Figure VII-15). These facilities include a decontamination solution preparation system, a deluge spray system, a high pressure jet spray system, and a decontamination solution recirculation system. All materials in contact with decontamination solutions are either type 304L stainless steel, bonstrand, or similar corrosion resistant materials.

Decontamination solutions are prepared in a 2500 gallon mix tank in the service building. The tank is provided with a steam heater and propeller-type mixer. The decontamination solutions are then pumped by the decontamination transfer pump (150 gpm, 50 ft TDH) to the decontamination sump located in the basement of the containment building which serves as a supply reservoir for the two spray systems. The decontamination solutions are pumped to the spray systems by the decontamination pump (200 gpm, 370 ft TDH). The solutions then drain back to the decontamination sump for recirculation or disposal to the liquid waste system (after neutralization, if necessary).

The pressure reduction spray pump (see description of containment pressure reduction system) is piped so that it can be used as a standby for the decontamination pump. Both pumps are on emergency electrical power. Sampling connections are provided at the pumps.

Downstream of the decontamination pump a shell and tube heat exchanger, with steam on the shell side, is used to heat the decontamination solutions.

Following spraying with decontamination solutions, surfaces are rinsed with demineralized water.

The deluge spray system, consisting of headers with spray nozzles at appropriate intervals, is arranged inside the containment shell to insure a complete wash of all exposed containment building and equipment surfaces. The high pressure jet spray system consists of several rotating head nozzles, each with a capacity of 60 gpm at 115 psig. These high velocity nozzles are installed at various locations in the building. All permanent surfaces are no more than 20 ft from a nozzle to give complete and thorough coverage in all directions. Water to each group of three jet spray heads is controlled by remotely operated valves.

The decontamination sump is of concrete construction and is designed for the same pressure as the containment shell. Interior surfaces of the sump are lined with carbon steel coated with corrosion resistant plastic.

Facilities for decontaminating localized, highly contaminated areas after access to the containment building consists of four, 125 psig steam outlets installed around the periphery of the building, a portable solution tank and a portable, manually operated, steam detergent nozzle gun.

f. Halogen and Particulate Removal System (Figure VII-16)

The purpose of the halogen and particulate removal system ~~is to limit the total thyroid dose (inhalation plus ingestion) to persons~~ outside of the containment building. The system limits the release of halogens and particulates to the environment from the uncontrolled leakage through the containment shell. Uncontrolled leakage from the containment building is a function of the containment building overpressure which is limited by the pressure reduction spray system.

Operation of the halogen and particulate removal system requires that the containment building pressure be essentially atmospheric. This can be accomplished by use of the pressure reduction spray system in

approximately four hours after pressurization of the containment, or by allowing the temperature and pressure to decay through heat loss from the containment building to the atmosphere over a longer period of time.

The halogen and particulate removal system filter train is an addition to the containment building exhaust system. Ducts and dampers allow the filter train to be bypassed during normal operation (see Figure VII-17). The containment building heating and ventilating system is used for: (1) exhaust and recirculation of air during normal reactor operation bypassing the filter train, (2) a once through filtration exhausted to the stack, or (3) recirculation of the containment building gases through the filters and back to the building.

Double valves in the heating and ventilation system ducts permit automatic isolation of the containment building upon a high activity signal from the stack gas monitors.

(1) Filter System. The filter train consists of four filter elements placed in series; (1) demister, (2) roughing filter, (3) absolute filter, (4) charcoal adsorber filter. The sampling system, as part of the filter system, can obtain samples from locations: (1) ahead of the filter system, (2) downstream of the absolute particulate filter and ahead of the charcoal adsorber, (3) downstream of the charcoal adsorber. The system filtering efficiency and activity distribution by particle size can be measured with the filter sampling system.

The filters and exhaust blowers of the filter system are located in a shielded pit external to the containment building.

(2) Operation. The halogen and particulate removal system is operated subsequent to a nuclear blowdown after the containment building pressure is at or near atmospheric. The pressure reduction sprays are operated prior to filter system operation to reduce the pressure and to reduce the amount of airborne halogen and particulates in the containment building.

The airborne halogen and particulate removal system operation recirculates the gas in the containment building through the filter train and back to the building at the rate of 10,000 cfm. Filter system operation is continued for 20 hours which reduces halogen activity to approximately

2.5×10^{-7} $\mu\text{c/cc}$. The principle airborne radioactivity present in the containment building subsequent to the 20 hour filter operation is essentially noble gases; these gases are discharged through the filters and up the stack at a controlled release rate.

The sampling system provides information before and after filtration as to: (1) radioactivity as a function of particle size, (2) activity of halogens, and (3) activity of noble gases.

The filters in the filter train are removed into shielded containers and disposed of subsequent to filter operation. Iodine desorption from the charcoal filter material has not been evaluated during the transportation and disposal operation.

The activity present in the containment building gas is measured by portable air monitors downstream of the halogen and particulate removal filters. The monitors measure particulates, halogens and gaseous activity.

g. Radiological Grid

To assess the hazards resulting from a loss of coolant accident, extensive radiological measurements are required external to the containment building. These measurements are designed to evaluate both direct radiation from fission products contained in the containment building and the possible radiological doses resulting from fission products leaking to the surrounding environment.

The fission product leakage occurs for several hours following the core meltdown. Thus, an experimental technique is employed to obtain data during all conceivable meteorological conditions that may be expected at the site location to provide the required measurements. Radiological detection instruments are located on a grid consisting of concentric circles around the building extending from 100 to 6400 meters. Instruments on two 60-degree sectors of the grid are placed at intervals ranging from 3 to 20 degrees apart to provide extensive coverage during both inversion and lapse conditions. The remaining portion of the grid

contains instruments located at 15 and 30 degree intervals to provide data during the meteorological change between lapse and inversion conditions. Part of this instrumentation is also used to measure the direct radiation from the containment building caused by the release of radioactive effluent. The general layout of the grid and instrument locations are presented in Figure VII-18. The type of instruments employed and the information obtained from each is as follows:

<u>Instrument</u>	<u>Information Obtained</u>
1. High volume air sampler	Concentrations of airborne radioactivity
2. Cascade impactors	Airborne radioactivities as a function of particle size and density
3. Telemetering area monitoring stations	Concentrations of airborne radioactivities, direct radiation intensities, and doses
4. Fission gas detectors	Inert radioactive fission gas concentrations
5. Fallout plates	Fallout radioactivity
6. Film badges	Total beta-gamma doses (0.03 to 10^3 R)
7. Chemical and glass dosimeters	Total gamma ray (25 to 10^6 R)
8. Nuclear accident dosimeters	Thermal and fast neutrons doses
9. Remote area monitors	Gamma ray direct radiation intensities

Tracking and monitoring of the radioactivity plume beyond the external arcs is performed by mobile surveillance vehicles. As soon as permissible exposure conditions exist, these vehicles will also be used in the recovery operations of the various filters, instruments and samples.

A mobile field trailer complex is used as a radio control point for the installation and recovery of sampling media from the various sections of the grid. The trailer is also used for equipment and clothing storage, first aid facilities, a change room and decontamination. The location of the trailer is at the optimum position for use as a control center.

The power distribution system for the LOFT radiological grid instrumentation consists of a main feeder originating at the control and equipment building and branching out to a number of power transformers located at various points on the arcs. Power from the control and equipment building supplies the instrumentation on arcs at distances of 100, 200, 400, and 800 meters from the centerline of the containment building (see Figures VII-19 and -20).

h. Controlled Leakage

A means is provided for venting the containment shell pressure in a controlled manner to the atmosphere commensurate with postulated leakages from typical reactor containment shells. The controlled leakage to the atmosphere can be varied from 0% to 5% of the net volume per day at internal pressures ranging from five psig to the design maximum. A controlled vent system accomplishes the required venting. A remotely operated control valve is manually controlled to give the desired controlled leakage flow which is recorded on an indicating recorder. The controlled leakage systems schematic is shown on Figure VII-21.

i. Leakage Test Systems

Inasmuch as the techniques of leakage evaluation and measurement on a vessel of this magnitude have not been developed to a standard, various methods are being investigated and compared by Phillips Petroleum Company. Several of these techniques are described below.

(1) Reference System Method. In this method, the change in pressure of the containment vessel air is measured relative to the pressure of a hermetically sealed and pressurized reference system which is located within the containment vessel structure. The difference in pressures is measured by connecting the vessel interior with the reference system through a U-tube water manometer.

The leakage out of the container is calculated according to the following equation.

$$\frac{\Delta w}{\Delta t} = \frac{W_o}{\Delta t P_o} \left[\frac{T_o (\Delta p)_1}{T_1} - (\Delta p)_o \right]$$

where $(\Delta p)_1$ = final pressure difference between container and reference system

$(\Delta p)_o$ = initial pressure, etc.

W_o = original container air contents

Δt = period of test

P_o, P_1 = initial and final absolute pressure, respectively

T_o, T_1 = initial and final absolute temperatures, respectively.

The effect of poor sensitivity in temperature measuring instruments is minimized in the reference system method due to the way the ratio T_o/T_1 enters the leak rate equation. In this equation the temperature ratio is modified by the factor $(\Delta p)/p$.

Temperature gradient cause many errors in the reference system. For this reason extreme care must be exercised in geometrical design and location of the reference system in the containment vessel. It has been found experimentally that temperature gradients are at a minimum from midnight to 8:00 a.m. due to the absence of solar heating on the containment shell.

Errors in the leak rate formula can also appear as a result of poor sensitivity of measuring instruments. In the reference system method the probability of error is reduced since a water manometer is used.

When using the reference system, the validity of all data is dependent upon the tightness of the reference system. Therefore, the reference system must be checked periodically by local leak test methods to insure reliable system integrity.

(2) Absolute Pressure Drop Method. In this method, the containment building is pressurized to the test pressure and the subsequent pressure loss as a result of leakage is observed over a period of time.

As with the reference system method, the perfect gas law is assumed, and containment volume changes are neglected. The same temperature and pressure errors evident in the reference system method are inherently evident in the latter method; however, these errors may be modified or magnified. Temperature instrument errors are more serious in the absolute method than in the reference method and two authorities state that in the first system an error of 0.045°F in temperature measurements is equivalent to an error of 0.025% when considering the containment vessel volume.

Temperature gradients are less severe with the absolute method because they may be compensated by proper placement of thermistors and thermocouples. Also, temperature lag times are less evident in the absolute method because the response times of all instruments can be made the same. Because of the nature of the absolute method, the restriction concerning average temperatures of the containment vessel in the reference system does not exist in the absolute pressure drop method.

(3) Makeup Volume Technique. In this method, instead of allowing the pressure to decay (as is done in the other two methods), the pressure is held constant by adding air to compensate for leakage. The leak rate is found by measuring the quantity of air added during a certain period. The air is measured by using either a flow meter or by weighing the compressed air bottles.

It does not appear that this test offers any advantages since it is subject to the same sources of temperature and pressure errors as are the other two methods. For continuous monitoring, however, this method has an advantage since the air flow into the container could be measured each day. This method will give only gross changes and can be used only to augment one of the other methods.

(4) Local Leak Test Methods. These methods are more accurate than the integral test methods since they are not affected noticeably by temperature and pressure variations. The only errors associated with local-leak tests are in the measurements of the leakage flow. The accuracy of these measurements is affected by the cross

sectional area of the leak. Babcock and Wilcox give the following limits as the maximum accuracy available.

Soap Bubble and Halide Leak Detector

$$10^{-2} \text{ cm}^3/\text{hr}$$

With Mass Spectrometer Added

$$10^{-3} \text{ cm}^3/\text{hr}$$

Listed below are local leak tests which are used. These are generally impracticable due to the immensity of the containment vessel.

(a) Soap Bubble Test. A pressure differential is created across the leakage path to be tested. A soap solution is then brushed over the test area and the formation of bubbles is observed. The leakage rate is proportional to the rate of growth of the bubbles.

(b) Halide Leak Detector Test. Some type of halide gas such as Freon is dispersed in low concentration into the volume of air to be tested. To determine the leakage, the air in the region of the leakage path is analyzed to determine the amount of Freon which has escaped.

In performing the air analysis, the reduction in ion emission from a heated metal surface as a result of contact with the halide gas is measured.

(c) Air-Ammonia Test. Ammonia is dispersed in the container and leakage is detected by noting the white "fog" which forms when a swab soaked in hydrochloric acid is placed in the vicinity of the leak.

(d) Vacuum Test. A vacuum box is placed over the test area and the increase in pressure in the vacuum box is a measure of the leakage rate.

5. Utility Systems

a. Water Systems

(1) Supply. Water is supplied by two 1000 gpm, 350 ft TDH 125 hp deep well pumps and is automatically chlorinated before entering the 500,000 gallon above grade storage tank. The pumps and storage tank, part of the original FET facility, were adequate to meet the LOFT requirements.

The storage tank is equipped with a level indicator, high and low level alarms and high and low level pump control. Alarms are connected into the ADT system. A bypass around the storage tank permits system operation when it is necessary to remove the tank from service for maintenance.

(2) Domestic and Service Water (see Figure VII-22). The original FET facility provided a domestic and service water system which was considered inadequate for LOFT. It included one 500 gpm pump at 255 ft Hd, and one 100 gpm pump at 255 ft Hd, plus a cooling tower and auxiliary equipment. The FET cooling tower was dismantled, and stored, and a "once through" system was substituted.

The 500 gpm pump is being used in LOFT. The 100 gpm pump was replaced with two 387 gpm 150 ft Hd pumps.

(3) Demineralized Water (see Figure VII-23). The demineralized water system is sized in accordance with the following criteria:

- (1) Two primary coolant system flushes in one hour, followed by filling of the primary coolant system.
- (2) Decontamination using 150 gpm of demineralized water for four hours.
- (3) Boiler makeup.
- (4) Control rod seal and cooling, estimated at approximately 6 gpm.
- (5) Instrumentation ion chamber cooling, estimated at 10 gpm.
- (6) Secondary coolant system fill.
- (7) Primary and secondary coolant system makeup.

Mixed bed demineralizers are used to produce deionized water with a specific resistance of 500,000 ohms/cm³. Concentrated sulphuric acid for regeneration of the mixed beds is supplied from a portable 500 gallon carbon steel tank. The acid is transferred to the mix tank by gravity. Flake caustic is mixed with demineralized water to provide caustic for regeneration.

Acid and caustic are introduced into the demineralizers through separate eductors during regeneration of the beds.

(4) Fire Protection Water

(a) Source of Fire Protection Water (see Figure VII-24). The existing 500,000 gallon steel water tank provides storage for fire protection water as well as service and potable water. No separation is made between service water storage and fire protection storage since service water is required at all times during a fire or during a power failure. Fire protection water and service water are supplied through a buried 20 in. diameter pipe from the 500,000 gallon tank to the pump room at the north side of the control and equipment building. The fire pump header is connected to this 20 in. pipe.

(b) Fire Pumps. Two horizontal centrifugal pumps, each rated at 1500 gpm and 300 ft discharge head, supply fire protection water to the hazard areas. One fire pump is electric-motor driven (normal power) and the other is gasoline-engine driven. Each pump is capable of supplying the design fire demand of 1200 gpm.

(c) Building Drainage. The service building floor is below ground level. This prevents gravity drainage of fire protection water from the building in the event of fire and it is necessary, therefore, to provide a sump and sump pumps to prevent building flooding at the time of a fire.

(d) Hazard Areas and Fire Protection Features

1) Fuel Pump Room. The fuel pump room is located at the southeast corner of the FET control and equipment building. The foam extinguishing system installed in the pump room is

supplied from a fire pump header and from a buried foam storage tank north of the control and equipment building.

2) Boiler Area. The oil burning boilers, located at the south side of the existing control and equipment building, are protected by fog nozzles supplied from the fire pump header.

3) Diesel Generator Area. The diesel generator is protected by fog nozzles connected through a manually controlled valve to the fire header. An automatic high pressure, bottle storage carbon dioxide local application system is provided at the diesel generator. The manually controlled fog nozzles thus provide water backup for the CO₂ primary protection system. CO₂ is the preferred fire protection at the diesel generator since it prevents water damage and avoids electrical hazards.

4) Other Interior Hazards. All other interior hazards are of a general nature and are not as localized as the fuel pump room, boiler area and diesel generator. These hazards consist of accumulations of trash or waste paper, spilled paint thinner, grease or oil fires at mechanical equipment, overheated electrical motors, or short-circuit fires at electrical switch gears. Protection against all of these general hazards is secured by small hose outlets. The hose outlets in the FET hangar building and control and equipment building are connected to the fire water header. Small hose outlets are also provided in the containment building and the service building.

5) Yard Hazards. Such hazards as burning vehicles, burning trash, electrical fires at the substation, or fires at the diesel and fuel oil unloading areas may be expected in the general yard area. Protection is provided by hydrants so located that all areas of the yard are within 300 ft of a fire hydrant. The hydrants are supplied from the fire main loop, a portion of which was relocated to make room for the new construction. ID standard hose houses are provided at all hydrants.

b. Waste Systems

(1) Liquid Waste Disposal System (see Figure VII-25.)

(a) General. The ID Manual, Chapter 0500-7 sets forth policies and procedures to be followed in waste disposal at the National Reactor Testing Station. The basic guides for the disposal of the liquid radioactive waste are: "The concentration of liquid radioactive waste discharged to the ground or ground water shall be maintained at levels such that the concentration in water at any point of use shall not exceed a concentration which will result in a dose to individuals in excess of 1/10 of the Radiation Protection Guide (RPG) values".

Figure VII-26 establishes the concentration limits for liquid waste discharged to the ground at facilities in the NRTS. This chart is a supplement to the provisions of the ID Manual 0500-7.

The values obtained by the use of this chart are applicable to the point of discharge, or at the points of last sampling or monitoring. Decay, absorption and dilution have been considered in establishing the above standards. The values are applicable to radioactive liquids for which the isotopic composition is known. If composition is not known, the isotopes will of necessity be considered all long-lived (greater than 1000 days).

(b) Classification of Liquid Wastes

1) Hot Wastes. Hot wastes includes those effluents which are above the radioactivity tolerance for discharge to the disposal well. The sources of this class of effluent include containment building decontamination solutions, quench tank discharge, spent resin casket drain, primary coolant system blowdown, fuel storage cell drains, and sample room drains.

2) Warm Wastes. Warm wastes includes those effluents which will normally have a small amount of radioactivity such that the material is within the allowable tolerance for discharge to the disposal well. The principle source of this effluent is the decontamination rinse water and equipment room drains in the containment building.

3) Cold Wastes. Cold wastes includes those effluents which may be considered as never containing radioactivity such as equipment drains, boiler blowdown, floor drains from equipment rooms and tunnels in the control and equipment building.

4) Corrosive Wastes. Corrosive wastes includes those effluents which never contain radioactivity but are highly corrosive and require neutralizing before disposal. Sources of this class of effluent are the acid and caustic regeneration waste from the demineralizers and laboratory wastes.

(c) Liquid Waste Systems

1) Hot and Warm Waste System. During normal reactor operation, service and demineralized water used for cooling dolly-mounted equipment flows into the dolly-mounted equipment drain. From there it normally flows by gravity to the disposal well. A radiation monitor in the drain line sounds an alarm in the control room, on high activity, permitting the cooling water to be diverted to the hot waste storage tanks.

Prior to the primary coolant system blowdown, all isolation valves (remotely operable), in the liquid waste lines penetrating the containment shell, are closed. Drains from dolly-mounted equipment then flow into the 5000 gallon condensate sump in the operating floor.

Following the blowdown, liquid wastes and condensate on the operating floor collect in a peripheral trench and flow by gravity into the filter sump, where debris is removed, and then into the condensate sump. Drains from dolly-mounted equipment flow into the condensate sump and mix with condensate from the blowdown. Provisions are made to obtain samples of the liquid in the condensate sump. Following containment pressure-reduction operations and prior to decontamination operations, waste in the condensate sump is pumped, by the containment spray pump, to either the hot waste storage tank or to the disposal well, depending upon activity level. The piping is arranged so that the decontamination pump can be used as a standby for the containment spray pump. Both pumps are on emergency power and are operated from the control room.

Drains from the primary coolant system purification demineralizers flow from the spent resin cask on the dolly, to a 250 gallon stainless steel hot waste transfer tank. The quench tank and the fuel element storage cells drain into the hot waste transfer tank. Provisions are made to obtain samples of the liquid in the hot waste transfer tank; from this tank the hot waste transfer pump can pump to either the hot waste storage tanks or to the disposal well, depending on the activity level.

A hot waste gravity drain line from the sample room discharges into a sump and is pumped to the hot waste storage tank.

Liquid wastes in the LOFT containment building equipment rooms collect in a sump, from which samples of liquid can be obtained. The equipment room sump pump can transfer the waste to either the disposal well or the hot waste storage tank, dependent upon the level of activity.

Liquids drain by gravity from the railroad door trench through a filter and then to a waste sump from which it is pumped to either the disposal well or hot waste storage tank.

Parts of the liquid waste system are used in conjunction with the containment pressure-reduction system and also in conjunction with the decontamination system.

The FET liquid waste system consists of a collection system and a 15,000 gallon carbon steel buried tank which receives warm waste from the wash and shower areas in the control and equipment and the FET test building. The tank is equipped with a 100 gpm pump which transfers liquid waste to a truck connection or to grade. The waste tanks are equipped with provisions for obtaining samples to determine liquid waste activity levels prior to discharge of the effluent.

2) Hot Waste Storage Tanks. Two 25,000 gallon stainless steel hot waste storage tanks are located underground in the yard adjacent to the containment building. A 150 gpm 50 ft TDH vertical turbine pump mounted on each tank is provided to discharge the hot wastes from the tank. Each pump is remotely manually operated and operation is governed by the remote level indicating devices.

The discharge piping is manifolded so the fluid from the tanks can be pumped to the truck loading connection or to the disposal well. Sample connections are provided for sampling the tank contents which dictate whether the liquid is pumped to the disposal well or to a truck connection for transfer to the TAN liquid waste evaporators. Other provisions include vent lines to stack and 50 gpm steam eductors to remove liquid wastes from the tanks in the event of a pump failure.

3) Cold Wastes. Floor drains for cold wastes are provided in the LOFT service area. Two emergency cold waste pumps having a rating of 200 gpm at 50 ft TDH discharge water to grade in the yard. The pumps are controlled by level switches.

4) Corrosive Wastes. The corrosive waste drain system carries the acid and caustic regeneration wastes from the demineralizers and other acid wastes. Waste flows by gravity in separate corrosion resistant lines to a neutralizing pit located in the yard area. From the neutralizing pit the neutralized wastes are pumped to the leaching pond.

5) Piping. Materials of construction for pipes that normally handle hot and warm wastes are stainless steel type 347 or 304L. Lines that normally handle cold wastes are carbon steel inside buildings and cast iron mechanical joint in the yard.

6) Disposal Well. A continuous sample is taken of the waste entering the disposal well. In addition, total flow and activity are recorded.

7) Sanitary Waste. Sanitary waste is discharged from the control and equipment building through a 6 in. concrete pipe to a septic tank. The septic tank effluent is chlorinated, discharged to a sand filter, and thence to the disposal well. All of the components of the sanitary system are large enough to accommodate the LOFT sanitary discharge (100 gpm maximum flow). The only change required for LOFT was to re-route the 6 in. concrete pipe between the manhole and the septic tank in order to avoid the new buildings.

8) Service and Potable Water Waste. The once-through discharge from the service and potable water systems (1200 gpm maximum flow) discharges to the disposal well.

(2) Solid Waste. The NRTS burial ground is adequate for the amount of solid waste expected from LOFT. Handling equipment for this waste also is adequate.

(3) Gaseous Waste. The warm and hot waste tanks are vented to the suction side of the exhaust filter equipment. The exhaust filter system also handles the exhaust air from the containment building. The exhaust filters remove particulates and halogens from the exhaust stream before it is discharged to the stack. Installed in the 150 ft stack are monitoring probes connected to monitoring equipment located in the filter vault near the base of the stack (see Section VII-B-4-f, Halogen and Particulate Removal).

c. Plant and Instrument Air System

The two FET air compressors are utilized to supply the requirements of both LOFT and the necessary FET facilities (see Figure VII-27). Both compressors are oil-lubricated, reciprocating types. One compressor has a free air capacity of approximately 100 cfm and a discharge pressure of 200 psig, while the smaller compressor has a free air capacity of approximately 55 cfm and a discharge pressure of 250 psig. Oil vapor and moisture are removed from the compressed air by means of absorbent materials. There is a total of 900 ft³ storage capacity (two receivers; one 750 ft³, one 150 ft³) at a minimum pressure of 150 psig.

d. Heating and Ventilating Systems

(1) Control and Equipment Building Heating and Ventilating. The FET control and equipment building heating and ventilating system was adequate, with minor modifications, for the LOFT facility. These modifications were: (1) install motor operated dampers in the heating and ventilating air intake and exhaust shafts to isolate the building during periods when radioactive contamination is present in

the outside air, (2) supply combustion air to the diesel engine from an outside air duct, (3) provide for shutdown of the steam boilers during periods of isolation, and (4) shutdown the existing exhaust fans and provide isolation valves in ducts.

The control and equipment building is equipped with O_2 generating and CO_2 absorption canisters for use during periods of building isolation.

(2) Containment and Service Building Heating and Ventilating. The containment and service building heating and ventilating systems consist of two heating and ventilating units serving the containment building and one air washer unit for the service building. The two heating and ventilating units and the air washer are located in the service building equipment room. All units are supplied from one common outside air intake which is equipped with adjustable louvers. The shielded air intake shaft extends down through the equipment room roof.

(a) Containment Building Heating and Ventilation System. The two heating and ventilating units supplying the containment building consists of outside air, motor operated isolation valves, filters, steam heating coil, face and bypass dampers and supply fan. The two units supply air into one common duct which penetrates the containment shell through a labyrinth and terminates inside. Two isolation valves in series in the supply duct are located outside the containment shell.

The containment building has five air changes per hour with at least two air changes being outside air. During periods of high temperatures all five air changes can be outside air. The recirculation duct in the exhaust-recirculation system contains a steam coil to maintain the building at 70°F during normal operation. The steam coil has the capacity to maintain the building at 100°F prior to blowdown by heating the recirculating air.

The exhaust-recirculation ventilation system, as shown on the flow diagram (see Figure VII-17) is designed for operation when high radioactivity in the containment building necessitates closure of the outside air ventilation isolation valves. A 10,000 cfm capacity filter

train in the exhaust-recirculation system provides for removal of radioactive contaminants. During normal operation, the filters in the exhaust-recirculation system are bypassed.

The two exhaust-recirculation fans equipped with variable inlet vanes are capable of circulating wide ranges of air volume through the exhaust-recirculation loop. The motor-operated dampers in the exhaust fan duct controls air discharge to the stack and the amount of air recirculated through the heating coils and back into the containment building.

(b) Service Building Heating and Ventilating. The service building heating and ventilating system consists of an outside air intake, isolation dampers, outside air dampers, return air dampers, air filter, preheat coil, air washer reheat coil, and supply air fan.

The service building heating and ventilating system supplies the equipment room, electric switch gear room, pipe labyrinth spaces, remote control room, and containment building basement. Air is supplied to the equipment room and circulated through the electric switch gear room and battery room then exhausted to the stack by means of an exhaust fan. Air supplied to the remote control room is also exhausted to the stack by an exhaust fan. The air supplied to the pipe labyrinth spaces leaks back between the pipes and the labyrinth wall into the equipment room. Air is not normally returned from the service building. A return air duct is provided to allow for return air during isolation of the service building.

Air is supplied to the containment shell basement through two motor operated isolation valves in series. Exhaust air is taken from the basement through two motor operated isolation valves by an exhaust fan which discharges into the stack.

(3) Well Houses Heating and Ventilating. Well pump houses No. 632 and 639 can be isolated to prevent possible contamination from outside air during a LOFT test. A cooling coil and fan are provided for equipment cooling during the periods of isolation.

e. Steam System (See Figure VII-28)

There are two existing 300 boiler horsepower "packaged" boilers which supply 15 psig steam to both FET and LOFT. These boilers burn No. 6 fuel oil. The auxiliary operations and equipment are conventional and standard, viz., the fuel oil system, the feedwater chemical treatment system, the feedwater supply system, the boiler blowdown system, and the condensate return system. In general, condensate is recovered and reused as feedwater, the only exception being condensate from fuel oil heating which goes to waste.

For decontamination operations (steam to the decontamination nozzles, heater, and solution mix tank), a new 100 boiler horsepower "package" boiler supplies steam at 125 psig. This boiler also supplies 125 lb steam to the eductors at the hot waste storage tanks. Condensate from the decontamination heater discharges to the hot waste system; condensate from the decontamination solution mix tank discharges, normally, back to the feedwater tank, but can be diverted to the leaching pond if a conductivity element indicates a leaking tube in the heater. The high pressure boiler has its own preheating feedwater tank and feedwater pumps. One of the feedwater pumps is on emergency power.

f. Electrical Power

(1) Transmission and Distribution System. Commercial NRTS electric power is supplied to the combined FET-LOFT loads from the TAN substation over the rebuilt FET 13.8 kv, wood pole, overhead transmission line (see Figures VII-29 and -30). Power is supplied to the overhead transmission line via underground 15 kv cables from cubicle No. 13 in the TAN substation, 13.8 kv, 140 Mva, switchgear. Power is supplied to the new FET-LOFT substation via underground 15 kv cables from the transmission line terminal pole using underground ducts and manholes. Power is supplied to the FET well pump substations over the 13.8 kv overhead tap line.

Expulsion-type, 15 kv lightning arresters are provided on the transmission line at each overhead cable and substation termination and at intervals of 2000 ft along the length of the line. Open-type

15 kv fused disconnect switches are provided in the transmission line ahead of each substation.

The well pumps are supplied by their 112.5 kva, 13.8 kv to 480 v transformers. These transformers are located on the local wood pole, H-frame substations.

The FET-LOFT area is supplied by the new FET-LOFT substation. This substation consists of three transformers, each protected by an integral weatherproof 13.8 kv, 150 Mva, switchgear-type primary air circuit breaker. One transformer, rated 1500 kva, 13.8 kv to 480 v, is relocated from the previous FET substation and supplies power to the modified FET 480 v switchgear. The second transformer is a new 1000 kva, 13.8 kv to 2400 v unit and supplies power to the LOFT 2400 v switchgear. The third transformer is a new 750 kva, 13.8 kv to 480 v unit and supplies power to the LOFT 480 v switchgear.

(2) Normal Power Systems

(a) 2400-Volt Power System. The 2400 v power system starts at the terminals of the LOFT 1000 kva, 13.8 kv to 2500 v transformer and includes the LOFT 2400 v switchgear and feeders to motors and power loads. This power system supplies power to the LOFT primary coolant pump, the electrically driven fire pump, the radiological grid, and the test dolly photographic lighting system.

(b) 480-Volt System. The 480 v power system is divided into two subsystems. The first subsystem starts at the terminals of the FET 1500 kva, 13.8 kv to 480 v transformer and includes the normal power section of the modified FET 480 v switchgear and feeders to motor control centers, motors, lighting transformers, and power panel boards. This system supplies power to all FET loads and those LOFT loads located within the FET building complex which operate from normal power only. This system also supplies normal power for reactor instruments, health physics instruments, test instruments, process instruments and associated controls including the reactor safety system 120 v AC normal power bus.

The second subsystem starts at the terminals of the LOFT 750 kva, 13.8 kv to 480 v transformer and includes the LOFT 480 v switchgear, and feeders to motor control centers, motors, lighting transformers, and power panel boards located within the LOFT buildings.

(3) Emergency Power System. The emergency power system starts at the terminals of the FET 1000 kw, 480 v diesel engine-generator unit and includes the emergency power section of the modified, FET 480 v switchgear and feeders to motor control centers, motors, lighting transformers, and power panel boards. This system operates continuously and independently of the normal power system during periods when test operations are in progress to supply power to FET and LOFT critical auxiliaries, emergency lights, instruments, and controls which are required for safe shutdown of the test reactor on loss of normal power. The reactor safety system 120 v AC emergency power bus is supplied from this system.

During periods between tests or when the test conditions are not critical and when the test reactor does not require shutdown cooling, the diesel engine-generator is shut down and the emergency power system is supplied with normal power through a tie circuit breaker to the normal power section. This tie breaker is interlocked to prevent operating the diesel engine-generator unit in parallel with the normal power system.

(4) Failure-Free Power System. The failure free power system consists of a 10 kw motor-generator set normally running off plant power and charging a 125 v d.c. battery bank. Upon failure of commercial power, the flow of power instantaneously reverses and the battery bank drives the MG set to keep the AC bus energized with no noticeable voltage dip. All of the critical reactor controls and instrument systems are connected to this source of power.

(5) Lighting Systems

(a) Normal Lighting. The normal lighting systems for the FET area are supplied power at 480 v and 120/208 v. The FET test building lighting is 480 v from a power distribution panel connected

to the FET 480 v switchgear. The remainder of the normal lighting in the FET area is supplied from lighting distribution panels which receive power from a 225 kva, 480 v to 120/208 v, three phase transformer connected to the FET 480 v switchgear. Normal lighting in the LOFT area is supplied from a lighting distribution panel which receives power from a transformer connected to the LOFT 480 v normal power switchgear.

(b) Emergency Lighting. The emergency lighting systems for the FET area and the LOFT area are supplied power at 120/208 v. Emergency lighting in the FET area is supplied from emergency lighting distribution panels which receive power from a 30 kva, 480-120/208 v, three phase transformer which has been reconnected to the FET 480 v emergency power motor control center.

g. Communication and Alarm Systems

(1) Telephone Systems. The FET area telephone system is a commercial system, connected to the NRTS system. It provides operational and emergency communications for all buildings. This system includes the LOFT area remote control room, sample room, elevator, decontamination equipment room and electrical equipment room.

(2) Intercommunication System. The intercommunication system for the LOFT buildings is an extension of the FET area intercommunication system and provides two-way units in the remote control room, the sample room and outside at the entrance to the containment building. This system is controlled from the master paging control unit in the control room in the control and equipment building.

(3) Fire Alarm System. The fire alarm system for the LOFT facilities is an extension of the FET area fire alarm system with additional manual fire alarm boxes, alarm gongs, supervisory detector units and automatic alarm transmitters provided as required. Supervisory signals and coded fire alarm signals are annunciated at the TAN fire station, at the main TAN security station and at the NRTS central fire station as well as at the LOFT main control room.

(4) Evacuation Alarm System. The evacuation alarm system consists of an evacuation siren located in the containment building test area which may be actuated from either the main control room or the Technical Services Facility (TSF) area guard house. Annunciation of operation of the evacuation alarm system is provided at the LOFT area and the TSF area.

Fig. VII-1 - Containment Building - Outside Elevation

Fig. VII-2 - Containment Building - South Sectional Elevation

Fig. VII-3 - Containment Building - West Sectional Elevation

Fig. VII-4 - Containment Building - Basement Plan

Fig. VII-5 - Railroad Access Door

Fig. VII-6 - Railroad Door Seal System

Fig. VII-7 - Containment Building - Ground Floor Plan

Fig. VII-8 - LOFT-FET Plot Plan

Fig. VII-10 - LOFT Containment Building Shielding Labyrinth

Fig. VII-11 - Railroad Turntable Modifications

Fig. VII-12 - TAN Shielded Locomotive

Fig. VII-13 - Secondary Coolant System Flow Diagram

Fig. VII-14 - Secondary Coolant System Instrument Diagram

Fig. VII-15 - Spray System Flow Diagram

LEGEND

R	ROUGHING PARTICULATE FILTER
A	ABSOLUTE " "
M	MILLIPORE FILTER
I	IODINE REMOVAL FILTER
X	Xe " "
K	Kr " "

Fig. VII-16 - Particulate and Halogen Removal System

Fig. VII-17 - Heating and Ventilating System Flow Diagram

LEGEND:-

- - Film Badges & Fallout Plates
- - Telemetering Stations
- - Fission Gas Detectors
- △ - Ion Chambers
- ▲ - High Volume Air Samplers
- - Nuclear Accident Dosimeters
- ✦ - Low Volume Air Samplers
- P - Polymerization & Chemical Dosimeters
- M - Meters

Fig. VII-18 - LOFT Radiological Grid System

Fig. VII-19 - Electrical Power Distribution to Inner Radiological Grid

Fig. VII-21 - Controlled Leakage System Schematic

Fig. VII-22 - LOFT Service Water Distribution Schematic

Fig. VII-24 - Fire Protection Water Flow Diagram

Fig. VII-25 - Liquid Waste Disposal System

EXAMPLE: FIND THE MAXIMUM ALLOWABLE MONTHLY
CONCENTRATION. ENTER THE CHART ON THE
ABSCISSA WITH THE HALF-LIFE IN DAYS, INTERSECT THE CURVE, AND READ THE MULTIPLE ON THE ORDINATE
eg. ISOTOPE WITH 12-DAY HALF-LIFE MAY BE DISCHARGED SO THAT THE MONTHLY AVERAGE CONCENTRATION
DOES NOT EXCEED 30 TIMES M P C (FOR CONTINUOUS USE)

Fig. VII-26 - Maximum Discharge of Radioisotopes to the Ground

Fig. VII-27 - Plant and Instrument Air System

Fig. VII-28 - Steam System Flow Diagram

Fig. VII-30 - LOFT Electrical Distribution System

SECTION VIII

SAFETY ANALYSIS CRITERIA

VIII. SAFETY ANALYSIS CRITERIA

A. General

In this section are set down the basic criteria, ground rules and philosophy being used in analyzing accidents for the LOFT facility. This facilitates determination of credible accident sequences, and assessment of those accidents which may require design modifications or engineered safeguards as the detailed design proceeds.

The following criteria are not regarded as exacting requirements, but rather as general guidelines which must be tempered by judgment.

B. Criteria

1. Operator Action

In analyzing credible accident sequences for the LOFT facility, it cannot be assumed that the plant operators will respond instantaneously to various accident situations to take correct action or clear a fault. The following are criteria for operator action from various plant locations:

- (1) If action can be taken from the main control room, 20 minutes is the time interval within which the operator can be depended on to take action.
- (2) If the action must be taken outside the main control room in accessible areas within the control and equipment building, or the remote control and sample rooms, 45 minutes is the time interval within which the operators can be depended on to take action.
- (3) If the action must be taken in accessible areas of the test building basement, 60 minutes is the time interval within which the operators can be depended on to take action.
- (4) For the yard facilities and test area inside the containment shell, which will be inaccessible following a blowdown test or serious accident of MCA magnitude, operator action cannot be relied on.

The significance of the time at which operator action can be depended on is that the design must be such that the action will not be required within this time interval. These times are those required for the proper assessment of the accident situation and the first corrective action; however, it need not be assumed that an equal time would be required for each subsequent action.

2. Credibility of Accidents

Ideally, it would be desirable to evaluate the probability of each accident sequence to be considered. It would then be possible to trade accident probability for accident severity. However, at the present state of the art it is not feasible to factor accident probability into accident analyses in a quantitative way. It is necessary, nevertheless, to have a consistent set of assumptions as a basis for determining the credibility or incredibility of a postulated accident sequence. The following assumptions serve as a guide to evaluating the credibility of an accident.

a. Simultaneous Unrelated Accidents

The occurrence of simultaneous, unrelated accidents is considered incredible.

This is based on the qualitative reasoning that if two faults or failures are completely independent and each has a low probability, then the probability of both failures occurring simultaneously is vanishingly small and can be considered incredible. Caution must be exercised, however, that the failures are truly independent. In many cases, close examination may reveal that the failures are not, in fact, independent.

An example of independent failures might be rupture of a water storage tank at the same time a loss of commercial power occurred. However, consider a storage tank with a valve in the water supply line which is designed to fail closed on loss of commercial power. Then loss of water supply would be the direct consequence of a loss of power and not an independent accident.

Consider the example of a water line running directly over, or in immediate proximity to, an emergency power load center. It is credible

that failure of the pipe could lead directly to a loss of power to vital equipment connected to the load center.

b. Sequential Unrelated Accidents

As stated above, the occurrence of simultaneous unrelated accidents will be considered incredible. If enough time elapses following the first failure, the probability of a subsequent unrelated failure becomes increasingly large. Therefore, this sequential unrelated failure would have to be regarded as credible if enough time has elapsed.

Consider the example of a reactor plant whose characteristics are such that emergency heat removal from the reactor core must be provided for several months following a loss of coolant accident. If the components (pump, heat exchanger, valves, etc.) are inaccessible for maintenance during operation following the loss of coolant, or if repairs would take a considerable period, it might be considered prudent to duplicate the principal components of the emergency heat removal system. This would be on the basis that subsequent failure of a system component is credible during the prolonged period of operation.

c. Related Failures

In general, an initiating failure and one sequential related failure are considered credible. The occurrence of two or more sequential related failures are not considered credible unless it is known that the probability of each such failure is quite high.

An example of a sequential related failure would be an accident sequence involving the loss of power to a pump followed by failure of the pump's check valve to close or seat properly. Failure of the check valve to close is a related failure since it is only called on to function when flow from the pump has stopped. If the consequence of this accident sequence (pump failure followed by failure of its check valve to close) are unacceptable, the plant designers might decide to locate another check valve in series with the first valve. Failure of both check valves to close would be considered incredible.

Consider another example of an accident sequence initiated by a loss of coolant followed by a sequential related failure of a containment isolation valve to close or seat properly. This is a related failure since the isolation valve is called upon to function as a result of the first failure (loss of coolant). The consequences of this accident sequence can be reduced by locating a second isolation valve in series with the first. It would then be considered incredible that both isolation valves would fail to close, although either one separately might fail. If, however, there was a single signal source initiating closure of both valves, failure of this signal would effect both valves. Therefore, an independent signal source would have to be provided for each valve.

d. Consequential Failures

In some cases an initial failure will impose on other components loads and environmental conditions which exceed the requirements of normal operation. For instance, a loss of coolant accident will subject piping and certain equipment to pressure differentials, thermal and hydraulic shock, and environmental conditions of temperature, pressure and humidity. If the various components are not designed for the imposed load or environmental conditions, it must be assumed that they may fail, although not necessarily all at the same time. However, after careful analysis of the accident conditions it might be possible to design a vital component or system for the expected conditions, and the integrity of that component or system could be relied on for some period of time following the accident.

A particular item worth discussing is the primary heat exchanger or steam generator. The heat exchanger components which might fail during a loss of coolant accident are tubes, tube sheets, and internal baffles. There may be several hundred tubes in the exchanger. These tubes might fail due to the pressure differential across the tube walls or due to the loads imposed by the increased flow of fluids through the exchanger. If the tubes are not designed to withstand the expected loads, or if these conditions are not known, it must be assumed that many tubes could fail as a consequence of the depressurization. Even if the tubes are designed for the imposed load during the depressurization accident, it would still

be necessary to assume at least one tube failure. This is due to the fact that tube failures occur during normal operation because of pitting, erosion, vibration and rubbing against supports. In any event, therefore, it should be assumed that one tube may be in a weakened condition before the loss of coolant accident occurs.

e. Piping or Pressure Vessel Failure

For purposes of the preliminary SAR, the maximum credible accident rupture has been defined as a double-ended rupture located in the outlet piping of the reactor pressure vessel.

In general, all piping and vessels are considered as credible of failure. It may be possible to assume that certain vessels or sections of piping are incredible of failure, provided that certain conditions are satisfied. However, the number of instances where incredibility of failure is assumed will be kept to an absolute minimum, in order to maintain a position which can be most easily defended from a safety analysis standpoint. The conditions under which it might be possible to assume incredibility of failure are:

- (1) Design and fabrication are in accordance with recognized codes which require large safety factors.
- (2) The design is amenable to analysis, and all stresses have been thoroughly analyzed.
- (3) The loads which form the input to the analyses are well known. The loads considered should include transients to which the piping or component might be subjected, such as the thermal, hydraulic and shock resulting from a loss of coolant test.
- (4) Ultrasonic inspection of base material prior to forming, and in the case of ferro-magnetic materials a magnetic particle inspection would be performed on edges prepared for welding (to detect laminations).
- (5) Magnetic particle inspection of first weld pass, and depending upon the thickness of the material, several intermediate passes. Following thermal stress relief all finished welds are magnetic particle tested and radiographed. In some cases, such as

piping, the final assembly would be ultrasonically inspected. In the case of non-magnetic materials, fluid penetrant inspection would supplant magnetic particle inspection; all other steps remaining as above.

Some of the causes of service failures in piping systems are as follows:

- (1) Design notches or stress risers such as attachments of pipe hangers or supports, transition sections, branch connections, dissimilar metal welds, dissimilar wall thickness where heavy-walled vessel nozzles and valve bodies are welded to pipe of lighter wall thickness.
- (2) Unfavorable weld joint designs.
- (3) Graphitization of material next to weld deposits at elevated temperatures for materials such as carbon molybdenum steel.
- (4) Decarburization in weld deposit next to base material.
- (5) Carburization or decarburization at the inner or outer surfaces.
- (6) Mechanical notches such as laminations and those caused by machining or extruding.
- (7) Metallurgical notches at welds in those cases of use of improper electrode filler material.
- (8) Hydrogen embrittlement.
- (9) Stress corrosion cracking.
- (10) Mechanical and thermal fatigue or shock exceeding those for which the system was designed.

The type of failure normally expected in piping and vessels is a longitudinal ductile tear.

For a circumferential brittle failure to occur, it is necessary to have a stress raiser, a stress large enough to cause localized yielding near the stress raiser, and a service temperature below the nil ductility transition temperature (NDT) of the material. Those areas which have the highest probability of developing circumferential cracks are:

- (1) Those subjected to an integrated fast neutron exposure exceeding 10^{18} neutrons/cm² which in the case of carbon and low alloy steels can cause embrittlement.
- (2) Dissimilar metal welds.
- (3) Attachment welds of piping to valves or other components that are subjected to high transient thermal and mechanical stresses and shock.
- (4) Expansion joints.

One mode of operation which has been used to avoid the possibility of brittle failure in reactor vessels is to operate at 20% of the yield strength or less at temperatures below the fracture transition temperature for elastic loading (FTE). The FTE has been taken as the nil ductility transition temperature (NDT) plus 60°F.

3. Criteria for Design

The purpose of the LOFT facility is to simulate severe reactor accidents by intentionally releasing the primary coolant and fission products into the containment shell at a pre-determined time under favorable atmospheric conditions. There appears to be no justification for incorporating design features which are intended to reduce the probability of having less serious accident consequences than provided for the MCA. Therefore, the engineered safeguards adopted in the LOFT facility design are directed at controlling the accident severity of the maximum credible accident and the planned blowdown test (taking into account adverse situations which might develop during the course of the blowdown test).

a. Maximum Credible Accident (MCA)

The MCA is defined as that accident whose consequence in terms of dose to the public are not exceeded by any other known credible accident.

b. Nominal Activity Release Accidents

Accidents less severe than the maximum credible accident may result in the uncontrolled release of airborne radioactive material.

These nominal activity release accidents are less severe accidents which are more probable than the MCA, and yet are not expected to occur more frequently than once per year. Typical accidents which might fall into this category would be a loss of normal power accident and a planned blowdown test during which adverse situations develop.

c. Criteria for Public and Plant Personnel Doses

(1) Normal Operation. For normal operation of the LOFT facility, the permissible doses and maximum permissible concentrations (MPC) for various radionuclides will conform to the values given in 10 CFR 20, "Standards for Protection Against Radiation" for operating personnel and for the public.

(2) Blowdown Tests. The planned nuclear blowdown test under favorable atmospheric conditions (from the standpoint of public dosage) is not considered an accident situation.

The permissible doses and maximum permissible concentrations (MPC) of radionuclides for operating personnel inside the control and equipment building will conform to the values given in AEC-0524-01, Standard for Radiation Protection (August 12, 1963).

If adverse conditions (such as a change in wind direction, meteorological conditions, or steam generator tube failures) occur during the course of the planned blowdown test, this could become an accident situation. As far as operating personnel are concerned, the doses and MPC values should conform to the values in AEC-0524-01. The criteria for the public correspond to that for a nominal activity release accident.

(3) Criteria for Nominal Activity Release Accidents. The criteria for public exposure off the NRTS site are given in the following table:

<u>Part of Body</u>	<u>Exposure to Public (rem)</u>	
	<u>Suitable Sampling</u>	<u>Individual Sampling</u>
Whole body, gonads or bone marrow	0.17	0.5
Thyroid or bone	0.5	1.5

The above criteria apply to a planned blowdown test in which adverse situations develop. These criteria also would be utilized, together with the 20 minute criterion for operator action, to establish the stack monitor setting if high stack activity is used as a signal for automatic containment shell isolation. The activity level for automatic containment shell isolation would be based on a specific radionuclide such as iodine-131. If the activity concentration were just below the isolation level, with the most adverse meteorological conditions, and the operator failed to isolate the containment by remote manual actuation within 20 minutes in response to high stack activity alarms, the resulting public exposure should be less than or equal to 0.5 rem to the thyroid.

(4) Maximum Credible Accident. The public dose criteria given in 10 CFR 100, "Reactor Site Criteria" may be regarded as a guide to the upper limit for public exposures resulting from the maximum credible accident.

d. Design Modifications

In general, the following criteria will be applied in determining whether the consequences of an accident sequence warrant design modifications.

- (1) If an accident is expected to have a probability of occurring no more frequently than once during the lifetime of the facility (20 yrs) and the consequences are less than for the MCA, no design changes are necessary. If the consequences exceed those for the MCA, the MCA may be changed, provided the public exposure doesn't exceed the guides provided by 10 CFR 100. Alternatively, design modifications may be made to reduce the consequences to less than those for the MCA.
- (2) For accidents which are more probable but less serious than the MCA, and the consequences are less than the criteria for nominal release accidents, no design changes are necessary.

For an accident whose consequences exceed that for the nominal release accident, design modifications or changes in operating procedures will be made so that the consequences are reduced to acceptable values.

4. Philosophy

a. Reactor Safety System

The reactor safety system is independent from the process and reactor control systems.

No operator action is required prior to 20 minutes after the accident to prevent the consequences of an accident from exceeding those of the MCA, or in the case of a blowdown test, to prevent the dose criteria from being exceeded in the event of adverse situations which may develop.

Redundant and coincident circuits are used to assure that safety actions are accomplished and to minimize false shutdowns.

The reactor safety system is powered by two power sources. All devices are designed to initiate safety action on loss of power supply or component failure.

b. Containment

The containment consists of the volume within the containment shell including the test area and the basement area. The basement area is separated from the test area by a gas-tight membrane supported by reinforced concrete. The membrane keeps pressure and contamination away from the basement area, thereby permitting access to the basement following a blowdown test.

The containment system will be operated with the heating and ventilation ducts open during normal operation. Personnel occupancy of the test area is not permitted unless the reactor is shut down. Limited occupancy of the basement area is permitted while the reactor is operating and following a blowdown test under health physics supervision. Personnel airlocks, equipped with interlocked doors, are provided into the test area and into the basement area of the test building.

The containment shell penetrations which vent directly from the containment shell interior to the atmosphere surrounding the containment shell are each provided with two isolation valves in series. Those lines which vent directly from the containment shell interior to the environs include the inlet and outlet heating and ventilation ducts to

the test area and basement area, the vent line from the hot waste transfer tank, and the vent line from the decontamination sump. These latter two vent lines vent into the test area with a tee-connection to the stack. The vent line to the stack is provided with two isolation valves in series.

The containment shell penetrations can be categorized as follows:

- (1) Automatic containment isolation is provided for those lines which vent directly to the environs (two valves in series) and for those which provide non-essential services (single valves). The automatic actuation system will provide 1 out of 2 redundancy, but no coincidence. Each of the "automatically" actuated valves is capable of being opened or closed independently by remote operator actuation from the main control room. In addition, double valves are capable of being opened and closed as pairs.
- (2) Those lines which provide essential services are provided with single valves and isolated by remote manual actuation.
- (3) Some containment shell penetrations and service lines fall into the category of being either normally closed or provided with check valves. Some of these normally closed lines such as those for the pressure reduction sprays, decontamination sprays, and their associated services (steam, service water, fire protection water backup, decontamination solution transfer, and demineralized water) are provided with remote manually actuated valves so that these systems or services can be brought into operation following a blowdown test or accident.
- (4) In addition to the automatic containment shell isolation system, single isolation valves are provided in the secondary system feedwater line and in the steam line. These valves are automatically actuated in the event of a secondary system rupture with a sequential related failure of steam generator tubes. The signal employed for the automatic secondary system isolation is not the same as that used for containment shell isolation.

These signals which are used for effecting secondary system isolation are low primary coolant system pressure and high secondary system radioactivity. These secondary system isolation valves are provided with remote manual actuation which can override any automatic signals.

5. Divided Control Room Philosophy

The main control room has controls for items such as the reactor, primary and secondary systems, vehicular gates, radiation monitoring, containment shell isolation and experimental systems control (such as halogen and particulate removal system, pressure reduction sprays, and the decontamination system). There are annunciation and overriding controls for certain critical utilities in the main control room manned by non-union operators.

The experimental instrumentation and data gathering rooms are manned by technical (or non-union) operators.

The union-operated utilities are operated from local control panels with annunciation at a central process annunciation panel in the utility area. The union-operated utilities include air conditioning, heating and ventilation, diesel-generator, service water, and boilers.

There is no division between union and non-union operators in the remote control and sampling rooms.

6. Emergency Power System

The emergency power bus is energized by a diesel engine-driven generator which is run continuously during nuclear operation. A tie breaker between commercial power and the emergency power bus can be closed by manual operator action to re-energize the emergency power bus.

All critical equipment which would be called on to operate following a loss of commercial outside power are connected to the emergency power bus which is energized by the continuously running diesel during nuclear operation. This eliminates the requirement of having to start the diesel, sequence and pick up loads on the emergency power bus following a loss of commercial outside power.

Several loads are connected to the emergency power bus which might have to be started up following a loss of coolant accident or a blowdown test. These are the pressure reduction spray pump, the decontamination pump and the stack exhaust fan.

7. Critical Systems and Equipment

a. General

Equipment and services vital to the safe operation of the facility are duplicated or have an alternate source of supply. Operating philosophy requires the operators to shut down the reactor upon failure of the vital operating equipment item or service.

In general, critical systems and equipment are those whose failure could lead to an accident sequence with consequences exceeding the applicable public dose criteria, or those which must continue in operation for the duration of an accident. A system with a high degree of reliability such that failure of the system when it is needed is incredible is sometimes called a "failure-free" system.

The listing of critical systems has been made consistent with the safety analysis criteria and assumptions. The critical systems are those which must operate or be available to prevent the consequences of an accident sequence from exceeding: (1) those for the MCA (for accidents having an extremely low probability of occurrence, less than once during the facility lifetime), or (2) those for a nominal release accident (for less severe accidents with a higher probability than the MCA, but which are not expected more frequently than once per year). Therefore, those systems and services which must be available during the defined MCA or during a planned blowdown test are considered critical.

The functioning of a critical system depends on everything necessary to make the system operate during the accident conditions; such as electric centers, equipment cooling, power supply, instrument power, service water, and instrument air. The critical systems listed below are protected against steam, water, fire, missiles, and environmental conditions as far as practical.

b. Containment System

The containment system includes the containment shell, air locks, automatic containment shell isolation, and the automatic secondary cooling system isolation. The automatic containment shell isolation valves will fail closed on loss of electric power and/or instrument air.

c. Pressure Reduction Spray System

The spray system includes the pump, cooler, cooling water to the cooler, water for priming the system, filter, control valves, water to pump shaft seals, system headers, and the spray nozzles. Particular attention is given to the possibility of spray nozzles becoming plugged or malfunctioning and to the possibility of the filter becoming clogged. The problem of spray nozzles becoming unavailable is solved by designing the decontamination spray nozzles and piping to meet the performance requirements of the pressure reduction sprays. Filter clogging could be prevented by use of a self-cleaning type filter.

d. Molten Core Container

The molten core container must remain intact following a blowdown test and an MCA release in order to prevent the release of fission products from the molten core material into the containment shell by any route other than the blowdown nozzle or the system rupture. The critical services to the component include the cooling water for the container to maintain its integrity.

e. Emergency Power

Emergency power includes the emergency load center, power cables which supply critical equipment, the diesel engine driven generator, engine combustion air supply, diesel oil circulating pump, diesel fuel transfer pump, diesel starting system, and diesel engine cooling.

f. Control and Equipment Building

Those areas of the control and equipment building which must remain occupied and from which operators are required to perform various functions during a blowdown test or following the MCA must be regarded as critical. This includes the main control room as well as equipment areas containing critical equipment. The operators are provided with lights and a safe breathing air supply. The assured breathing air supply will be provided by filtering the supply air, sealing the building and recirculating, or by a separate bottled air system. Instrument power and/or instrument air is available for control of the critical systems and to put the plant in a safe condition.

g. Reactor Scram or Shutdown System

The reactor safety system is designed to initiate scram or safety action on loss of power supply or component failure. The reactor shutdown system, including control rods, is such that the reactor can be shut down and maintained in a sub-critical condition for any credible accident sequence.

h. Sump Pumps, Critical Equipment Areas

In those areas which are not drained by gravity and which contain critical systems or equipment subject to flooding in the event of ruptures in various water systems, the sump pumps are critical equipment.

i. Service Water

The service water supplies the following critical services:

- (1) cooling for containment spray cooler
- (2) cooling for decontamination "heater"
- (3) diesel engine cooling
- (4) air compressor cooling
- (5) containment spray pump shaft seals

- (6) decontamination pump shaft seals
- (7) water for priming the containment spray system (condensate sump)
- (8) water for priming the decontamination sprays (decontamination sump), since these sprays are used as a backup to the containment pressure reduction sprays.

j. Fire Protection Water

Demineralized water supplies the following critical service:

- (1) diesel engine cooling system fill and makeup
- (2) water for priming decontamination spray system (since the decontamination spray system is used as a backup to the pressure reduction spray)
- (3) control rods seal and cooling, if loss of demineralized water could impair the ability to scram or shut down the reactor.

k. Instrument Power

Instrument power will be available for sufficient time during any credible accident sequence to enable placing the reactor in a safe condition.

l. Instrument Air

All pneumatic control valves fail in a "safe" condition on loss of instrument air. The supply of instrument air and storage capacity allows sufficient time during any credible accident sequence to enable the operator to place the reactor in a safe condition.

m. Protection of Critical Systems

The critical systems and equipment listed above are protected against steam, water, fire, missiles, and other environmental conditions to the extent practical. The task of providing such protection is not a simple one because of the large number of accidents

which are not readily predictable. In general, water and steam lines are kept away from critical equipment, electrical load centers and power cables feeding the critical equipment. The steam and feedwater lines from the steam generator to the fin fans are kept separated from electrical cables and any critical process lines as much as possible. Also, design provisions are made so that steam released by a rupture of these lines can be vented without endangering occupied areas of the control and equipment building. Protection against fire and missiles is provided by separating the electric lines which supply power to the critical equipment from those supplying standby or duplicate equipment. This precludes the possibility that a single missile or fire could disable both pieces of equipment at the same time if their power lines were routed in the same tray.

SECTION IX

PRELIMINARY ANALYSIS OF REACTOR OPERATION

IX. PRELIMINARY ANALYSIS OF REACTOR OPERATION

A. Basic Operating Philosophy

The LOFT experimental program includes operating the LOFT reactor at power for a relatively short time. Safe and efficient reactor operation demands a thorough technical understanding of reactor behavior and strict adherence to written operating limits and procedures. In addition, detailed operator training, careful supervision, and the surveillance of a safeguard review committee comprise the basic operating philosophy for the LOFT reactor facility.

Operator training will be achieved through experience gained on operating reactors, studying design specifications, blue prints, and operating manuals for the reactor and plant facility, and by aiding in the preparation of the operating and test procedures for the LOFT reactor. Further training will be attained during checkout of the facility and during the initial critical loading and physics measurements on the reactor.

Operating manuals for the facility will be prepared by both the architect-engineer and by STEP Project. Operating procedures and limits will also be prepared which, in conjunction with the STEP Standard Practices Manual, will provide a detailed guide for all phases of reactor and plant operations. In addition to the manuals mentioned above, prior to any operation of the reactor a Test Series Proposal (TSP) will be prepared, outlining in detail the experiment to be performed, what the expected results of the experiment are, and what nuclear or radiation hazards are involved. This TSP must be approved by the STEP senior staff, the SPERT-STEP Safeguard Committee, the Assistant Manager, Technical, and by the Idaho Operations Office of the USAEC prior to performing the experiment. All rules, procedures, and test parameters outlined in these documents will be strictly followed, and all personnel are required to be familiar with them.

B. General Safety Considerations

Due to the preliminary status of the LOFT reactor and facility design, this analysis of the power operation phase of the test program

must of necessity be rather cursory. During the detailed design of the LOFT reactor and facility, however, an extensive analysis will be carried out to evaluate those factors which could affect the integrity or safe operation of the plant. Due to the relatively short (400 hrs) reactor operating time, mechanical deterioration or radiation induced effects are not of primary concern in the analysis. However, because of the conduct of multiple blowdowns prior to nuclear operation, the detailed design must be analyzed carefully to assure that the primary system or containment integrity is not compromised by these tests. This analytical effort, then, will be done to assure that the entire experimental program can be conducted without undue hazard to operating and other site personnel or to the general public.

In this preliminary safety analysis of the LOFT plant, the major effort has been directed toward the loss of coolant test and the postulated maximum credible accident. Certain other accidents have been considered for the time of reactor operation which, because the LOFT design is similar to the general class of pressurized water, low enrichment, UO_2 -fueled reactors, are generally the same as those presented for most reactors in this class. These accidents can be generally classed as reactivity accidents and mechanical accidents. These two types of accidents are discussed further in this section. In addition the general hazards of operation are discussed as they may relate to the two classes of accidents.

C. General Hazards

Errors made by test or experimental personnel could involve incorrect operation of process or heat removal equipment, incorrect setting of alarm or set points on safety instrumentation or incorrect movement of control rods. Because of the inherent potential hazards involved, administrative procedures and controls as discussed above are required to minimize the possibility of unplanned nuclear incidents. In addition, many safety features and protective devices are provided to protect against accidents which could occur as a result of operator error. Those errors that could cause serious nuclear incidents are discussed in connection with the major accidents postulated in this section.

The reactor possesses inherent safety and control characteristics and the safety of plant operation is further enhanced by the specific inclusion of many devices designed for this purpose. Single equipment failures or sequentially related failures which violate the integrity of the fuel cladding or primary coolant system are of major concern in regard to the direct release of radioactive products to the containment volume. These include: loss of flow, steam line break, primary system component failure, steam generator failure, and unplanned rod withdrawal.

The potential failure of equipment will be minimized through careful design and analysis and through the selection and preoperational inspection of equipment. The use of multiple instrumentation and redundant circuits which are designed to fail-safe will be employed to minimize these failures.

Storms could conceivably cause a loss of commercial power to the facility. The loss of power from the diesel generator is not conceivable, however, as a result of storms. Critical systems such as reactor instrumentation, decay heat removal equipment, and containment service equipment are all powered from the diesel generator. The loss of commercial power situation is discussed in regards to the loss of flow accident.

The design of the containment building will be such as to withstand the effects of wind loading. The probability of floods that could conceivably cause a nuclear incident is low due to the scanty rainfall at the NRTS and the good drainage on the site.

The NRTS regional classification of Zone 2 on the U. S. Uniform Building Code Map of Seismic Probability implies that the NRTS is reasonably free from severe earthquakes. The August, 1959, earthquake in the Madison River Canyon did not change this classification and did not result in any damage to reactor structures at the NRTS. The LOFT design is based on AEC-ID structural and architectural standards which conform with the Uniform Building Code as far as protection against earthquake is concerned. Due to the very low frequency of earthquakes, the design of the LOFT plant, and the short reactor operating time, earthquakes are not felt to be a credible hazard.

D. Reactivity Accidents

1. Startup Accident

The startup accident is defined as the continuous, uninterrupted withdrawal of control rods at the maximum rate. The reactor is assumed to be in a shutdown condition with the neutron flux in the startup or source range and the initial reactor temperature at the value which results in the maximum reactivity insertion rate. To precipitate such an accident requires that the operator and control room supervisor disregard startup and operating instructions and ignore all startup period alarms. In this event, the accident will, of course, be terminated safely by intermediate period or high power level scram of all rods.

In order for rod withdrawal to proceed beyond this point in an uncontrolled manner, it is necessary to further postulate the failure of automatic safeties provided in the control system. These include, sequentially, the period scrams in the intermediate range nuclear instrumentation, the level scrams in the power range nuclear and flux-to-flow comparator instrumentation, and pressure scrams in the primary coolant instrumentation. Simultaneous operational failure of all these automatic safeguards is considered incredible because of their reliability and complete channel duplication.

The analysis of this accident situation has been carried out for many power reactors similar to LOFT, i.e., pressurized water reactors using low-enriched uranium dioxide fuel. Due to the strong prompt negative Doppler coefficient that this type of reactor exhibits, the power transient is expected to be terminated nondestructively for conceivable reactivity changes even though all protective measures incorporated in the control system and operating procedures may fail.

2. Control Rod Withdrawal at Power

The initiation of continuous uninterrupted control rod withdrawal when the reactor is operating in the power range could only result from a serious malfunction of several automatic control system components or from disregard of the reactor operating instructions. As in the

startup accident, the protective instrumentation will be designed to terminate the power transient by scram before departure from nucleate boiling (DNB).

In order for rod withdrawal to proceed, the three independent power channels must fail. In addition the low-flow scram from the flux-minus-flow comparator and high-pressure scram circuits must also fail. Because of complete channel duplication the reactor is protected against damage in the event of continuous rod withdrawal.

3. Control Rod Drop Accident

The control rod drop accident is initiated by a control rod becoming unlatched and dropping into the scrambled position. The core power would immediately drop to a minimum value and would then subsequently rise to a new equilibrium position somewhat below the original power level. The dead-band T_{avg} controller would then attempt to adjust reactor power by rod withdrawal to the original level. A highly non-symmetrical power generation could result with fuel temperatures rising significantly in some regions of the core.

4. Cold Water Accident

A decrease in the primary coolant inlet temperature results in an increase in reactivity due to the rather large negative temperature coefficient of reactivity. This particular accident is primarily of concern in reactor plants having multiple primary coolant loops. Since the LOFT reactor system has only one coolant loop the initiation of a cold water accident during power operation is not credible. The primary coolant makeup pump is capable of delivering 5 gpm at full primary coolant pressure and at a minimum temperature of 350°F. The rate of reactivity change due to operation of this pump is far less than that capable of being controlled by rod motion and is therefore not of concern as regards to the cold water accident.

Plant operating procedures require that the primary and secondary coolant systems be brought up to design temperatures, pressures and flows with non-nuclear heat before the control rods are withdrawn. Control interlocks which insure these procedures are followed within safe limits are itemized below:

- (1) Proper operation of the two startup range nuclear channels at the core source level is assured by a rod withdraw prohibit interlock below 10 counts per second on both startup range instruments.
- (2) Proper overlap between startup and intermediate range instruments is assured by a rod withdraw prohibit interlock operated from the two intermediate range channels.
- (3) Control rod withdrawal is prohibited at any nuclear power levels above one kw thermal unless safe plant conditions have been established to satisfy the following interlocks:
 - (a) Design loop flow is established by redundant measurement of core ΔP .
 - (b) Near design core temperature and pressure is established by coolant T_{avg} measurement.

The emergency decay heat removal pump is also required to run during all plant operations as backup for the above described procedures and interlocks.

During startup conditions with no loop flow with the reactor coolant at 60°F, the core heats up at approximately 2°F per hour for 1 kw nuclear power. If the main coolant pump is started up after one hour, the core will experience a step of about 2°F of colder water from the pump. If nuclear operation at or near 1 kw is continued for periods longer than one hour before starting a loop pump, the step change in temperature experienced by the core will be slightly higher. However, the temperature difference is limited to very nearly 2°F for 1 kw nuclear operation by convective cooling of the core.

The severity of a cold water accident during critical tests is inherently limited by void formation in the coolant and the negative temperature coefficient as the average core temperature increases. In addition, any possible power excursion would be limited by the combined effects of the intermediate range period scram and the flux-minus-flow overpower scram.

E. Mechanical Failure

1. Fuel Cladding Failure

In the event of a fuel cladding failure, the condensible fission products released to the primary coolant are removed in the coolant purification system. A portion of the primary coolant (70 gpm) is passed through ion exchange demineralizer columns which will reduce the activity in the primary coolant system following a fission break by a factor of 100 in approximately six hours.

Non-condensable gases will collect in the pressurizer above the operating power level. They are then vented to the quench tank and finally to the stack under favorable meteorological conditions.

2. Loss of Coolant Flow Accident

In the LOFT reactor, the coolant is circulated through the reactor and primary coolant loop by a single 15,000 gpm pump. A loss of coolant flow accident would result from the loss of power to, or seizure of, the main coolant pump. Seizure of the pump presents the most drastic situation in that flow coast-down is much more rapid due to the loss of pump inertia.

When a loss of flow occurs, the decreased flow rate will automatically cause the reactor power to be cut back to a safe level. Since the emergency decay heat removal pump is continuously operating with 800 gpm, coolant flow is immediately available to the core upon loss of the primary coolant pump.

A complete loss of commercial power in the plant which results in a loss of flow also automatically causes scram of all rods. The emergency decay heat removal pump, however, is powered from the independent diesel power source and would, therefore, continue to operate.

3. Secondary Coolant Steam Line Break Accident

A break in the steam line between the steam generator and the air-cooled condensers would cause an immediate decrease in secondary system pressure. As a result, higher steam flow rates or back flow

would exist through the steam generator for a short period. The increased heat removal would subsequently lower the temperature of the primary coolant system and the automatic control system (T_{avg} dead-band controller) would thereby attempt to restore the average coolant temperature to its initial value, if the temperature decrease is sufficient to change the average value out of the dead-band zone. In addition, the negative temperature coefficient of the moderator would add reactivity to the system. If the reactor power increases beyond the flux-minus-flow comparator setpoints, the reactor is then scrammed.

The secondary heat dissipation system employed in the LOFT plant consists of a steam generating and condensing loop requiring the use of a pressurized closed loop piping system. In the event of a cold leg or condensate piping rupture, the coolant would then flash and consequently steam flow past the generator may be reduced sharply. The primary coolant temperatures would then rise with a corresponding decrease in reactor power from the negative temperature coefficient and rod insertion would be actuated by the automatic control system.

Operational procedures, in the event of a break in the steam line or cold leg, will require that the operator initiate reactor shutdown and isolate the steam break. Isolation valves are provided in the feedwater line and steam line. These valves will be automatically actuated in the event of a rupture to isolate that portion of the system within the containment. Shutdown decay heat can then be dissipated through the nonregenerative heat exchanger in the purification system loop.

4. Steam Generator Tube Rupture Accident

The rupture of a tube in the steam generator could result in the venting of primary coolant directly to the containment atmosphere. The relief valve on the secondary side of the steam generator is vented to the containment building and is set to actuate at approximately 1100 psig. Upon rupture of the steam generator tube the primary system would undergo depressurization and the secondary system would correspondingly overpressure, causing an actuation of the relief valve. The primary coolant would then be expelled to the containment until the pressure was

reduced below 1100 psig. The automatic protection system on primary coolant pressure and flow rate would initiate scram, thereby reducing reactor power during the blowdown period. All isolation valves, including those on the secondary system, would automatically close and the operator would initiate plant shutdown.

Once the relief valve has closed, the primary coolant would then be circulated by means of the emergency decay heat removal pump and heat removal effected by flow through the nonregenerative heat exchanger. Failure of the relief valve to reclose, although highly improbable, would then constitute a loss of coolant accident since the primary coolant could then be completely discharged to the containment volume.

5. Loss of Coolant Accident

A loss of coolant accident may occur as a result of a rupture of the primary coolant system or connecting piping lines, such as in the purification loop or makeup system. A large rupture would result in loss of system pressure, rapid expulsion of coolant to the containment, followed by heatup of the core and eventual clad melting.

The probability of a severe rupture can be greatly reduced by careful design, fabrication, inspection, and installation of the piping systems. In addition, the relatively short time of power operation planned for this reactor further reduces the probability of such an accident. However, the consideration of this accident and its consequences leads to the hypothesis of the Maximum Credible Accident. This accident is considered separately in Section XII, including the analysis of the accident and the radiological hazards attendant to it. The discussion of the coolant blowdown and core meltdown is discussed in Section X.

F. Radiological Hazards During Normal Operation

Major sources of gamma radiation during normal reactor operation are:

- (1) Core gammas (both fission and capture gammas in the core).

- (2) Capture gammas in the core support barrel, thermal shield, reactor pressure vessel, and in the reactor shield tank including the lead shield.
- (3) Decay gammas from nitrogen-16 in the primary coolant system.

Neutron leakage from the shield tank and the top and bottom head of the pressure vessel is sufficiently low so that activation of components in the containment shell is no problem. The thickness of the reactor shield tank is such that thermal neutron flux at the outside surface of the tank is less than 10^5 n/cm²-sec.

As described in Section VII, all areas which require continuous access during reactor operation either are located underground or are shielded locally, so that the radiation level does not exceed 1.0 mrem/hr. The areas which require limited access during operation such as the remote control room and the basement of the containment building are shielded so that the radiation level does not exceed 10 mrem/hr.

The biological shield which surrounds the containment building reduces the radiation level at a distance of 100 m from the containment shell centerline to less than 20 mrem/hr.

Figure IX-1 shows the dose rate as a function of distance from the containment shell centerline. The total dose rate includes both skyshine and direct radiation through the shield. Based on a ten-meter high biological shield and assuming the highest source of radiation to be seven meters above the ground floor of the containment building, the skyshine contribution peaks at approximately 50 m from the containment shell centerline, but does not exceed 1 mrem/hr. The skyshine contribution can be conservatively approximated by an inverse distance law for those points located farther than 100 m.

As seen from Figure IX-1, the major contribution to the radiation level 100 m away from the containment shell is the direct radiation through the biological shields. Skyshine contributes less than 15% of the total dose rate. At a distance approximately 500 m, both direct and skyshine contribution become approximately equal in magnitude, but the total dose rate is less than 0.3 mrem/hr. At distances greater than 500 m, the scattered gamma rays predominate. The dose rate at a distance of one mile is less than 0.05 mrem/hr.

The disposal of radioactive wastes is discussed in conjunction with the facility systems for accommodating this in Section VII. The venting of radioactive gases to the atmosphere will only be done under favorable meteorological conditions in which the wind is blowing in the on-site direction.

Fig. IX-1 - Direct Radiation from the Containment Vessel During Operation

SECTION X

ENGINEERING ANALYSIS OF THE LOSS OF COOLANT TEST

X. ENGINEERING ANALYSIS OF THE LOSS OF COOLANT TEST

A. General

The final phase of the LOFT experimental program consists of simulating a primary coolant pipe rupture allowing the core to melt and fission products to escape. To conduct such tests safely, heavy emphasis is placed on containment design, verification of containment leak tightness, monitoring to determine the magnitude of the fission products released, - and in the event of an emergency - a means of terminating the experiment. The test program described in Section II includes those tests needed to evaluate each of the items on which safety is so heavily dependent. Within this framework, an extensive analytical effort has been performed to investigate the parameters affecting the potential hazards which could exist during the conduct of the loss of coolant test. These analyses includes a study of the coolant blowdown characteristics, core meltdown, fission product behavior, and containment leakage. The results of these analyses are thus applied to the analytical prediction of the radiological hazards from released fission products to on-site personnel and the general public in Section XI.

A review of these analyses, results obtained, and the general assumptions which the radiological hazards for the test conditions are based are discussed below. In addition, a review of the tests to be performed on the engineered safety devices, general test sequence, administrative procedures, and emergency procedures are also presented.

B. Pre-Test Checkouts

As outlined in the experimental program (Phase I and II), a number of tests will be performed prior to the core meltdown to evaluate those systems and facilities that may tend to influence the overall safety of the test. These tests will include a checkout and evaluation of:

- (1) containment leakage, (2) pressure reduction sprays, (3) exhaust filter system efficiency, (4) techniques to be used for measuring the fission product concentration in the containment vessel, and (5) control room ventilation system.

A brief review of each test and the checkout to be performed on each system or facility prior to the final loss of coolant test are presented below.

1. Containment Leakage

During the initial test program, measurements will be made to determine the leakage rate from the containment vessel at design pressure. These tests will be performed prior to and following the non-nuclear blowdowns to determine the effects of the blowdown forces and the environment (temperature and humidity) on the leakage characteristics. It appears reasonable to assume that if the leakage remains constant following a series of non-nuclear blowdowns it will not significantly change following the blowdown involving core meltdown.

It is recognized that the major leaks in the containment vessel originate at penetrations and at seals around access doors. Thus, to reduce or eliminate the leakage directly to the atmosphere, consideration is being given to monitored penetrations with the provisions for filtering the gas before it is released through the exhaust stack. With this arrangement and with the design considerations given to the large railroad doors, it is reasonable to assume that the halogen and particulate leakage to the atmosphere will be relatively small.

A final leakage test at test pressure will be performed after the containment vessel has been sealed in preparation for the extended power run.

2. Pressure Reduction Sprays

The pressure reduction spray system will be turned on during several of the non-nuclear blowdowns to verify the time required to reduce the internal pressure to less than 1 psig. Following at least one of these blowdowns, iodine containing trace quantities of radioiodine will be released to provide information on the effectiveness of the spray for removing iodine from the atmosphere. These tests plus a complete system performance evaluation immediately prior to the power run will provide reasonable assurance that the leakage from the containment can be terminated in the event of an accident.

3. Exhaust Filter System

Following the installation of the filter system and again immediately prior to the power run, the system will be examined for damage or for leakage paths around the filters. In addition to these tests, several tests will be performed during the non-nuclear blowdown to verify the effectiveness of the demister for removing moisture and the effectiveness of the overall filter bank for iodine removal.

4. Fission Product Sampling System

All techniques to be employed for determining the fission product concentration in the containment vessel as a function of time will be checked out and proved to be operating satisfactorily prior to the start of the power run.

5. Services Building Ventilation

To guard against possible contamination of the internal atmosphere of the service building, it is contemplated that the air intake will be turned off during all periods when a possibility of airborne activity exists. Thus, prior to the actual core meltdown tests, the ventilation system will be turned off for a period of five days to demonstrate that the atmospheric conditions that may prevail in the building will not affect the normal operation of the facility.

C. General Test Sequence

The simulated pipe rupture will be initiated as soon as the acceptable meteorological conditions can be attained following the 400 hours of operation at full power. Within approximately 10 minutes after the rupture, the stainless steel clad surrounding the fuel will start to melt and fission products will be released to the reactor vessel environment. Very little experimental data are available with which to accurately predict the fission product behavior during this phase. However, it is postulated that as the clad melts and the temperature inside the vessel

increases, a large fraction of the released fission products could be transported to the containment vessel by the steam or by diffusion due to the gas expansion in the reactor vessel. After reaching the containment vessel a certain fraction of the isotopes, particularly iodine, will plateout on the exposed surfaces, be trapped in the water particles, or remain suspended in the atmosphere and be available for leakage from the containment vessel.

Because of the uncertainty involved in predicting the fission product behavior, extensive radiological measurements will be made during the entire test to assure the safety of both on-site personnel and the general public. As previously stated, the test will be initiated during lapse conditions with the wind blowing to the on-site direction and reasonable assurance will be obtained that these conditions will prevail for approximately six hours. Throughout this time, samples of the containment atmosphere and the atmosphere downwind of the vessel will be collected and analyzed to assure that in the event of a change in the weather conditions a hazard will not prevail. The samples from inside the building will be collected by the removable sampling devices discussed in Section VII and from one of the gas sampling lines that penetrates the vessel. Samples external to the building will be obtained with mobile surveillance vehicles equipped with high volume air samplers. These vehicles will enter the downwind area and commence taking samples approximately 10 minutes after the rupture. By this technique, sufficient data will be obtained and analyzed within two hours to determine if a hazard is likely to exist. If it is concluded that a hazard would exist, the leakage from the building will be terminated by the procedure outlined in Section X-J. A more detailed discussion of the postulated fission product behavior following clad melting is presented in Sections X-E and X-F.

During the first few minutes following blowdown, the pressure-time history inside the containment vessel will be followed very closely to detect if any major leakage paths develop during the blowdown. Prior to initiation of the test, a series of pressure curves based on the non-nuclear blowdown and theoretical predictions will be compiled for comparison with the actual pressure decay.

At the end of the test, the pressure reduction spray system will be activated to reduce the internal pressure of the building to atmospheric pressure. The remaining fission products, except for noble gases, will then be reduced by recirculating the containment atmosphere through the exhaust filter system. After the fission products have been reduced to a safe level, they will then be exhausted through the stack. The release will occur during controlled meteorological conditions with the wind blowing in the on-site direction. Samples of the containment atmosphere will be obtained immediately prior to the release to assure that an excessive quantity of fission products are not being released.

Prior to gaining entry to the building for the purpose of removing the test package to the hot shop, the valve between the blowdown nozzles and the primary coolant system will be closed and the internal surfaces of the building and the external surfaces of the test package will be remotely decontaminated to preclude the spread of fission products when the railroad doors are opened. Personnel access to the building will be controlled by the Health Physicist in charge.

D. Administrative Controls

Administrative controls and procedures are required throughout the test program to insure the safety of STEP and other NRTS personnel and to minimize the hazards to the general public in the event of an accident or during the planned test. These procedures and controls will be reviewed and approved by the SPERT-STEP Safeguards Committee and the AEC prior to reactor operation. They will also be contained in the STEP Standard Practices Manual and all personnel associated with the test will be familiar with them.

Prior to the loss of coolant test, the entire reactor area, except for the shielded control room, will be cleared of personnel for a distance of at least one mile. The downwind direction will be cleared for several miles except for the mobile surveillance group who will enter the downwind area approximately 10 minutes after the blowdown to collect samples of the released fission products. This group, however, will not enter areas

where the direct radiation from the containment building or airborne fission products exceeds the levels to be specified by Phillips Petroleum Company Health and Safety Group.

To safeguard personnel during testing, the test will be conducted in accordance with the Test Series Proposals which have been reviewed and approved by the SPERT-STEP Safeguard Committee and the AEC. All rules, procedures, and test parameters outlined in these documents will be strictly followed and all personnel participating in the test are required to be familiar with them. All planned tests in which a significant quantity of radioactive fission products may be released, including the final loss of coolant test, will be initiated under controlled meteorological conditions with the wind blowing in the on-site direction.

Although protection of the reactor and plant equipment is secondary to safeguard of personnel, it is still of major importance. Thus, appropriate operating procedures and controls will be included in the test procedures.

Pre-test check lists will be completed by personnel participating in the test to insure that instrumentation and equipment are operating correctly and that the plant is in readiness for the test.

E. Coolant Blowdown

1. Introduction

The period of coolant blowdown is considered to commence at the instant that the primary system is opened, and lasts until a pressure equilibrium is attained between the reactor vessel and the containment building atmosphere. During the blowdown process, there may exist a potential for damage to various system components, due to the rather severe temperature and pressure transients that are expected to occur. The safety analysis of these transient effects on the LOFT system requires a fairly complete knowledge of the blowdown phenomena. For this reason, and because the LOFT program represents a sizeable and important experimental effort, it is imperative that the severity of the various phases of the loss of coolant test be accounted for apriori.

An analysis is underway to determine the dynamic response of the system components during the loss of coolant. In addition, scale model experiments may be necessary to study the dynamic stress and behavior of critical components. This work is expected to provide the information necessary to assure that the system will indeed allow the experiment to be carried out safely and as planned. Also, the 20 non-nuclear blowdowns for which the system is adequately designed will be run prior to the actual loss of coolant test. Since these tests will be heavily instrumented for measuring pressure, temperature, strain, etc., a good picture of the response to be expected in the final test will be established. Some of the potentially damaging effects of the coolant blowdown on the primary system, the reactor dolly, and the containment building are discussed in this section. The results of some of the preliminary investigations are included.

2. Effects of Coolant Blowdown on the Primary System Components

a. Introduction

In both the design of the reactor system and the selection of test parameters for simulating a loss of coolant accident, consideration must be given to two things. These are: (1) the influence of blowdown on the mechanical integrity of the core and primary coolant system components, and (2) the influence of the design and test parameters on the transport of fission products. In analyzing the effects of blowdown, it has been determined that the time required to rupture a primary coolant pipe is of considerable importance in determining the hydrodynamic forces which exist. ~~Insufficient experimental information exists at present regarding~~ this time, however, the assumption that the rupture will take place instantaneously is completely unrealistic.

In the case of a truly instantaneous rupture, the fluid at the rupture would suddenly "see" the pressure decrease from the system pressure to an ambient pressure condition. This decompression would then propagate as a rarefaction wave through the main piping system, the heat exchanger, the pressurizer, and into the reactor vessel at the acoustic velocity of the fluid. During its passage through the system, the magnitude of the wave would decrease continually by energy loss effects such as friction, area changes, bends, etc. In those localities where

the pressure drops below the saturation pressure, rapid steam formation will immediately restore the pressure to saturation conditions. Depending on the system geometry and coolant condition, reflection of the rarefaction wave at the various boundaries of the system may tend to reinforce or to diminish the effect of the original disturbance. Although a detailed knowledge of the transient effects of an instantaneous break is not presently available, it is reasonable to expect that pressure forces approach a maximum.

The range of rupture times for which damage occurs from the sudden decompression is not well defined. However, for rupture times which are long compared to the time of travel of a rarefaction wave throughout the system, the acoustic, or wave, phenomenon does not apply. Even though the rupture time may be long enough to avoid the rarefaction wave induced force, there may be severe pressure forces due to the rapid acceleration of coolant during the first few instants of blowdown. These "standard" hydraulic pressure drops are unavoidable, and their maximum values will also be dependent on rupture time.

In the discussion which follows an attempt is made to define the major areas regarding the safety of the primary system components during the coolant blowdown portion of the loss of coolant experiment. The system response can tentatively be split into two types - the local pressure changes occurring within a fraction of a second after the start of system rupture, and the more long term (several seconds) thermal and pressure transients lasting for at least the duration of blowdown. Various system components can be expected to respond differently to the additive effects of related transient blowdown phenomena.

b. Reactor Vessel and Piping

The rapid coolant expulsion during the blowdown will cause a large decrease in the system temperature and pressure in a very short time. This could, of course, produce rather severe temperature gradients in the walls of the vessel and piping, with high thermal stresses resulting.

Some indication of the gross changes in coolant pressure and relative mass within the vessel during the blowdown transient can be obtained from Figures X-1 and -2. These figures are not precisely

representative of the blowdown transient because many simplifying assumptions and techniques were used in the calculations, such as homogeneous expansion in the vessel.

Although the previously mentioned figures give an overall picture of the approximate severity and duration of the transient, they are not sufficient for use in predicting the local heat transfer coefficients in the system, which are needed in the thermal stress analysis. This is one reason why an extensive analytical and possibly experimental effort is necessary to obtain a more detailed and exact prediction of the coolant state and behavior during blowdown.

There is no possibility of damage to the system from the reaction forces produced by the changing momentum of the fluid as it travels toward the break. These forces will be accounted for in the design of the supporting structure for the primary system. In addition, the non-nuclear blowdowns will verify the mechanical integrity of the system.

c. Rupture Tee

As presently conceived, the rupture tee will be sealed off from the rest of the primary coolant loop, during normal operation, by means of a quick-opening valve in the line immediately preceding the tee. Rupture of the system is then simulated by rapidly opening the valve or by breaking rupture disks at the ends of the tee. To assure that when the break is started the sudden impingement of hot water on the tee does not cause failure, great emphasis will be placed on determining the sufficiency of its design by considering the mechanism and thermal stresses acting on it. Preliminary stress analysis of the rupture design indicates that the tee will withstand the blowdown forces to be expected during the loss of coolant test. The data obtained during the non-nuclear blowdowns will also verify the structural adequacy of the tee.

d. Flow Distribution Baffle

The flow distribution baffle at the reactor vessel inlet is specifically designed to withstand forces in excess of those which can credibly be expected to act during blowdown. Thus, even though it may represent one of the major flow resistances in the primary system, failure of this component does not appear to be a conceivable consequence of the loss of coolant experiment.

e. Reactor Core and Core Supports

The preliminary investigations into determining the pressure drop across the core during the loss of coolant has not yielded results of sufficient accuracy to completely assure whether a given core and supporting structure would remain intact. It has been indicated, however, that large blowdown forces could occur. It is presently unknown whether a system designed for conventional reactor operation is adequate, since it is not specifically built to take the large pressure differences associated with blowdown. For this reason, two types of supports are being designed for LOFT. One is a conventional system, and the other is to be specially built to withstand any blowdown force that could reasonably be predicted to occur. Further, an extensive analytical study is being undertaken partly for the purpose of obtaining the information needed in evaluating the blowdown forces acting on the core. In the preliminary, non-nuclear blowdown experiments, the heavy-duty support structure will be used. These experiments will then give evidence of what pressure forces may be expected in the actual loss of coolant test. At that time, on the basis of the detailed analytical studies and the data from the previous blowdown tests, a decision will be made as to which support structure is employed for the final test.

Some preliminary studies have been made to determine the range of maximum core pressure drop which may be expected. In general, the peak core pressure drop is dependent on the initial state of the coolant, the length of pipe and the restrictions between the vessel and the break, the location (elevation) of the break, the maximum

break area, and the time elapsed in developing the break. A simplified analytical model was employed which is based on the following assumptions: an isothermal system, compressible fluid in the vessel but incompressible flow through the pipes, constant flow resistance coefficients, and a break in the inlet piping (cold leg). The continuity and force balance equations for the system were solved by means of an analog computer. Only the decompression to saturation pressure was analyzed and no attempt was made to describe the acoustic effects of a rarefaction wave, even though an instantaneous break was generally assumed. Some of the results are listed in Table X-A, and curves obtained from a typical run are shown in Figure X-3. The nominal run is taken to be the case which most nearly represents the LOFT geometry, resistances, etc. Since it is not completely clear whether or not critical flow would exist during the subcooled blowdown, two cases were run. In the first, the exit pressure was held at atmospheric. In the second, the exit pressure was assumed to drop to atmospheric only until the flow rate solved for by the analog computer was equal to a critical flow rate based on an exit pressure of 0.75 of the saturation pressure, after which the exit pressure was held at 0.75 of the saturation pressure for the duration of the subcooled blowdown. While neither assumption for the exit pressure is necessarily correct, the comparison presented is helpful. The latter assumption for the exit pressure is thought to be more nearly correct.

Independent of this model, the pressure difference across the core as produced by a sonic decompression wave resulting from an instantaneous break has been estimated. With the assumption of a top break, negligible hydraulic losses, and an isothermal system, it was found that a pressure drop of about 500 psi could occur across the core with a duration time of about 2 milliseconds.

The studies completed to date have indicated that the rupture time is a very significant factor in determining the maximum core pressure differential. However, little applicable data on the failure of large pipes carrying hot, high pressure water, particularly with regard to rupture time, have been found. It is currently felt that

DECOMPRESSION PRESSURES

<u>Run Description</u>	<u>Maximum Core Pressure Drop, psi</u>
Nominal LOFT Design	345
Nominal LOFT Design - With Critical Flow	275
Nominal LOFT Design - Exit Pressure Stays at 0.75 p _{sat}	210
Nominal LOFT Design - Rupture Time 20 msec	300
Nominal LOFT Design - Rupture Time 20 msec With Critical Flow	250
Nominal LOFT Design - Rupture Time 50 msec	220
Nominal LOFT Design - Rupture Time 50 msec With Critical Flow	205
Core Resistance Halved	185
Core Resistance Doubled	600
Exit Pipe (length/area) Changed by $\pm 20\%$	345 $\pm$ 25
Cold Leg (resistance/area) Changed by $\pm 20\%$	345 $\pm$ 25

the time between the start of crack propagation and the establishment of a fully developed breach of the system is apt to be in the 50 to 100 msec range for most failures with a small probability that a 5 - 10 msec rupture time could occur in some cases. The LOFT design is fairly flexible in this respect, since by using a "quick-opening" valve a 50 msec rupture time can be attained and by using a rupture disc method an order of magnitude faster break time is possible. It appears feasible to incorporate either of these rupture mechanisms into the LOFT test package. Thus, when a more extensive investigation is completed and the conclusions are reached, a decision will be made as to which rupture mechanism to employ. It can be concluded the transient effects of the blowdown during the loss of coolant test will be reasonably well-established prior to the test, both by previous detailed analysis and by actual data obtained during the non-nuclear blowdowns. For this reason, the safety of the system can be assured.

f. Control Rods and Drives

The control rods will experience about the same transient pressure forces as does the reactor core. For the planned loss of coolant test, the rods will not be scrammed, but will remain connected to the control rod drives. Each drive will be designed to hold the rod in position against an upward or downward force of 2500 pounds, corresponding to a pressure difference of approximately 350 psi. It is considered that this criterion is adequate with respect to blowdown forces which may credibly occur.

g. Steam Generator

Results of the detailed analysis of the expected blowdown forces will be factored into the design of the steam generator. For this reason, a failure of the steam generator during blowdown is not considered likely. Since the steam generator is located in the line between the vessel and the break having the higher flow resistance, the pressure forces should not be too severe, except possibly during the initial decompression.

h. Pressurizer

Because the pressurizer is joined to the main coolant piping by a 4 in. line, the transient pressure and temperature changes it undergoes will be comparatively less severe than that experienced by the reactor vessel. For this reason, no serious consequences are expected from the blowdown to this system component.

i. Primary Pump

The effect that the multiple coolant blowdowns to be conducted in LOFT would have on the primary pump is being given much consideration. In the event that discharging through the pumps is not feasible, a pump by-pass can be incorporated to avoid any damage. For the final test, of course, the pump will be in-line.

3. Effects of Coolant Blowdown on the Reactor Dolly

The large thrust forces produced by the discharging coolant have been accounted for in the LOFT design. The coolant will issue symmetrically from both ends of a tee placed at the end of the blowdown pipe. Thus, the forces produced by the fluid discharging equally to either side of the dolly will effectively be cancelled. Since no other consequential blowdown forces will act on the reactor dolly, it is concluded that the loss of coolant test will not impair the safety of this system.

4. Effects of Coolant Blowdown on the Containment Building

Several possible results of the coolant blowdown must be considered in assessing the containment building response. Among these are: (a) possibility and magnitude of a shock loading due to the sudden release of coolant, (b) possibility of missiles penetrating the shell, (c) pressure rise in the building due to the release of coolant, (d) temperature changes in the containment wall, and (e) the possibility and effect of coolant impinging on the wall of the containment building. A brief discussion of each follows:

a. Shock Loading

The worst case from the standpoint of shock wave generation in the containment is that in which the primary system is ruptured nearly instantaneously. Even in this case, however, the shock effect is negligible. This is largely due to the fact that the impedance of air is very much lower than that of water. For this reason, the pressure pulse which propagates into the containment is found to be on the order of one psi. Geometric spreading further attenuates this pulse. Therefore, the shock loading on the containment and any equipment within it represents a negligible factor in this safety evaluation.

b. Missile Damage

It is considered highly unlikely that the primary system will fragment as a result of the loss of coolant test to

produce missiles which could penetrate the containment barrier. The primary system is specifically designed to allow the conduct of blowdown tests and the tests run during the non-nuclear blowdowns will verify the adequacy of this design. As a backup, a thick missile barrier of reinforced concrete will be provided on the inside surface of the shell. In the very unlikely event of missile penetration, the spray system can be activated to reduce the pressure in the building, thus decreasing the potential for fission product transport to the outside atmosphere. The generation of a missile and its penetration of the containment barrier is considered an accident situation and as such is discussed in Section XII, the analysis of the Maximum Credible Accident. For the purpose of the analysis which covers the loss of coolant test the penetration of the containment by a missile is not considered.

c. Pressure Rise

The design pressure for the LOFT containment building is 40 psig. However, by the end of the blowdown period the pressure will be only 24 psig. Even in the event of a secondary system rupture the pressure will not exceed the design pressure. As noted earlier, a pressure reduction spray system is also available in an emergency. For this reason, the pressure rise during blowdown should not compromise the integrity of the building.

d. Temperature Rise

During the blowdown period (about 10 sec) the inside surface temperature of the containment building is expected to rise to approximately 250°F. Although this represents a thermal shock to the interior concrete wall, no major damage to the containment building is expected.

e. Coolant Stream Impingement on the Wall

To avoid any possible damage to the containment building wall from the direct impingement of the hot coolant stream, it is planned to build steel baffles near the wall to absorb the mechanical

and thermal loading. This is mainly as a safeguard, since the nearest wall is more than 20 ft from the blowdown nozzles and the escaping fluid is expected to dissipate rapidly with distance.

F. Core Meltdown

1. General

As discussed in Section X-E, following a rupture of the primary system, the major portion of the primary system coolant will be expelled to the containment volume with a very short time. Early in this period of time (blowdown), the power generation in the core will fall promptly to a value of approximately 6-7% of the reactor full power due to the formation of voids in the coolant-moderator and control rod scram. Due to the high discharge rate, high coolant velocities and hence high heat transfer rates existing during this period could cause the fuel and cladding temperatures to decrease rapidly.

With the primary heat removal mechanism diminished as a result of the loss of coolant, fuel and cladding will then undergo rapid heatup due to residual fission product decay heat generation. The heatup will continue until other heat removal mechanisms such as radiation and convection heat transfer become large enough to balance the internal generation rate, or until the onset of melting and subsequent rearrangement of the core enhances the heat removal rate.

The phenomena associated with such an excursion, in particular, the fraction of fuel and clad melting, the amount and type of fission products released, and the extent of damage to the reactor system, are of interest in the safety analysis of power reactors. A preliminary thermal transient analysis has therefore been undertaken to investigate the temperature-time history of the LOFT core following blowdown. The analysis has been divided into a study of 3 phases: (1) core heatup following blowdown, (2) thermal behavior following core "slumping", and (3) thermal behavior of the reactor vessel bottom head. These 3 phases are discussed below.

2. Core Heatup Following Blowdown - Phase I

When the reactor system becomes void of coolant, the entire reactor core will be subject to rapid heatup due to residual fission product decay. The general treatment used in preparing nuclear power reactor safety analysis reports is to consider the core as being intact and to assume no heat losses (adiabatic process) until the entire core melts⁽¹⁻⁶⁾. This method, though perhaps sufficient for reactor safety analysis, does not present a realistic model for predicting the transient thermal behavior of the fuel during the meltdown resulting from a loss of coolant accident.

This phase deals with the heatup rates that can be expected if heat losses by means of radiation, conduction, and convection are considered. The analysis includes the credible possibility that melting of fuel pin cladding will occur with subsequent random dispersion of UO_2 fuel pellets, particles, and powder. In addition, the less credible case of the core remaining intact until the UO_2 melting point is reached is analyzed.

This phase ends at the time when the reactor core either reaches a point of equilibrium temperature and no further melting can occur or changes configuration such that a new heat transfer phenomenon exists.

-
- (1) "Yankee Nuclear Power Station Technical Information and Final Hazards Summary Report", Yankee Atomic Electric Company, License Application, AEC Docket No. 20-29, USAEC Report YAEC-167, 1959.
 - (2) "PWR Loss of Coolant Accident - Core Meltdown Calculations", Westinghouse Electric Corporation, Bettis Atomic Power Division, WAPD-SC-544, May 1957.
 - (3) "Preliminary Hazards Summary Report", Consolidated Edison Company of New York, Inc., Ravenswood Nuclear Generating Unit A, Exhibit A, Vol. I and II, AEC Docket No. 50-204.
 - (4) "Nuclear Merchant Ship Reactor Final Safeguards Report", BAW-1164, June 1960.
 - (5) "Hazard Summary Report", Consolidated Edison Thorium Reactor, Exhibit K-5 (Rev. 1) AEC Docket No. 50-3, January 1960.
 - (6) "Final Safeguard Report for Saxton Reactor", Westinghouse Electric Corporation, NP-10094, October 1961.

The parameters and heat transfer characteristics which may influence the core heatup are: decay heat generation; heat capacity; and free convection, condensation, conduction, and thermal radiation losses.

- (1) Heat Generation: The decay heat rate was determined from time-decay curves⁽¹⁾ for a 50 Mw(t) power level after 400 hours operation. Figure X-4 shows the calculated decay heat generation rate versus time.
- (2) Heat Storage: Heat storage was accounted for in the fuel, cladding, control rods, flow skirt and support barrel. The radiative heat transfer characteristics of the core filler pieces as well as the heat sink property of the thermal shields and reactor vessel were investigated. Heat transmission to the fuel element end boxes and various support plates can be accomplished by axial conduction, radiation, and free convection only. The storage capabilities of these components, as well as the core filler pieces are assumed to be negligible. The heat storage capacity of in-core vapor was assumed to be negligible, since the ratio of the heat capacity of the vapor to UO_2 heat capacity is small.
- (3) Free Convection: Following blowdown, the reactor vessel is assumed to contain a steam atmosphere. Free convection from all surfaces to this environment is analyzed. A reduction was made in the time decay curves to account for losses by this mechanism. Water vapor loss by condensation onto the vessel wall was not considered in the analysis.
- (4) Conduction: The effect of axial heat conduction from UO_2 fuel, fuel cladding, control rods, and fuel assembly end

(1) K. Shure, "Fission Product Decay Energy", WAPD-BT-24, December 1961.

boxes has been investigated and was reported ⁽¹⁾ to have negligible effect on axial heat conduction for in-core components. Since the configuration under investigation was similar to the LOFT core, the losses due to axial conduction between the fuel and in-core hardware was assumed to have negligible effect on the core heat losses of this analysis. The assumption of negligible heat losses is extended to core baffle since axial heat conduction in the baffle is negligible compared to the corresponding radial heat conduction. The heat conduction for in-core vapor was determined to be negligible due to the low thermal conductivity of the steam and the relatively large conduction path. Due to the relatively high value of thermal conductivity for steel, the core baffle was assumed to be a heat sink.

- (5) Thermal Radiation: All sources of thermal radiation heat loss were considered. The effect of radiant heat transfer with steam was also included.

The analysis performed to evaluate heat losses by free or natural convection encompassed two different methods ⁽²⁾⁽³⁾. The magnitude of the heat losses by free convection ranged from approximately 3% to 20% of the total decay heat generation. It was therefore assumed that 10% of the total decay heat generated would be lost by free convection during the first 15 minutes following blowdown.

The next step in the analysis was to investigate the magnitude of heat loss resulting from thermal radiation to the in-core steam ⁽³⁾. It was determined that the magnitude was approximately the same as the heat loss by natural convection. It must be noted, however, that the

-
- (1) A. W. Lemmon, Jr., C. A. Alexander, L. E. Hulbert, and R. B. Filbert, Jr., "Core Temperature Excursions Following a Piping Failure in the Plutonium Recycle Test Reactor", BMI-1356, July 9, 1959.
- (2) J. G. Knudsen and D. L. Katz, Fluid Dynamics and Heat Transfer, McGraw-Hill Book Company, Inc., New York, 1958.
- (3) W. H. McAdams, Heat Transmission, 3rd Edition, McGraw-Hill Book Company, Inc., New York, 1954.

effective heat loss from the fuel pins to the in-core steam atmosphere cannot be obtained by a direct summation of these two terms. In the lower region of the core, thermal radiation heat loss are additive; however, in the upper region of the core the steam temperature exceeds that of the surface of the fuel pin. Consequently, the steam then transfers heat to the fuel pins. The overall effect of gaseous radiation then becomes a perturbation of the core temperature profile.

The heat loss from the fuel rods by radiation from pin-to-pin was evaluated. For this analysis a model consisting of 29 rows of equally spaced pins from the core centerline to the edge of the flow skirt was assumed. Expressions were obtained for the heat generation rate and volume of fuel in terms of row number. View factors were determined⁽¹⁾ for various rows of the selected model and then were weighted to obtain average values. The mean value view factors from core centerline to the edge of the flow skirt are shown in Figure X-5.

In order to carry out a study of the influence of radiative heat loss on the thermal behavior of the fuel, a IBM-650 computer program, HEAT-1⁽²⁾, was utilized. This code is a one dimensional transient conduction code which allows for variation of the material properties and heat source with temperature and time within a maximum of ten regions and over a maximum of 50 mesh intervals. Since the code does not include expressions for computing radiation heat transfer, it was necessary to express radiation heat loss as an "equivalent conductive heat loss". An equivalent conductivity was computed by equating an annular region conduction equation to the general radiation equation and solving for the conductivity. This solution results in a cubic equation in temperature which can be handled by the computer code. The radiation heat loss from the outer boundary was handled in a similar manner, as discussed in the general code writeup.

-
- (1) J. A. Landoni, "View Factors for Heat Radiation in HTGR Core Configuration", General Atomics Division, General Dynamics Corporation, GAMD-3799, December 10, 1962.
 - (2) R. J. Wagner, "HEAT-1 A One-Dimensional Time Dependent or Steady State Heat Conduction Code for the IBM-650", IDO-16877, April 1963.

The first model studied assumed radial radiation heat loss without core disassembly when the cladding temperature reached the melting point. The effect of core radius was investigated for a system in which the power density was average with respect to space and constant with respect to time, hereinafter referred to as constant average power density. This power corresponded to the decay heat produced from power operation at a power density in the UO_2 fuel of 256 kw/liter. The results of these calculations are presented in Figure X-6. The temperature-time history of an infinite core (insulated pin condition) is also presented for comparison. It is important to note that regardless of the core size the temperature does not depart from the insulated pin case until well above the melting point for the cladding. It can also be seen that the slope of the curve in the region below the departure depends primarily on the operating power density in the fuel, $\frac{dT}{dT} = \frac{\frac{P}{P_0} Q}{\rho C_p}$, where Q is the power per liter of UO_2 , and P/P_0 is the fraction of operating power for any given time after shutdown. The effect of a time varying decay heat generation rate is shown in Figure X-7. It is seen that the effect is to reduce the time at which clad melting is reached, since the initial generation rates are several times greater than the average. The center pin average temperature rise is also shown in Figure X-7. This curve is based on the LOFT average axial power density and the maximum radial power density.

The next model analyzed was one in which the power density was allowed to vary spatially and with time. The effect of "fuel slumping" or core disassembly was considered. It was assumed that as the temperature in a given region reached the clad melting temperature, the strength of the fuel pin was lost and the fuel cladding subsequently dropped from the core. As the region of clad melting spreads from the center radially outward, a larger void then appears in the core interior. This phenomenon has two effects; first, the heat source becomes smaller with time, and second, the inner fuel pins begin to "see" the cooler structural components and vessel walls, thereby increasing the relative radiation heat loss from the system. A plot of the fraction of cladding melted as a function of time is shown in

Figure X-8. The effect of the decreasing source (fuel slumping) and the increased radiation heat transfer can be seen from the continuously decreasing slope of the curve. With the model used, the error in computed fractions of melted cladding becomes significant when 75% of the cladding is melted, due to the fact that the fuel pins cannot be described explicitly. The model therefore tends to overestimate the radiant heat loss from the inner surfaces and causes the temperature profile across a region to be exaggerated with respect to the actual pin profiles. In addition, the view factors computed for radiation from the internal surfaces were based upon the assumption that the fuel dropped completely away from the core when the cladding melted thereby allowing the internal surfaces to "see" a cooler area above and below the core. This assumption may not necessarily be valid in that the fuel will fall only as far as the fuel element restraining grids and block one end of the core.

The time to reach clad melting under these conditions is estimated to be approximately 730 seconds. Although the effect of several simultaneous heat removal mechanisms could not be described explicitly with the code used, it has been estimated that natural convection heat transfer to flowing steam may remove from 3% to 20% of the total decay heat generated. The rate of temperature rise would therefore decrease slightly causing the time to reach clad melting to be increased from 730 seconds to a maximum of 900 seconds.

Axial conduction is not felt to a major heat removal mechanism during the core meltdown. Although heat loss by this mechanism is probably negligible, axial conduction may tend to flatten the temperature profile both in the UO_2 and in the cladding. The cladding in a particular region of the core would therefore tend to be at a higher average axial temperature at the onset of melting. This phenomenon could reduce solidification of the stainless steel as it flows downward along the pin. The effective contact conductance between internal surfaces of the stacked column of pellets appears to be the controlling variable in predicting axial temperature profiles in the fuel.

On the basis of the present analysis, it can reasonably be predicted that the onset of clad melting will occur within 1000 seconds after blowdown. It can also be reasonably predicted that in excess of 75% of the cladding melts. The fuel pellets will no longer remain free standing due to the ceramic nature of the material and to the thermal cracking. The core will therefore lose its integrity and move to lower regions of the vessel.

If, for some unaccountable reason, the fuel pellets remain in a free-standing configuration, some fuel can reasonably be expected to reach melting temperature at a time greater than 3000 seconds.

3. Core Rearrangement - Phase II

This phase of the meltdown analysis concerns the period of time between initial distortion of the core and the time at which the fuel and stainless steel conglomerate reach some final position either on the in-vessel core support structures or on the bottom head of the reactor vessel.

a. Physical Disassembly

Few detailed analyses have been attempted to predict the various phenomena which occur during this phase of the loss of coolant accident. Experimental evidence is not available on which to base a description of the physical changes occurring within the reactor core during and subsequent to melting of the fuel cladding. The cladding material as it melts, will attempt to flow toward the bottom of the reactor core. If the lower region of the fuel or fuel support structures are at a significantly cooler temperature than the molten clad, the stainless steel may then solidify on the cool surfaces of the fuel pins and supporting structure. The lower portion of the fuel pins may resemble a candle on which the molten wax has run down and solidified on the sides and near the base. As the heat transfer mechanisms change and more heat is generated, this material may eventually become molten again and deposit upon the core support plate.

b. Restraining Grid Melt-Through

At the lower end of each fuel assembly (Figure IV-6), there is a 3/8 in. stainless steel grid plate which restrains the LOFT fuel pins from movement downward. These grids could become clogged with fuel and cladding materials preventing movement past the area. The thermal behavior of the fuel and clad resting on the restraining grid has been analyzed by two different methods. In the first method, when the average temperature of the fuel and cladding reaches the melting point of stainless steel (2500°F), the core is assumed to "slump" onto the restraining grids. The temperature of the material is therefore assumed to be the clad melting point, and the initial grid temperature is assumed to be the average coolant temperature (540°F). The analysis consisted of calculations assuming a core in the shape of a squat cylinder made up of UO_2 and stainless steel. It was further assumed that no heat losses occur and that the fuel-clad mixture remains at a uniform temperature throughout. This model is not realistic, but was used only to obtain an "order-of-magnitude" estimation of the time required for the core to melt through the restraining grid. With this model, a minimum contact time of 40 seconds is required to melt-through the restraining grids.

A similar problem was run on an IBM-650 computer using the HEAT-1 code previously described. This problem assumed only fuel (UO_2) on the grid and allowed for radiation heat transfer in the axial direction from both the upper and lower surfaces. A plot of temperature of the bottom surface of the restraining grid versus time is shown in Figure X-9. The restraining grid was assumed to fail when the lower surface reached its melting temperature. The second curve shown in Figure X-9, demonstrates the effect of assuming a higher thermal conductivity for the UO_2 region. The UO_2 conductivity used in the HEAT-1 code has the form of a linear function of temperature. The higher conductivity shown in Figure X-9 was obtained by increasing the linear conductivity function by a factor of 3.6. This increased conductivity enables heat to be transferred more readily into the grid, thus causing melt-through at an earlier time. Molten stainless steel remaining in the core is expected to have an effect similar to that of increasing the conductivity of UO_2 .

c. Lower Support Plate Melt-Through

After melt-through of the restraining grid, a portion of the core may become trapped on the lower core support plate and be held for a finite period of time. This situation has been analyzed in a manner identical to that described for the restraining grid. The results of the HEAT-1 problem for this case are shown in Figure X-10 and -11. Figure X-10 shows that if the core does come to rest on the lower support plate, the possibility exists that some fuel melting may occur before the 1-1/4 in. stainless steel support plate fails. Figure X-11 shows the temperature profiles of the core and lower plate obtained from the computer results for different contact times. It is evident from the shapes of these curves that the temperature of the support plate surfaces changes only slightly compared to the center temperature of the fuel. This is apparently caused by the low thermal conductivity assumed for the UO_2 region, which does not allow heat to be readily transferred from the center of the core to the outer surfaces where it may be lost by radiation. The configuration, particle size, and density of the UO_2 -SS mixture during and after meltdown cannot be accurately estimated and will exert a strong influence on the effective thermal conductivity of the mixture.

It should be pointed out that in the previous Phase II calculations, the effects caused by the presence of various core components such as fuel assembly side plates, end boxes, control rods, and control rod components have not been considered.

4. Final Core Geometry - Phase III

a. Analytical Assumptions

The final phase of the LOFT meltdown analysis is concerned with the temperature attained in the fuel and reactor vessel should the UO_2 -SS mixture reach the bottom of the reactor vessel. The time at which this will occur is uncertain. Some of the material may pass through the holes in the restraining grid and the lower support plate and reach the bottom of the reactor vessel relatively early, while other material could be held up by the supporting structure

and would not reach the bottom until some time later. The calculations performed for Phase I and II analyses indicate, however, that a portion of the core can be expected to reach the reactor vessel.

The extent to which melting of the reactor vessel bottom head can be expected to occur cannot be analyzed with any degree of certainty unless the physical situation in and around the bottom of the reactor vessel can be determined. Since the physical status of the melted core and the vessel are not known after meltdown has begun, this problem cannot be solved directly. The analytical approach considered four different cases, thus approximating four ways in which the melt-through situation might occur. The four cases considered are as follows:

Case I: All heat losses are neglected and the bottom head is assumed to contain no water and to be perfectly insulated.

Case II: The reactor vessel is assumed to be uninsulated and to contain no water. No heat is considered to be lost to the vessel internals, but radiation and natural convection heat losses are calculated for the outer surface of the reactor vessel.

Case III: The vessel is assumed to be insulated and to contain 2200 pounds of water. Heat losses inside the vessel due to conduction, convection, and radiation were considered and heat losses from the insulated outer surface were accounted for.

Case IV: The same conditions as for Case III apply except that the vessel is assumed to be uninsulated.

Several basic assumptions were made to facilitate analysis of the various cases. It has been assumed that the entire core melts and that each portion, as it melts, falls immediately to the bottom head. The heat transfer area for the core vessel interface was assumed to be that for a level mass on the hemispherical bottom head (approximately 20 ft²). Heat stored in the dissociated fuel and clad when they reach the reactor bottom head is not accounted for. A 3 in. thick bottom head is assumed for the reactor vessel in these calculations.

b. Case I Analysis

Case I with no heat loss represents the extreme upper limit for the vessel melt-through problem, as melting must eventually occur if a heat source is present and no means exist for dissipation of the heat energy. The heat source for the problem is calculated from the core meltdown data presented under Phase I of the initial investigations. As each portion of the core melts, it is assumed to fall instantaneously to the reactor vessel bottom head where it accumulates, thus heating up and eventually melting through the reactor vessel.

The Case I analysis, although it represents an extreme condition, provides an estimate of the minimum time required for the vessel to be breached. For the assumed conditions, this minimum time is found to be about one hour. The temperature profiles that will exist across the reactor vessel at various times are shown in Figure X-12 for the insulated case.

c. Case II Analysis

Case II is similar to the first case except that the vessel is assumed to be uninsulated. Radiation and natural convection heat losses from the external surface of the vessel are calculated using standard heat transfer relationships and the transient conduction mechanism. The temperature profiles for this uninsulated case are also shown in Figure X-12 for Case II, the melt-through time of the 3 in. bottom head is calculated to be about 1.5 hours after loss of coolant.

d. Case III Analysis

In the third case considered, the effect of water remaining in the vessel was investigated by assuming that 5% of the primary coolant remains in the bottom head after blowdown. The in-vessel heat losses accounted for were conduction into the vessel walls, heat transfer to the stagnant pool, and radiation from the fuel to the water vapor in the vessel after the pool was vaporized. The vessel outer surface was considered to be well insulated.

It was found that the assumed water reservoir will effectively remove all the heat generated in the fuel for approximately two hours. If no other heat loss media are important, the primary effect of the water is to delay melting of the bottom head.

After vaporization of the water is complete, radiation from the fuel to the steam atmosphere in the vessel will commence. The heat loss by thermal radiation from the fuel to the vapor is not easily estimated as the effective temperature of the fuel surface, the vessel walls, and other internal surfaces are not known, the geometric configurations of the fuel mass must be estimated, and values of the thermal radiation parameters can only be approximated. In order to estimate the magnitude of the radiation heat loss, the following assumptions were made:

- (1) the temperature of the fuel is 3000°R ,
- (2) the effective vessel internal surface temperature is 1000°F ,
- (3) the steam pressure is approximately 1.5 atm,
- (4) the emissivity of the steam is 0.5, and
- (5) the emissivities of both steel and fuel are taken to be 0.8.

The heat loss from the fuel mass to the steam vapor was equated to the heat loss from the steam vapor to the internal surfaces. The steam vapor temperature was then found to be approximately 2360°R .

The heat loss from the fuel by this mechanism is calculated to be 4×10^5 Btu. Since the accumulated heat source on the bottom head is greater than 10^6 Btu/hr during the first four hours, it is felt that the loss rate from the fuel by radiation to the steam atmosphere is not great enough to prevent melt-through of the bottom head.

By comparing the heat losses as a function of time with the heat generated versus time, it was found that the integrated heat rate (energy) necessary for vessel melting is reached approximately 4.5 hours after the fuel falls to the bottom head.

It is assumed that the heat losses from the fuel remain essentially constant after melting of the vessel. An energy balance indicates that enough energy is left in the fuel to raise approximately 50% of the fuel mass to the UO_2 melting temperature.

e. Case IV Analysis

The fourth case considered was the same as the previous one except that the effect of convection cooling of the reactor vessel bottom head was included in the calculations. If one considers only the energy necessary to complete or maintain the various heat loss processes, an estimate of the time necessary to melt through the bottom head can be obtained by integrating the curve of heat production versus time and assigning various portions of the integrated energy to the heat loss processes. In this manner, as previously discussed, it is estimated that a minimum time of approximately 4.5 hours is required to melt through the bottom head if conduction losses to the vessel and a pool of stagnant water are considered in addition to radiation heat loss and deposition of sufficient energy in the bottom head to add the heat of fusion for steel. The additional effect, therefore, of accounting for convective losses adds approximately 0.6×10^6 Btu which must be generated by the fuel. This additional energy is sufficient to make a complete melt-through of the reactor vessel improbable. However, local melting of the vessel in the vicinity of the fallen fuel is highly probable, and the resultant stresses induced by the severe temperature gradients may invalidate the integrity of the vessel for further use as a high pressure system.

f. Effect of Stainless Steel in the Core

In the cases previously discussed, the possibility that the pile of core material on the bottom head may consist of an intimate mixture of UO_2 and stainless steel as a heat sink has not been considered. If the configuration on the head contains all the fuel and clad material, there will be 1660 lbs of cladding and 4400 lbs of UO_2 in the pile. Since the melting and boiling points of all the major constituents of stainless steel are below the UO_2 melting temperature, a large portion of the stainless steel may vaporize before any fuel melts.

The heats of fusion and vaporization for the constituents of stainless steel are given in Table X-B. It is calculated that approximately 1.8×10^6 Btu's are required to heatup, melt, and vaporize all

the stainless steel cladding in the core. Hence, it is evident that the steel, if it becomes interspersed with fuel, represents a heat sink of sufficient magnitude to retard or prevent fuel melting.

TABLE X-B

LATENT HEATS FOR STAINLESS STEEL CONSTITUENTS

<u>Element</u>	<u>Heat of Fusion</u>	<u>Heat of Vaporization</u>	<u>Room Temp To Vaporization</u>	<u>Boiling Temp</u>
Cr	42 Btu/lb	818 Btu/lb	1087 Btu/lb	4500°F
Fe	36	842	1115	4960
Ni	41	825	1076	4945
Mn	36	556	814	3900

g. Conclusions

Additional analysis to predict what might occur when fuel reaches the reactor bottom head have been made using the HEAT-1 code and assuming slab configurations of various thicknesses to represent the fuel on the bottom of the reactor vessel. The bottom and sides of the slab were assumed to be insulated, and heat is transferred only from the fuel surface by radiation. The vessel bottom head was considered to be 2 in. carbon steel, and the UO_2 thickness was varied from 4 to 12 inches. The maximum fuel temperature for the different thicknesses is plotted versus time in Figure X-13.

Most cases run using the HEAT-1 code assumed a UO_2 core and neglected any other material which might have been present. The accuracy of this model may be subject to some question. The physical properties used for UO_2 may not be an accurate approximation of these properties for the conglomerate core. The thermal conductivity is one factor which might be questioned. The effect of varying the thermal conductivity in the vessel melt-through problem for an 8 in. thick core is shown in Figure X-14. For use in the HEAT-1 code, this property was approximated by a linear function of temperature in the form $k = \alpha_1 T + \alpha_0$. To determine the effect of varying conductivity the factor α_1 was doubled in one case, and the factor α_0 was

multiplied by five in another. Since α_1 is negative and α_0 is positive, the former case results in a decrease in the conductivity, while the latter amounts to an increase. The higher conductivity allows heat to be removed more readily from the fuel and gives a lower temperature versus time curve than for the case using the normal UO_2 conductivity. Thus, it appears that the physical state of the core, inasmuch as it effects the thermal conductivity can have a significant bearing on the core meltdown, vessel melt-through and possible fuel melting.

As the design of the vessel and molten core container is not yet firm, the transient thermal analysis for Phase III has been of necessity rather general in nature and based upon features which may not exist in the final design. The analysis has shown, however, that the reactor vessel bottom head may be severely damaged due to local melting and may, in fact, be breached. Local melting of UO_2 also appears to be a distinct possibility even before a vessel breach would occur. This conclusion again depends on the heat sink provided by the boiling of stainless steel and the increased heat transfer capability of boiling steel. The core size may have an effect on the severity of damage to the head but does not appear to be a controlling factor as far as the maximum attainable temperature in the fuel is concerned.

In general, it may be concluded that with the present LOFT core design, a loss of coolant from the reactor primary system would result in significant and widespread clad melting. Fuel pellets and parts of the core structure can subsequently be expected to migrate to the vessel bottom head. As a result, the head may be severely damaged and temperatures within the fuel could conceivably attain the melting point of UO_2 .

G. Fission Product Release, Transport and Plateout Phenomena

The subsequent sections include the general fission product behavior criteria used in the prediction of potential radiological dosages during a loss of coolant test. Due to the magnitude of fission product isotopes that could be explored, only the major

inhalation and ingestion dose contributor, radioiodine, will be included in the following discussion. The radioactive noble gases, bromines and solids are treated in the same manner as presented in TID 14844⁽¹⁾, that is, 100% of the noble gases, 25% of the bromines, and 1% of the solids are assumed to be available for release from the containment building at the time the core fuel cladding begins to melt.

In establishing the quantity of radioiodine available for release as a function of time after the simulated pipe rupture, use is made of the available experimental data pertaining to the release fractionation from UO_2 fuel as a function of fuel burnup and temperature and to the data pertaining to the plateout on the containment building surfaces. The mechanisms used in the transport of the iodine from the reactor vessel to the containment building are considered to be by steam generation, temperature and pressure equilization of the steam and diffusion by means of a concentration potential.

The resulting fractionations of the total radioiodine inventory available for release are shown in Figure X-15. A discussion of the mechanisms and their postulated effects are presented in the subsequent paragraphs.

1. Radioiodine Release from Fuel

A number of experiments have been performed^(2,3) at ORNL to investigate the fission product release from sintered UO_2 fuel as a function of fuel burnup, fuel temperature, and carrier atmosphere. Since these parameters encompass those for the LOFT core and test conditions, with the exception of the atmosphere, this experimental data has been employed to determine the quantity of radioiodine that

-
- (1) J. J. DiNunno, et al, Calculations of Distance Factors for Power Test Reactor Sites, TID-14844, March 1962.
 - (2) Nuclear Safety Program Semiannual Progress Report for the Period ending June 30, 1962, ORNL 3319, August 24, 1962.
 - (3) Chemistry Division Annual Progress Report for the period ending June 20, 1961, ORNL 3176, October 10, 1961.

may be liberated from the LOFT core during the loss of coolant test. Most of the experiments used a helium or CO₂ atmosphere. However, as concluded in the reference material, the release in a steam atmosphere, as is expected in LOFT, will not differ significantly from that in a helium atmosphere.

To arrive at the quantity of radioiodine liberated from the LOFT core as a function of time after the simulated pipe rupture, the temperature history (Section X-F) and the ORNL release data for UO₂ fuel, having a burnup of 1000 MWD/TONNE in a helium atmosphere (Figure X-16), were employed. The conservative approach is taken that when the average temperature of the core reached a specified level the appropriate fraction of the radioiodine is instantaneously released from the fuel to the reactor vessel. The release time sequence derived from the available data are enumerated below.

(1) During the time interval from the rupture of the coolant line to 10 minutes after the simulated break, the fuel is expected to be in a transient heating stage but below the temperature required to melt the clad. Since the clad remains integral, no fission product release is expected.

(2) From 10 minutes to 3-1/2 hours after the simulated break, the exposed fuel is expected to be at a maximum average temperature of 2500°F. This time interval, however, is divided into sub-groups where various fuel melting and vaporization phenomena occur.

During the 10 to 30 minute interval the fuel progressively melts from the center out until approximately 75% of the fuel pins have a cladding failure caused by melting. Due to the flux distribution, the central 75% of the core contains approximately 80% of the total fission product inventory. During the clad melting process the average temperature of the exposed fuel is assumed to approximately equal the clad melting temperature (2500°F). A conservative estimate of the temperature dependent fission product release is employed in that the release from a partially clad fuel element is the same as that from a fully exposed fuel element.

During the period between 1/2 and 1-1/2 hours the fuel is in a process of melting through the restraining grid plates, the fuel element housings and the lower core support structure. The temperature of the exposed fuel in the melting process, is assumed to closely approximate the liquefaction temperature of the material in question. The material in all cases considered is similar to that of the fuel clad, so that the melting temperature is about 2500°F. As soon as the fuel melts through the lower core support structure it falls to the bottom head where its temperature is reduced considerably by the water remaining in the bottom head.

During the interval between 1-1/2 to 3-1/2 hours all of the exposed fuel is assumed to be in the bottom head vaporizing the remaining reactor vessel water. The vaporization of the water reduces the exposed fuel considerably below the previous specified temperature. At 3-1/2 hours all of the water has been assumed to be vaporized from the reactor vessel.

The total iodine inventory released from the fuel to the reactor vessel during the time interval between 10 minutes and 3-1/2 hours is 1.6%. To illustrate the instantaneous release assumption, the release of 1.6% occurs at 10 minutes after the pipe rupture.

(3) From 3-1/2 hours to 4 hours the decay gamma heat expands the steam vapor to a temperature equilibrium state. At 4 hours the expansion is assumed to be complete. The average temperature of the fuel during this interval is 3100°F, liberating 12% of the total iodine inventory to the reactor vessel.

(4) After 4 hours the core is assumed to be at an average temperature of 3500°F and the transport phenomena occurring from the reactor atmosphere is produced by a pressure equilization process and a concentration differential diffusion process. The iodine released to the reactor vessel is 40% of the total amount in the core.

2. Transport to the Containment Building

The transport of radioiodine from the primary coolant system to the containment building is the second of the series of

events that must occur before the iodine is available for leakage to the atmosphere. For the purpose of computing the concentration in the containment building four transport mechanisms have been considered: (1) the steam current generated by the vaporization of the water remaining in the reactor vessel, (2) the steam expansion due to temperature equilization, (3) the vapor equilization due to a pressure differential between the reactor vessel and the containment enclosure, and (4) the diffusion due to a concentration differential.

The coolant blowdown calculations indicate that water will remain in the reactor for 3-1/2 hours after the initiation of the test. Throughout this time, the eddy turbulences of the escaping vapor will provide an excellent transport media for carrying the radioiodine from the reactor vessel. Thus, the conservative assumption is made that the radioiodine released from the fuel is instantly transported to the containment building.

Because of the high heat-up rate of the steam after all of the water has been vaporized, the expansion of the vapor will be of sufficient magnitude to warrant the conservative assumption that the iodine is instantaneously released from the reactor vessel to the containment enclosure. The time interval in which this occurs is from 3-1/2 hours to 4 hours.

As an example to illustrate the above assumptions it is assumed that at 10 minutes the fuel instantaneously releases 1.6% of the total iodine inventory to the reactor which in turn is instantaneously released to the containment building. The instantaneous release to the containment environment also occurs at 3-1/2 hours, releasing an additional 10.4% (total 12%) of the total iodine inventory.

The transport of the iodine subsequent to completion of the thermal expansion equilization (4 hours) is accomplished by two mechanisms. The first mechanism is caused by a differential pressure equilization process between the reactor vessel and the containment building. The other mechanism is the transmigration of the iodine resulting from a diffusion concentration potential.

With the assumption that the radioiodine is uniformly distributed throughout the reactor vessel, a differential equation results taking into account both pressure equilization and diffusion.

$$\frac{dC}{dt} = -\frac{Q}{V} C - \frac{DA}{LV} C$$

Where: C = fraction of iodine present at any time t

t = time (hrs)

Q = flow rate of steam due to a pressure differential (Ft³/Hr)

D = diffusion coefficient for iodine $\frac{Ft^2}{Hr}$

A = area of flow pipe (Ft²)

L = length of exit pipe (Ft)

V = volume of reactor vessel

Obtaining an analytical solution for Q as a function of time,

$$Q = \frac{V\alpha}{t}$$

Where: α = Analytical pressure decay exponent

Integrating the above equations the following relationship can be obtained:

$$\frac{C}{C_0} = \left(\frac{t}{t_0} \right)^{-\alpha} e^{-\frac{DA}{LV} (t-t_0)}$$

Where: C₀ = initial fraction of iodine present

t₀ = initial time (hrs)

A plot of the iodine transport to the containment enclosure can be found in Figure X-15.

3. Plateout Inside the Containment Vessel

The last of the series of factors that is considered in determining the quantity of radioiodine available for release to the atmosphere is the plateout phenomenon inside the containment building.

Several experiments have been performed in the United Kingdom to study the plateout phenomenon on various surfaces and in various environments. In one of these experiments⁽¹⁾, a study of the iodine behavior in steam atmosphere on chlorinated rubber base painted surfaces reveals that the iodine remaining suspended in the containment atmosphere as a function of time can be expressed by the following equation.

$$\frac{C}{C_0} = e^{-at}$$

Where: C = fraction of iodine present at any time t

C_0 = initial fraction of iodine present

a = decay constant (hrs⁻¹)

t = time (hrs)

This expression was determined to be valid for a period of 8 to 10 hours, thereafter the initial dispersion of iodine was observed to decrease by deposition to approximately 1% in a period of 1 day.

The results of this experimental data with that obtained from Zenith, Dido and Pluto, where the concentrations of the initial release were several orders of magnitude less, were found to be in close agreement. The concentrations used at Zenith, et al, are of the same order-of-magnitude as those expected at LOFT.

For the time intervals up to 4 hours after the LOFT simulated pipe rupture, the previously described equation is applicable. However after four hours, a modified equation must be employed to account for the time dependent fraction of iodine being liberated. The iodine being liberated, however, has not been subjected to the plateout phenomenon so it can be safely assumed that the time limitation on the empirical equations can be initiated at the time of iodine liberation from the reactor vessel. With this assumption the following analytical approximation can be obtained:

$$C = C_0 e^{-at} + \alpha B e^{-at} \left[\ln t - at + \frac{a^2 t^2}{2 \cdot 2!} - \frac{a^3 t^3}{3 \cdot 3!} + \dots - K \right]$$

(1) J. F. Croft, et al, Experiments of Surface Deposition of Airborne Iodine of High Concentration, AEEW-R265, June 1963.

Where: C = fraction of iodine present at any time $t > 4$ hrs

C_0' = initial fraction of iodine present

a = decay constant (hrs^{-1})

t = time (hrs)

α = pressure decay exponent

B = constant

K = constant for initial time t_0

It is postulated that as the plateout phenomenon progresses with time, the percentage of the less reactive iodine compounds found in the containment atmosphere increases due to the plateout of the more reactive free iodine. Since the plateout characteristics of these compounds are unknown, all of the compounds formed have been considered to be available for leakage to the atmosphere. The quantitative amounts for the formation of compounds have been experimentally determined for a CO/CO_2 and a CO_2/air atmospheres⁽¹⁾. The fractionation was found to be in the range of approximately 10% of the iodine released for the CO/CO_2 environment, however, a significant reduction was observed for the CO_2/air mixture.

Since the fractionation of compounds formed in a steam atmosphere is unknown, the iodine compound fractionation for a CO/CO_2 atmosphere is assumed to be applicable for the LOFT environment. With the above assumption, the fractionation of iodine compounds formed in the LOFT containment environment is approximately 2% of the total iodine in the core which is equivalent to 10% of the iodine released to the containment shell. This fraction is assumed to exist from the time of occurrence (approximately 9 hours) to 5 days.

H. Containment Vessel Leakage

The leakage rate from the containment vessel is greatly dependent on the geometry of the leakage path. If the leakage path behaves like an orifice, the gas flow will be proportional to the square root of

(1) D. A. Collins, et al, unpublished report, TRG Memorandum 1909 (W), October, 1963.

the differential pressure. If, however, the leakage path represents a porous medium, the leakage rate will then be proportional to the differential pressure. The results of a leakage test performed by Atomics International indicates that in reality the leakage rate is proportional to some combination of the differential and the square root of the differential pressure⁽¹⁾.

Since the geometry of the leakage paths through the LOFT containment building are not known, the conservative approach is taken that the leakage rate is proportional to the square root of the differential pressure. By using this approach, the postulated radiological doses resulting from fission product leakage could be over estimated by at least a factor of 5. The fission product leakage rate, which is assumed to be equivalent to the gas leakage, is presented in Figure X-17. This curve is also based on the design criteria that a leakage rate of 0.2% of the free volume per day is attained at 24 psig.

The containment vessel pressure decay history is shown in Figure X-18. The assumptions for the calculation of pressure versus time after loss of primary coolant are as follows:

- (1) 260,000 cu.ft. containment free volume,
- (2) 650 cu. ft. of primary coolant water at an average temperature of 540°F is released into the containment vessel,
- (3) Air in the vessel before blowdown is at 12.5 psia, 100°F and has a relative humidity of 100 percent,
- (4) 70°F average outside ambient temperature,
- (5) 250 tons of steel (containment vessel not included) at 100°F at blowdown,
- (6) 200 tons of steel (reactor vessel included) at 540°F at blowdown,
- (7) Dome is 1/2 in. steel with 6 in. concrete liner, and shell is 1 in. steel with 12 in. concrete liner. The concrete liner is ordinary concrete and there is no additional insulation,
- (8) Core decay heat is based on 800 hr. at 50 Mw(t),

(1) R. L. Koontz, et al, "Low Pressure Containment Building", NAA-SR-7234, March 1963.

- (9) Average solar heat for day and night is used,
- (10) All water that remains liquid after it is released inside the containment vessel is discharged from the vessel through the drains,
- (11) All steam that is condensed due to cooling inside the containment vessel is discharged from the vessel through the drains,
- (12) After blowdown, the heat is absorbed into and through the concrete lining according to the "Schmidt Numerical Method" as given in McAdams "Heat Transmission", Third Edition,
- (13) Concrete specific heat is $0.22 \text{ Btu}/(\text{lb})(^{\circ}\text{F})$, thermal conductivity is $0.5 \text{ Btu}/(\text{ft}^2)(^{\circ}\text{F})$ and density is $144 \cdot \text{lb}/\text{ft}^3$,
- (14) The average outside radiation and convection heat transfer coefficient for the containment dome and top portion of the shell is $3 \text{ Btu}/(\text{ft}^2)(\text{hr})(^{\circ}\text{F})$,
- (15) For the lower portion of the shell which is behind the biological shield the average convection coefficient is $0.3 \text{ Btu}/(\text{ft}^2)(\text{hr})(^{\circ}\text{F})$. The space between the shell and the top of the biological shield is assumed to be open for air circulation,
- (16) The pressure reduction sprays are not used during the test, and
- (17) The amount of heat lost through leakage from the containment vessel during the loss of coolant test is very small compared to the total heat loss, therefore, the amount of heat lost through leakage is not included in the pressure decay calculations.

The vapor temperature inside the containment vessel is estimated and the estimated values are used to calculate the internal pressure. The internal pressure is calculated from an energy balance on the total system considering the amount of energy generated and the amount lost. The vapor temperature corresponding to the calculated pressure inside the containment vessel must be equal to the estimated temperature. If the temperature corresponding to the calculated pressure does not agree with the estimated temperature, the calculations are repeated

using a revised estimated temperature. This process is repeated for various times after blowdown until the calculated temperatures are in agreement with the estimated temperatures.

Because of condensation on the inside surface of the vessel, the surface is assumed to be at the same temperature as the inside vapor. The inside radiation and convection coefficient is very high and for practical purposes, it is assumed to be infinity with no temperature drop between the vapor and the inside surface of the vessel.

The calculated pressure decay curve is only an indication at the containment pressure following a planned loss of coolant test and is not used in designing the facilities for a maximum credible accident. The maximum pressure for the test is much less than the design pressure for the facility.

I. Meteorological Conditions During the Loss of Coolant Test

The simulated pipe rupture for the loss of coolant test involving core meltdown will be initiated during strong lapse meteorological conditions with the wind blowing at least 4 mph from the direction encompassed by an arc extending from 350° to 70°. Reasonable assurance will also be given that these conditions will prevail for a period of approximately 6 hours to permit an assessment of the fission product release before the wind changes to the off site direction.

Since the meteorological conditions cannot be accurately predicted for periods in excess of approximately 6 hours, the conditions for the remainder of the test period are assumed to be in accordance with the most probable conditions based on the past 5 year history for the months of February at the TAN area. These conditions are as follows:

(1) From 6 hours to 24 hours, strong inversion conditions will prevail with the wind blowing at 3 mph from the direction encompassed by arc extending from 110° to 240°, 10% of the time and from the arc 300° to 70°, 90% of the time.

(2) From 24 to 30 hours, strong lapse conditions will again prevail with the wind again blowing at approximately 4 mph from the direction encompassed by an arc extending from 350° to 70°. At

the end of 30 hours the conditions will again change to inversion conditions as described in item (1). This cycle is assumed to be repeated throughout the test period.

The above meteorological data were supplied by the U. S. Weather Bureau at NRTS⁽¹⁾.

J. Test Termination

If it is concluded at any time during the test that a hazardous quantity of fission products is being released to the environment, the emergency procedure will be initiated. The first step to be taken will include energizing the pressure reduction sprays to reduce the containment vessel pressure to atmospheric conditions in order to eliminate the driving force for transporting fission products from the vessel. After the pressure has been reduced, the containment air will then be recirculated through the exhaust filter system until the concentration in the vessel is reduced sufficiently to be released through the stack. The release will be made under controlled meteorological conditions with the wind blowing to the on-site direction.

The pressure reduction spray system is designed to reduce the containment pressure to less than 1 psig in approximately four hours at the maximum pressure conditions. Using the experimental information obtained in performance testing of the pressure reduction systems, it can be determined that the building leakage can be effectively reduced to preclude excessive exposures to the general public.

(1) Private Communication, Letter from J. R. Horan, Director, Health and Safety Division, ID-AEC, to D. E. Williams, Director of Reactor Division, ID-AEC, dated February 17, 1964.

Fig. X-1 - Pressure in Reactor Vessel During Blowdown

Fig. X-2 - Relative Coolant Mass in Reactor Vessel During Blowdown

(A) CORE PRESSURE DROP, PSI
205 PSI MAX.

(B) FLOW RATE THROUGH HOT LEG, $\frac{\text{lbm}}{\text{sec}}$
1150 $\frac{\text{lbm}}{\text{sec}}$ MAX.

(C) FLOW RATE THROUGH
COLD LEG, $\frac{\text{lbm}}{\text{sec}}$
9600 $\frac{\text{lbm}}{\text{sec}}$ MAX.

(D) FLOW RATE THROUGH CORE, $\frac{\text{lbm}}{\text{sec}}$
6100 $\frac{\text{lbm}}{\text{sec}}$ MAX.

(E) EXIT PRESSURE, PSI

TIME, msec., ON ALL ABSCISSAS

50 msec. RUPTURE TIME; CRITICAL FLOW ATTAINED AT 40 msec.

Fig. X-3 - Typical Results from the Subcooled Blowdown Study

Fig. X-4 - Decay Heat Generation Rate Versus Shutdown Time for LOFT
(50 Mw Operation for 400 Hours)

Fig. X-5 - Mean View Factor Versus Row Number for LOFT Core

Fig. X-6 - Core Centerline Temperature Versus Time After Rupture

Fig. X-7 - Temperature Rise Versus Time After Rupture
(Effect of Power Density)

Fig. X-8 - Time for Average Axial Fuel to Reach the Melting Temperature of the Clad (Maximum/Average Radial Power Density = 1.6)

TS-X

Fig. X-9 - Bottom Surface Temperature of Restraining Grid Versus Contact Time

Fig. X-10 - Core and Lower Support Plate Temperature Versus Contact Time

Fig. X-11 - Core and Lower Support Plate Temperature Profiles at Various Times

Fig. X-12 - Reactor Vessel Bottom Head Temperature Profiles

Fig. X-13 - Maximum Fuel Temperature Versus Time for Various Core Thicknesses on the Bottom Head

Fig. X-14 - Average Fuel Temperature Versus Time for Various UO_2 Thermal Conductivities

Fig. X-15 - Iodine Release and Availability as a Function of Time

Fig. X-16 - Iodine Release as a Function of Temperature

Fig. X-17 - Leakage Rate from the Containment Vessel

Fig. X-18 - Pressure Decay in the Containment Building

SECTION XI

RADIOLOGICAL ANALYSIS OF THE LOSS OF COOLANT TEST

XI. RADIOLOGICAL ANALYSIS OF THE LOSS OF COOLANT TEST

A. Analysis

The radiological analysis is based upon realistic assumptions in areas where reliable experimental data exists and upon conservative assumptions in areas where experimental data are not available.

The radiological exposures are tabulated as a function of distances and segments to indicate the related effects of wind persistency with respect to the calculated doses. Table XI-A is included for the purpose of relating the sectors shown in Figure XI-1 to various towns and facilities for which the radiological exposures could occur.

The radiological dosages presented in Tables XI-B through XI-O are based upon the assumption that favorable meteorological conditions exist immediately following initiation of the loss of coolant test and are in accordance with those conditions shown in Table XI-P for the remainder of the five day release period. The meteorological conditions upon which Table XI-P is based were obtained from a five year recorded history by the United States Weather Bureau at the NRTS and are considered to be the most realistic conditions which could exist for a five day release period during the month of February.

In addition to the meteorological conditions and parameters listed in Table XI-P the following assumptions and analytical techniques were used to determine the radiological dosages.

1. Assumptions

- (a) Reactor has operated at 50 Mw for 800 hours,
- (b) fission products released to the containment vessel are 100% of the noble gases, 50% of the bromines, 1% of the solids. The predicted iodine release fractions are presented in Section X-6,
- (c) 50% of the bromine activity released to the containment vessel is removed by plateout, fallout, entrainment in the condensing atmosphere, or by other removal mechanisms,

- (d) the iodine inventory available for release to the external atmosphere is presented in Figure X-15.
- (e) fission product leakage external to the containment shell is proportional to the leak rate curve as shown in Figure X-16,
- (f) depletion of the radioactive cloud from fallout is neglected, however, it can be analytically shown that if this reduction factor is taken into account, the dosages presented in Tables XI-B through XI-O may be significantly reduced dependent upon such parameters as meteorological conditions, release height, deposition velocity, and downwind distance^(1,2), and
- (g) the radiological dosages calculated by the use of the following equations have been modified to account for cloud meandering and wind persistency. The cloud meandering factor is to account for centerline deviation during any given release period and is obtained by averaging the cloud centerline concentration, as given by Sutton's diffusion equation (eq 1) over a $22-1/2^\circ$ sector. The wind persistency factor is to account for wind shift during any given release period and is in accordance with the basic meteorological assumptions as shown in Table XI-P as well as the wind rose data for winter time lapse and inversion conditions as shown in Figures VI-6 and -7. The numerical values for wind persistency, cloud meandering, and the product of wind persistency and cloud meandering factors used to modify the dose equations presented in the following section are shown in Tables XI-Q, -R, and -S, respectively.

2. Analytical Techniques

The analytical techniques upon which the radiological dosages are based employs a modified Sutton's diffusion equation for a continuous point source to compute the deposition gamma, cloud gamma, and inhalation dosages.

-
- (1) "Calculations of the Deposition of Aerosols from Elevated Sources", ORO-171, Walter M. Culkowski.
 - (2) "Aspects of Travel and Deposition of Aerosol and Vapor Clouds", AERE HP/R 1261, A. C. Chamberlain.

a. Sutton's Diffusion Equation for a Continuous Release of Radioactive Effluents⁽¹⁾

$$\frac{X}{Q} = \frac{2}{U \pi C_y C_z D^{2-N}} e^{-\left[\frac{1}{X^{2-N}} \left(\frac{Y^2}{C_y^2} + \frac{H^2}{C_z^2} \right) \right]} \text{ (sec/m}^3\text{)} \quad (1)$$

where:

- X = airborne concentration at any detector position downwind (units/m³)
- U = wind speed (m/sec)
- C_y = horizontal diffusion coefficient (m $\frac{N}{2}$)
- C_z = vertical diffusion coefficient (m $\frac{N}{2}$)
- D = distance to detector position (m)
- N = Sutton's stability parameter (dimensionless)
- Y = distance to crosswind detector position (considered as zero in all cases) (m)
- H = height of release (considered as zero in all cases) (m)
- Q = source strength (unit/sec).

It is necessary to integrate Sutton's equation with respect to time in order to determine the integrated dosage for the various cases considered. Since Q (activity released at any time t) is assumed to vary with time, it is necessary to include this varying parameter in the analytical model and integrate the equation between the appropriate time intervals. The varying parameter, Q(t), as well as the appropriate factors to convert air or ground concentration to units of dosage are considered below.

b. Cloud Gamma Dose

$$D(i) = \sum_{i=1}^{i=N} \left(\frac{X}{Q} \right) \int_{d/u}^{d/u + T} C_{f-1} \cdot L_n \left(T - \frac{d}{u} \right) \cdot Q(T) dt \text{ (rem)} \quad (2)$$

(1) G. A. DeMarrais, N. F. Islitser, "Diffusion Climatology of the NRTS", IDO-12015, U. S. Weather Bureau, April 1960.

where:

- $D(i)$ = cloud gamma dose in segment (i) and is the summation of all the separate meteorological cases which contribute to the dose in segment i.
- C_{f-1} = conversion from air activity to dose rate
 $\left(\frac{0.246 \text{ rem-m}^3}{\text{sec-curie}} \right) (1)$.
- $L_n(T-\frac{d}{u})$ = leak rate which applies during the meteorological case under consideration (sec^{-1}), and
- $Q(T)$ = fission product activity which applies during the meteorological case under consideration and is computed by the IBM-7040 computer. The fission product activity which is used to define $Q(T)$ is shown in Figure XI-2.

Since the solution of the above equation is quite time consuming if a number of atmospheric cases are considered, equation 2 has been programmed for the IBM-7040 computer and the integrated dose is then approximated by the numerical integration technique involving the use of the trapezoidal rule.

c. Deposition Gamma Dose

$$D_{d.g}(i) = \sum_{i=1}^{i=N} \frac{X}{Q} C_{f-2} \left[\int_{d/u}^{d/u+t_1} \int_{d/u}^{d/u+t_1} [V_g \cdot L(t-\frac{d}{u}) \cdot Q(t) + K] dt^2 + \int_{t_1+d/u}^{t_f+d/u} K_{d/u+t_1} dt \right] \quad (3)$$

where:

- V_g = deposition velocity (0.01 m/sec for halogens and 0.002 m/sec for particulates)
- $Q(t)$ = fission product activity available for release at any time t and is shown in Figures XI-3 and XI-4

(1) "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants", WASH-740 (March 1957).

C_{f-2} = conversion factor from units of activity /m² to rem/sec

$$(2.78 \times 10^{-3} \frac{\text{rem-m}^2}{\text{sec-curie}})^{(1)}$$

t_1 = time the deposition occurs over (sec)

t_f = time the dose rate is integrated over (sec)

K = activity on ground at $t - \Delta t$ which is corrected for radioactive decay and added to the amount deposited during Δt seconds to yield the total activity on the ground at any time t between $t = d/u$ and $t = t_f$

Since an accurate determination of the deposition gamma dose involves consideration of small increments of time equation 3 has been programmed for the IBM-7040 computer and the integrated dose is obtained by the use of the trapezoidal rule.

d. Inhalation Dose

The inhalation dose to the various organs shown in Tables XI-B through XI-O is the total dose contribution from the individual elements and isotopes listed in Table XI-T. The differential equation which has been programmed for the IBM-7040 computer and which was used to determine the integrated dose to any individual organ from any selected isotope is shown below.

$$\frac{dA_k(t)}{dt} = -\lambda_b A_k(t) - \lambda_r A_k(t) + B_R \cdot C_f \cdot \frac{X}{Q} \cdot L(t) \cdot N_k(t) \quad (4)$$

where:

$A_k(t)$ = number of atoms in the organ at any time t

λ_b = biological decay factor (sec⁻¹)

λ_r = radioactive decay factor (sec⁻¹)

B_R = breathing rate (2.32×10^{-4} m³/sec)

$L(t)$ = leak rate function (sec⁻¹)

(1) AECU-3066, "Meteorology and Atomic Energy".

C_f = fraction of inhaled material which reaches the critical organ

$N_k(t)$ = decay fission product activity at any time t (atoms)

The numerical solution to equation 4 could become quite time consuming depending upon the number of organs considered and isotopes affecting each organ. Therefore, the solution to equation 4 along with the appropriate decay constants, breathing rate, leak rate, critical fraction, etc., have been programmed for the IBM-7040 computer with the numerical results obtained shown in Tables XI-B through XI-O.

e. Ingestion Dose

The ingestion dosages shown in Tables XI-B through XI-O were obtained by using a numerical value obtained from the ratio of ingestion to inhalation and multiplying this value by the thyroid dose shown as a function of distance in the above tables. This ratio, which is based upon the CERT test⁽¹⁾ was assumed to be 120. The ratio has been reduced by a factor of 20 to allow for the appropriate reduction in the amount of food obtained from grazing during the winter months⁽²⁾. The ingestion dose shown in the above figures is that which would be received by an adult if he were to consume one quart of milk per day for an infinite period of time. Since the mass of a child's thyroid is a factor of ten less than an adult's the ingestion dosage a child would receive would be approximately a factor of ten greater than an adult dose.

f. Direct Radiation From Containment Vessel

The potential radiation exposures from gamma rays emanating from fission products inside the containment vessel following the core meltdown are presented in Figure XI-5. Several curves are presented

(1) C. A. Hawley, "Controlled Environmental Radioiodine Test", IDO-12034.

(2) Private Communications. J. R. Horan, Director Health and Safety ID-AEC to D. E. Williams, Director Reactor Division ID-AEC, February 25, 1964.

showing total gamma dose as a function of distance from the building for a number of exposure times. As shown by these curves, the total exposure to personnel at the Technical Support Facility, the nearest unshielded facility to the containment vessel (approximately 1800 m), is approximately one mrem for a five day exposure. Therefore, it is concluded that the direct radiation from the containment vessel will not disrupt the normal routine at the other facilities in the TAN complex. The area within 1000 ft of the containment vessel will be considered as an exclusion area and access to this area will be regulated by the TAN Health and Safety group.

The direct radiation doses are based on the assumption that 15 percent of the gross fission products are distributed uniformly within the containment volume above the ground floor. For analytical purposes, the containment vessel is separated into three sections in accordance with vessel wall thickness or shielding available to reduce the radiation from the building. The first evaluation considered only the hemispherical dome which is shielded with six inches of concrete. The second calculation considered the cylindrical volume from 35 ft above the grade to spring line of the containment vessel. This region is shielded with twelve inches of concrete. The region below the 35 ft level was also considered, however, because of the concrete biological shield (2 ft of concrete) surrounding this region, the doses were found to be negligible.

B. Conclusions

The radiological analysis for the loss of coolant test is based upon, where possible, realistic assumptions and analytical models. Areas in which sufficient experimental data are not available to justify the use of realistic assumptions are of necessity based upon conservative values. Two of these areas in particular, are cloud depletion due to fallout and the relationship of leakage rate to pressure differential. Cloud depletion has been neglected due to the unavailability of data which would indicate what the true particle size and chemical state of the release particulates and halogens will be after they have migrated through various leakage paths, and hence filtering media, to the external.

atmosphere. The relationship of leakage rate to pressure is in all probability proportional to a combination of a linear and square root function of the pressure differential. However, until the actual leak rate tests are conducted on the LOFT facility the true proportionality constants cannot be determined. It is believed that the combination of these two factors alone (leak rate and cloud depletion) will result in the off-site dosages being reduced by at least a factor of twenty.

Examination of the radiological dosages indicate the maximum off-site exposures occur in the Howe area (Segment K, Table XI-K). The nearest farm land in this direction is approximately seven miles south-east of the LOFT facility. The dosages at this location, as well as the values recommended in the AEC's standards for radiation protection manual⁽¹⁾, are as follows:

Whole Body Dose From Fallout and Cloud Gamma (rem)	8×10^{-4}
Acceptable Dose ⁽¹⁾ (rem)	0.17 ^(a) 0.5 ^(b)
Inhalation Thyroid Dose to a Child* (rem)	4.3×10^{-1}
Ingestion Thyroid Dose to a Child** (rem)	2
Acceptable Thyroid Dose (rem)	0.5 ^(a) 1.5 ^(b)

(a) Based on exposure to individuals.

(b) Based on an average exposure to a suitable population sample.

* Child's inhalation dose is 1.5 times adults inhalation dose.

** Child's ingestion dose is 10 times adults ingestion dose.

The only dosage which appears to exceed the recommended values is the ingestion dose to the thyroid gland. However, if only two of many parameters which will tend to reduce the radiation dosage are considered, namely cloud depletion and leakage rate, it can be concluded that the loss of coolant test can be conducted without exceeding even the lower

(1) Chapter 0524, Standards for Radiation Protection, USAEC.

recommended limits. It must be emphasized that the doses presented are based upon the assumption that the test is continued for the complete five day test period. During this time, however, extensive radiological measurements will be made both in the containment and downwind of the facility. If the fission product release to the atmosphere at any time exceeds that which would result in integrated doses greater than the acceptable limits, the test can be safely terminated before these limits are reached. It should also be noted that if monitored penetrations are installed the potential doses can be significantly reduced.

TABLE XI-A

DISTANCE AND RELATIVE LOCATIONS OF VARIOUS TOWNS
AND FACILITIES WITH RESPECT TO LOFT

Town or Facility	Distance from LOFT		Sector Where The Town or Facility is Located													
	Miles	Meters	A	B	C	E	F	G	H	I	J	K	M	N	O	P
TSF	1.5	2.41×10^3				x										
Monteview	12	1.93×10^4			x											
Howe	15	2.25×10^4										x				
NRF	17	2.74×10^4								x						
EBR-II	18	3.06×10^4						x								
CFA	25	4.18×10^4								x						
Hamer	29	4.67×10^4				x										
Atomic City	30	4.83×10^4							x							
Roberts	30	4.99×10^4				x										
Idaho Falls	40	6.76×10^4					x

TABLE - XI-B

Radiation Dose in North-Northeast Segment
Five Day Release Period

DISTANCE FROM LOFT (METERS)	SEGMENT A INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	1.8×10^0	1.3×10^2	4.4×10^{-3}	2.2×10^1	2.4×10^0	2.4×10^0	1.6×10^0
5×10^2	1.1×10^{-1}	8.2×10^0	2.7×10^{-4}	1.4×10^0	1.5×10^{-1}	1.8×10^{-1}	9.6×10^{-1}
10^3	3.1×10^{-2}	2.4×10^0	7.7×10^{-5}	3.9×10^{-1}	4.2×10^{-2}	5.1×10^{-2}	2.8×10^{-1}
5×10^3	1.8×10^{-3}	1.7×10^{-1}	4.6×10^{-6}	2.8×10^{-2}	2.5×10^{-3}	3.0×10^{-3}	1.6×10^{-2}
10^4	5.4×10^{-4}	3.9×10^{-2}	1.4×10^{-6}	6.7×10^{-3}	7.5×10^{-4}	9.0×10^{-4}	4.8×10^{-3}
5×10^4	2.8×10^{-5}	1.9×10^{-3}	8.4×10^{-8}	3.6×10^{-4}	4.4×10^{-5}	5.3×10^{-4}	2.8×10^{-4}
10^5	7.2×10^{-6}	4.4×10^{-4}	2.4×10^{-8}	9.3×10^{-5}	1.2×10^{-5}	1.4×10^{-5}	7.6×10^{-5}

DISTANCE FROM LOFT (METERS)	SEGMENT A EXTERNAL GAMMA DOSE		
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	INGESTION DOSE (rem)
10^2	2.0×10^0	9.9×10^{-1}	*
5×10^2	1.2×10^{-1}	6.0×10^{-2}	*
10^3	3.4×10^{-2}	1.7×10^{-2}	*
5×10^3	2.0×10^{-3}	1.0×10^{-3}	*
10^4	6.0×10^{-4}	3.1×10^{-4}	2.4×10^{-1}
5×10^4	3.3×10^{-5}	1.7×10^{-5}	1.2×10^{-2}
10^5	8.4×10^{-6}	4.7×10^{-6}	2.6×10^{-3}

* CONTROLLED GRAZING LAND

TABLE - XI-C
Radiation Dose in Northeast Segment
Five Day Release Period

DISTANCE FROM LOFT (METERS)	SEGMENT B INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	1.2×10^0	9.0×10^1	2.9×10^{-3}	1.5×10^1	1.6×10^0	1.6×10^0	1.1×10^0
5×10^2	7.3×10^{-2}	5.5×10^0	1.8×10^{-4}	9.0×10^{-1}	9.7×10^{-2}	1.2×10^{-1}	6.4×10^{-1}
10^3	2.3×10^{-2}	1.7×10^0	5.6×10^{-5}	2.8×10^{-1}	3.0×10^{-2}	3.7×10^{-2}	2.0×10^{-1}
5×10^3	1.2×10^{-3}	9.1×10^{-2}	3.1×10^{-6}	1.5×10^{-2}	1.7×10^{-3}	2.0×10^{-3}	1.1×10^{-2}
10^4	3.6×10^{-4}	2.6×10^{-2}	9.2×10^{-7}	4.5×10^{-3}	5.0×10^{-4}	6.0×10^{-4}	3.2×10^{-3}
5×10^4	1.9×10^{-5}	1.3×10^{-3}	5.6×10^{-8}	2.4×10^{-4}	2.9×10^{-5}	3.5×10^{-4}	1.9×10^{-4}
10^5	4.8×10^{-6}	2.9×10^{-4}	1.6×10^{-8}	6.2×10^{-5}	7.9×10^{-6}	9.3×10^{-6}	5.0×10^{-5}

DISTANCE FROM LOFT (METERS)	SEGMENT B EXTERNAL GAMMA DOSE		
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	INGESTION DOSE (rem)
10^2	1.3×10^0	6.7×10^{-1}	*
5×10^2	7.8×10^{-2}	4.0×10^{-2}	*
10^3	2.5×10^{-2}	1.3×10^{-2}	*
5×10^3	1.4×10^{-3}	6.9×10^{-4}	*
10^4	4.0×10^{-4}	2.1×10^{-4}	1.6×10^{-1}
5×10^4	2.2×10^{-5}	1.2×10^{-5}	7.7×10^{-3}
10^5	5.6×10^{-6}	3.1×10^{-6}	1.8×10^{-3}

* CONTROLLED GRAZING LAND

TABLE - XI-D
Radiation Dose in East-Northeast Segment
Five Day Release Period

DISTANCE FROM LOFT (METERS)	SEGMENT C INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	9.1×10^{-1}	6.8×10^1	2.2×10^{-3}	1.1×10^1	1.2×10^0	1.2×10^0	8.0×10^0
5×10^2	5.5×10^{-2}	4.1×10^0	1.4×10^{-4}	6.8×10^{-1}	7.3×10^{-2}	8.9×10^{-2}	4.8×10^{-1}
10^3	1.6×10^{-2}	1.2×10^0	3.9×10^{-5}	2.0×10^{-1}	2.1×10^{-2}	2.6×10^{-2}	1.4×10^{-1}
5×10^3	9.3×10^{-4}	6.9×10^{-2}	2.3×10^{-6}	1.1×10^{-2}	1.3×10^{-3}	1.5×10^{-3}	8.3×10^{-3}
10^4	2.7×10^{-4}	2.0×10^{-2}	7.0×10^{-7}	3.4×10^{-3}	3.8×10^{-4}	4.5×10^{-4}	2.4×10^{-3}
5×10^4	1.4×10^{-5}	9.6×10^{-4}	4.2×10^{-8}	1.8×10^{-4}	2.2×10^{-5}	2.6×10^{-5}	1.4×10^{-4}
10^5	3.6×10^{-6}	2.2×10^{-4}	1.2×10^{-8}	4.7×10^{-5}	6.0×10^{-6}	7.0×10^{-6}	3.8×10^{-5}

DISTANCE FROM LOFT (METERS)	SEGMENT C EXTERNAL GAMMA DOSE		
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	INGESTION DOSE (rem)
10^2	9.9×10^{-1}	5.0×10^{-1}	*
5×10^2	5.9×10^{-2}	3.0×10^{-2}	*
10^3	1.7×10^{-2}	8.7×10^{-3}	*
5×10^3	1.0×10^{-3}	5.2×10^{-4}	*
10^4	3.0×10^{-4}	1.6×10^{-4}	1.2×10^{-1}
5×10^4	1.6×10^{-5}	8.7×10^{-6}	5.8×10^{-3}
10^5	4.2×10^{-6}	2.3×10^{-6}	1.3×10^{-3}

* CONTROLLED GRAZING LAND

TABLE - XI-E
Radiation Dose in East-Southeast Segment
Five Day Release Period

DISTANCE FROM LOFT (METERS)	SEGMENT E INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10 ²	3.0x10 ⁰	2.3x10 ²	7.4x10 ⁻³	3.7x10 ¹	4.0x10 ⁰	4.1x10 ⁰	2.6x10 ¹
5 x 10 ²	1.8x10 ⁻¹	1.4x10 ¹	4.5x10 ⁻⁴	2.3x10 ⁰	2.4x10 ⁻¹	3.0x10 ⁻¹	1.6x10 ⁰
10 ³	5.3x10 ⁻²	3.9x10 ⁰	1.3x10 ⁻⁴	6.5x10 ⁻¹	7.0x10 ⁻²	8.6x10 ⁻²	4.6x10 ⁻¹
5 x 10 ³	3.1x10 ⁻³	2.3x10 ⁻¹	7.8x10 ⁻⁶	3.8x10 ⁻²	4.2x10 ⁻³	5.1x10 ⁻³	2.8x10 ⁻²
10 ⁴	9.0x10 ⁻⁴	6.6x10 ⁻²	2.3x10 ⁻⁶	1.1x10 ⁻²	1.3x10 ⁻³	1.5x10 ⁻³	8.1x10 ⁻³
5 x 10 ⁴	4.8x10 ⁻⁵	3.2x10 ⁻³	1.4x10 ⁻⁷	6.1x10 ⁻⁴	7.4x10 ⁻⁵	8.8x10 ⁻⁴	4.7x10 ⁻⁴
10 ⁵	1.2x10 ⁻⁵	7.3x10 ⁻⁴	4.0x10 ⁻⁸	1.6x10 ⁻⁴	2.0x10 ⁻⁵	2.3x10 ⁻⁶	1.3x10 ⁻⁴

DISTANCE FROM LOFT (METERS)	SEGMENT E EXTERNAL GAMMA DOSE		
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	INGESTION DOSE (rem)
10 ²	3.3x10 ⁻⁰	1.7x10 ⁰	*
5 x 10 ²	2.0x10 ⁻¹	1.0x10 ⁻¹	*
10 ³	5.7x10 ⁻²	3.7x10 ⁻²	*
5 x 10 ³	3.4x10 ⁻³	1.7x10 ⁻³	*
10 ⁴	1.0x10 ⁻³	5.2x10 ⁻⁴	3.9x10 ⁻¹
5 x 10 ⁴	6.0x10 ⁻⁵	2.9x10 ⁻⁵	1.9x10 ⁻²
10 ⁵	1.4x10 ⁻⁵	7.7x10 ⁻⁶	4.4x10 ⁻³

* CONTROLLED GRAZING LAND

TABLE - XI-F
Radiation Dose in Southeast Segment
Five Day Release Period

DISTANCE FROM LOFT (METERS)	SEGMENT F INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	6.5×10^0	4.9×10^2	1.6×10^{-3}	8.1×10^1	8.7×10^0	8.9×10^0	5.7×10^1
5×10^2	4.0×10^{-1}	3.0×10^1	9.7×10^{-4}	4.9×10^0	5.3×10^{-1}	6.5×10^{-1}	3.5×10^0
10^3	1.1×10^{-1}	8.5×10^0	2.8×10^{-4}	1.4×10^0	1.5×10^{-1}	1.6×10^{-1}	1.0×10^0
5×10^3	6.7×10^{-3}	4.9×10^{-1}	1.7×10^{-5}	8.2×10^{-2}	9.1×10^{-3}	1.1×10^{-2}	3.6×10^{-2}
10^4	2.0×10^{-3}	1.4×10^{-1}	5.0×10^{-6}	2.4×10^{-2}	2.7×10^{-3}	3.3×10^{-3}	1.8×10^{-2}
5×10^4	1.0×10^{-4}	6.9×10^{-3}	3.1×10^{-7}	1.3×10^{-3}	1.6×10^{-4}	1.9×10^{-4}	1.0×10^{-3}
10^5	2.6×10^{-5}	1.6×10^{-3}	8.6×10^{-8}	3.4×10^{-4}	4.3×10^{-5}	5.1×10^{-5}	2.7×10^{-4}

DISTANCE FROM LOFT (METERS)	SEGMENT F EXTERNAL GAMMA DOSE		
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	INGESTION DOSE (rem)
10^2	7.1×10^0	3.6×10^0	*
5×10^2	4.3×10^{-1}	2.4×10^{-1}	*
10^3	1.2×10^{-1}	6.3×10^{-2}	*
5×10^3	7.3×10^{-3}	3.8×10^{-3}	*
10^4	2.2×10^{-3}	1.1×10^{-3}	8.5×10^{-1}
5×10^4	1.2×10^{-4}	6.3×10^{-5}	4.2×10^{-2}
10^5	3.1×10^{-5}	1.7×10^{-5}	9.5×10^{-3}

* CONTROLLED GRAZING LAND

TABLE - XI-G

Radiation Dose in South-Southeast Segment

Five Day Release Period

DISTANCE FROM LOFT (METERS)	SEGMENT G INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	1.3×10^{-1}	9.8×10^{-2}	3.2×10^{-2}	1.6×10^{-1}	1.7×10^{-1}	1.8×10^{-1}	1.1×10^{-2}
5×10^2	7.9×10^{-1}	5.9×10^{-1}	1.9×10^{-3}	9.8×10^0	1.1×10^0	1.3×10^0	7.0×10^0
10^3	2.3×10^{-1}	1.7×10^{-1}	5.6×10^{-4}	2.8×10^0	3.1×10^{-1}	3.7×10^{-1}	2.0×10^0
5×10^3	1.3×10^{-2}	9.9×10^{-1}	3.4×10^{-5}	1.7×10^{-1}	1.8×10^{-2}	2.2×10^{-2}	1.2×10^{-1}
10^4	4.0×10^{-3}	2.9×10^{-1}	1.0×10^{-5}	5.0×10^{-2}	5.5×10^{-3}	6.7×10^{-3}	3.6×10^{-2}
5×10^4	2.1×10^{-4}	1.4×10^{-2}	6.1×10^{-7}	2.6×10^{-3}	3.2×10^{-4}	3.8×10^{-4}	2.0×10^{-3}
10^5	5.2×10^{-5}	3.2×10^{-3}	1.0×10^{-7}	6.7×10^{-4}	8.6×10^{-5}	1.0×10^{-4}	5.5×10^{-4}

DISTANCE FROM LOFT (METERS)	SEGMENT G EXTERNAL GAMMA DOSE		
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	INGESTION DOSE (rem)
10^2	1.5×10^{-1}	7.3×10^0	*
5×10^2	8.5×10^{-1}	4.4×10^{-1}	*
10^3	2.5×10^{-1}	1.3×10^{-1}	*
5×10^3	1.5×10^{-2}	7.5×10^{-3}	*
10^4	4.5×10^{-3}	2.3×10^{-3}	*
5×10^4	2.4×10^{-4}	1.3×10^{-4}	8.3×10^{-2}
10^5	6.1×10^{-5}	3.3×10^{-5}	1.9×10^{-2}

* CONTROLLED GRAZING LAND

TABLE - XI-H
Radiation Dose in South Segment
Five Day Release Period

DISTANCE FROM LOFT (METERS)	SEGMENT H INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	1.3×10^1	1.0×10^3	3.2×10^{-2}	1.6×10^2	1.8×10^1	1.8×10^1	1.6×10^2
5×10^2	8.0×10^{-1}	6.2×10^1	2.0×10^{-3}	9.9×10^0	1.1×10^0	1.3×10^0	7.0×10^0
10^3	2.3×10^{-1}	1.8×10^1	5.7×10^{-4}	2.9×10^0	3.1×10^{-1}	3.7×10^{-1}	2.0×10^0
5×10^3	1.4×10^{-2}	1.0×10^0	3.4×10^{-5}	1.7×10^{-1}	1.8×10^{-2}	2.2×10^{-2}	1.2×10^{-1}
10^4	4.0×10^{-3}	3.0×10^{-1}	1.0×10^{-5}	5.0×10^{-2}	5.6×10^{-3}	6.7×10^{-3}	3.6×10^{-2}
5×10^4	2.1×10^{-4}	1.4×10^{-2}	6.1×10^{-7}	2.7×10^{-3}	3.2×10^{-4}	3.8×10^{-4}	2.0×10^{-3}
10^5	5.2×10^{-5}	3.2×10^{-3}	1.0×10^{-7}	6.8×10^{-4}	8.6×10^{-5}	1.0×10^{-4}	5.5×10^{-4}

DISTANCE FROM LOFT (METERS)	SEGMENT H EXTERNAL GAMMA DOSE		
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	INGESTION DOSE (rem)
10^2	1.5×10^1	7.5×10^0	*
5×10^2	8.7×10^{-1}	4.4×10^{-1}	*
10^3	2.5×10^{-1}	1.3×10^{-1}	*
5×10^3	1.5×10^{-2}	7.7×10^{-3}	*
10^4	4.5×10^{-3}	2.3×10^{-3}	*
5×10^4	2.4×10^{-4}	1.3×10^{-4}	8.4×10^{-2}
10^5	6.1×10^{-5}	3.3×10^{-5}	1.9×10^{-2}

* CONTROLLED GRAZING LAND

TABLE - XI-I
Radiation Dose in South-Southwest Segment
Five Day Release Period

DISTANCE FROM LOFT (METERS)	SEGMENT I INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	1.1×10^{-1}	8.5×10^{-2}	2.5×10^{-2}	1.3×10^{-2}	1.4×10^{-1}	1.4×10^{-1}	9.2×10^{-1}
5×10^2	6.3×10^{-1}	5.1×10^{-1}	1.5×10^{-3}	7.7×10^0	8.3×10^{-1}	1.0×10^0	5.6×10^0
10^3	1.8×10^{-1}	1.4×10^{-1}	4.4×10^{-4}	2.2×10^0	2.4×10^{-1}	2.9×10^{-1}	1.6×10^0
5×10^3	1.1×10^{-2}	8.1×10^{-1}	2.6×10^{-5}	1.3×10^{-1}	1.4×10^{-2}	1.7×10^{-2}	9.3×10^{-2}
10^4	4.1×10^{-3}	2.3×10^{-1}	7.8×10^{-6}	3.8×10^{-2}	4.2×10^{-3}	5.1×10^{-3}	2.7×10^{-2}
5×10^4	1.6×10^{-4}	1.1×10^{-2}	4.7×10^{-7}	2.1×10^{-3}	2.5×10^{-4}	3.0×10^{-4}	1.6×10^{-3}
10^5	4.1×10^{-5}	2.5×10^{-3}	1.3×10^{-7}	5.2×10^{-4}	6.7×10^{-5}	7.8×10^{-5}	4.2×10^{-4}

DISTANCE FROM LOFT (METERS)	SEGMENT I EXTERNAL GAMMA DOSE		
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	INGESTION DOSE (rem)
10^2	1.5×10^{-1}	6.0×10^0	*
5×10^2	6.8×10^{-1}	3.5×10^{-1}	*
10^3	1.9×10^{-1}	1.0×10^{-2}	*
5×10^3	1.2×10^{-2}	6.0×10^{-4}	*
10^4	3.4×10^{-3}	1.8×10^{-4}	*
5×10^4	1.8×10^{-4}	9.9×10^{-5}	6.5×10^{-2}
10^5	4.7×10^{-5}	2.6×10^{-5}	1.5×10^{-2}

* CONTROLLED GRAZING LAND

TABLE - XI-J
Radiation Dose in Southwest Segment
Five Day Release Period

DISTANCE FROM LOFT (METERS)	SEGMENT J INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	8.6×10^0	7.2×10^2	2.0×10^{-2}	1.1×10^2	1.1×10^1	1.1×10^1	7.3×10^1
5×10^2	5.1×10^{-1}	4.3×10^1	1.2×10^{-3}	6.3×10^0	6.7×10^{-1}	8.1×10^{-1}	4.4×10^0
10^3	1.5×10^{-1}	1.2×10^1	3.5×10^{-4}	1.8×10^0	1.9×10^{-1}	2.3×10^{-1}	1.3×10^0
5×10^3	8.5×10^{-3}	6.7×10^{-1}	2.1×10^{-5}	1.0×10^{-1}	1.1×10^{-2}	1.4×10^{-2}	7.5×10^{-2}
10^4	2.5×10^{-4}	1.9×10^{-1}	6.3×10^{-6}	3.0×10^{-2}	3.4×10^{-3}	4.1×10^{-3}	2.2×10^{-2}
5×10^4	1.3×10^{-4}	8.9×10^{-3}	3.8×10^{-7}	1.7×10^{-3}	2.0×10^{-4}	2.4×10^{-4}	1.3×10^{-3}
10^5	3.3×10^{-5}	2.0×10^{-3}	1.1×10^{-7}	4.2×10^{-4}	5.4×10^{-5}	6.3×10^{-5}	3.4×10^{-4}

DISTANCE FROM LOFT (METERS)	SEGMENT J EXTERNAL GAMMA DOSE		
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	INGESTION DOSE (rem)
10^2	9.3×10^0	5.0×10^0	*
5×10^2	5.5×10^{-1}	2.9×10^{-1}	*
10^3	1.6×10^{-1}	8.4×10^{-2}	*
5×10^3	9.3×10^{-3}	4.9×10^{-3}	*
10^4	2.7×10^{-3}	1.4×10^{-3}	*
5×10^4	1.5×10^{-4}	8.1×10^{-5}	5.4×10^{-2}
10^5	3.8×10^{-5}	2.1×10^{-5}	1.2×10^{-2}

* CONTROLLED GRAZING LAND

TABLE - XI-K
Radiation Dose in West-Southwest Segment
Five Day Release Period

DISTANCE FROM LOFT (METERS)	SEGMENT K INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	4.8×10^0	4.1×10^2	1.2×10^{-2}	5.9×10^1	6.3×10^0	6.4×10^0	4.1×10^1
5×10^2	2.9×10^{-1}	2.4×10^1	6.9×10^{-4}	3.5×10^0	3.8×10^{-1}	4.6×10^1	2.5×10^0
10^3	8.5×10^{-2}	6.9×10^0	2.0×10^{-4}	1.0×10^0	1.1×10^{-1}	1.3×10^{-1}	7.1×10^{-1}
5×10^3	4.8×10^{-3}	3.8×10^{-1}	1.2×10^{-5}	5.9×10^{-2}	6.4×10^{-3}	7.8×10^{-3}	4.2×10^{-2}
10^4	1.4×10^{-3}	1.1×10^{-1}	3.5×10^{-6}	1.7×10^{-2}	1.9×10^{-3}	2.3×10^{-3}	1.2×10^{-2}
5×10^4	7.3×10^{-5}	5.0×10^{-3}	2.1×10^{-7}	9.8×10^{-4}	1.1×10^{-4}	1.3×10^{-4}	7.1×10^{-4}
10^5	1.8×10^{-5}	1.2×10^{-3}	6.0×10^{-8}	2.4×10^{-4}	3.0×10^{-5}	3.6×10^{-5}	1.9×10^{-4}

DISTANCE FROM LOFT (METERS)	SEGMENT K EXTERNAL GAMMA DOSE		
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	INGESTION DOSE (rem)
10^2	5.2×10^0	2.8×10^0	*
5×10^2	3.1×10^{-1}	1.6×10^{-1}	*
10^3	9.0×10^{-2}	4.7×10^{-2}	*
5×10^3	5.2×10^{-3}	2.8×10^{-3}	*
10^4	1.6×10^{-3}	8.2×10^{-4}	6.4×10^{-1}
5×10^4	8.4×10^{-5}	4.5×10^{-5}	3.0×10^{-2}
10^5	2.2×10^{-5}	1.2×10^{-5}	6.9×10^{-3}

* CONTROLLED GRAZING LAND

TABLE - XI-L
Radiation Dose in West-Northwest Segment
Five Day Release Period

DISTANCE FROM LOFT (METERS)	SEGMENT M INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	6.0×10^{-1}	4.5×10^1	1.5×10^{-3}	6.4×10^0	8.0×10^{-1}	8.2×10^{-1}	4.6×10^0
5×10^2	3.7×10^{-2}	2.8×10^0	9.0×10^{-5}	4.6×10^{-1}	4.9×10^{-2}	6.0×10^{-2}	3.3×10^{-1}
10^3	1.1×10^{-2}	7.9×10^{-1}	2.6×10^{-5}	1.3×10^{-1}	1.4×10^{-2}	1.7×10^{-2}	9.3×10^{-2}
5×10^3	6.2×10^{-4}	4.6×10^{-2}	1.6×10^{-6}	7.6×10^{-3}	8.4×10^{-4}	1.0×10^{-3}	5.5×10^{-3}
10^4	1.8×10^{-4}	1.3×10^{-2}	4.6×10^{-7}	2.2×10^{-3}	2.5×10^{-4}	3.0×10^{-4}	1.6×10^{-3}
5×10^4	1.8×10^{-6}	6.4×10^{-4}	2.8×10^{-8}	1.2×10^{-4}	1.5×10^{-5}	1.8×10^{-5}	9.3×10^{-5}
10^5	2.4×10^{-6}	1.5×10^{-4}	7.9×10^{-9}	3.1×10^{-5}	4.0×10^{-6}	4.7×10^{-6}	2.5×10^{-5}

DISTANCE FROM LOFT (METERS)	SEGMENT M EXTERNAL GAMMA DOSE		
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	INGESTION DOSE (rem)
10^2	6.6×10^{-1}	3.3×10^{-1}	*
5×10^2	4.0×10^{-2}	2.2×10^{-2}	*
10^3	1.1×10^{-2}	5.8×10^{-3}	*
5×10^3	6.8×10^{-4}	3.5×10^{-4}	*
10^4	2.0×10^{-4}	1.0×10^{-4}	7.9×10^{-2}
5×10^4	1.1×10^{-5}	5.8×10^{-6}	3.8×10^{-3}
10^5	2.8×10^{-6}	1.5×10^{-6}	8.8×10^{-4}

* CONTROLLED GRAZING LAND

TABLE - XI-M
Radiation Dose in Northwest Segment
Five Day Release Period

DISTANCE FROM LOFT (METERS)	SEGMENT N INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	3.1×10^{-1}	2.3×10^1	7.4×10^{-4}	3.8×10^0	4.0×10^{-1}	4.1×10^{-1}	2.7×10^0
5×10^2	1.8×10^{-2}	1.4×10^0	4.5×10^{-5}	2.3×10^{-1}	2.5×10^{-2}	3.0×10^{-2}	1.6×10^{-1}
10^3	5.3×10^{-3}	4.0×10^{-1}	1.3×10^{-5}	6.6×10^{-2}	7.1×10^{-3}	8.6×10^{-3}	4.7×10^{-2}
5×10^3	3.1×10^{-4}	2.3×10^{-2}	7.9×10^{-7}	3.8×10^{-3}	4.2×10^{-3}	5.2×10^{-4}	2.8×10^{-3}
10^4	9.1×10^{-5}	6.6×10^{-3}	2.3×10^{-7}	1.1×10^{-3}	1.3×10^{-4}	1.5×10^{-4}	8.2×10^{-4}
5×10^4	4.8×10^{-6}	3.2×10^{-4}	1.4×10^{-8}	6.1×10^{-5}	7.4×10^{-6}	8.9×10^{-6}	4.7×10^{-5}
10^5	1.2×10^{-6}	8.3×10^{-5}	4.0×10^{-9}	1.6×10^{-5}	2.0×10^{-6}	2.4×10^{-6}	1.3×10^{-5}

DISTANCE FROM LOFT (METERS)	SEGMENT N EXTERNAL GAMMA DOSE		
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	INGESTION DOSE (rem)
10^2	3.3×10^{-1}	1.7×10^{-1}	*
5×10^2	2.0×10^{-2}	1.0×10^{-2}	*
10^3	5.8×10^{-3}	2.9×10^{-3}	*
5×10^3	3.4×10^{-4}	1.7×10^{-4}	*
10^4	1.0×10^{-4}	5.3×10^{-5}	4.0×10^{-2}
5×10^4	5.5×10^{-6}	2.9×10^{-6}	1.9×10^{-3}
10^5	1.4×10^{-6}	7.7×10^{-7}	5.0×10^{-4}

* CONTROLLED GRAZING LAND

TABLE - XI-N
Radiation Dose in North-Northwest Segment
Five Day Release Period

DISTANCE FROM LOFT (METERS)	SEGMENT 0 INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	3.4×10^{-1}	3.4×10^{-1}	1.1×10^{-3}	5.6×10^0	6.0×10^{-1}	6.2×10^{-1}	4.0×10^0
5×10^2	2.8×10^{-2}	2.1×10^0	6.7×10^{-5}	3.4×10^{-1}	3.7×10^{-2}	4.5×10^{-2}	2.4×10^{-1}
10^3	7.9×10^{-3}	5.9×10^{-1}	2.0×10^{-5}	9.8×10^{-2}	1.1×10^{-2}	1.3×10^{-2}	7.0×10^{-2}
5×10^3	4.7×10^{-4}	3.4×10^{-2}	1.2×10^{-6}	5.7×10^{-3}	6.3×10^{-4}	7.7×10^{-4}	4.2×10^{-3}
10^4	1.4×10^{-5}	9.9×10^{-3}	3.5×10^{-7}	1.7×10^{-3}	1.8×10^{-4}	2.3×10^{-4}	1.2×10^{-3}
5×10^4	7.2×10^{-6}	4.8×10^{-4}	2.1×10^{-8}	9.1×10^{-5}	1.1×10^{-5}	1.3×10^{-5}	7.0×10^{-5}
10^5	1.8×10^{-6}	1.1×10^{-4}	5.9×10^{-9}	2.3×10^{-5}	3.0×10^{-6}	3.5×10^{-6}	1.9×10^{-5}

DISTANCE FROM LOFT (METERS)	SEGMENT 0 EXTERNAL GAMMA DOSE		
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	INGESTION DOSE (rem)
10^2	4.9×10^{-1}	2.5×10^{-1}	*
5×10^2	3.0×10^{-2}	1.5×10^{-2}	*
10^3	8.6×10^{-3}	4.4×10^{-3}	*
5×10^3	5.1×10^{-4}	2.6×10^{-4}	*
10^4	1.5×10^{-4}	7.9×10^{-5}	5.9×10^{-2}
5×10^4	8.3×10^{-6}	4.4×10^{-6}	2.9×10^{-3}
10^5	2.1×10^{-6}	1.1×10^{-6}	6.6×10^{-4}

* CONTROLLED GRAZING LAND

TABLE - XI-0
Radiation Dose in North Segment
Five Day Release Period

DISTANCE FROM LOFT (METERS)	SEGMENT P INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	1.2×10^0	9.0×10^1	2.9×10^{-3}	1.5×10^1	1.6×10^0	1.6×10^0	1.1×10^1
5×10^2	7.3×10^{-2}	5.5×10^0	1.8×10^{-4}	9.0×10^{-1}	9.7×10^{-2}	1.2×10^{-1}	6.4×10^{-1}
10^3	2.3×10^{-2}	1.7×10^0	5.6×10^{-5}	2.8×10^{-1}	3.0×10^{-2}	3.7×10^{-2}	2.0×10^{-1}
5×10^3	1.2×10^{-3}	9.1×10^{-2}	3.1×10^{-6}	1.5×10^{-2}	1.7×10^{-3}	2.0×10^{-3}	1.1×10^{-2}
10^4	3.6×10^{-4}	2.6×10^{-2}	9.2×10^{-7}	4.5×10^{-3}	5.0×10^{-4}	6.0×10^{-4}	3.2×10^{-3}
5×10^4	1.9×10^{-5}	1.3×10^{-3}	5.6×10^{-8}	2.4×10^{-4}	2.9×10^{-5}	3.5×10^{-4}	1.9×10^{-4}
10^5	4.8×10^{-6}	2.9×10^{-4}	1.6×10^{-8}	6.2×10^{-5}	7.9×10^{-6}	9.3×10^{-6}	5.0×10^{-5}

DISTANCE FROM LOFT (METERS)	SEGMENT P EXTERNAL GAMMA DOSE		
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	INGESTION DOSE (rem)
10^2	1.3×10^0	6.7×10^{-1}	*
5×10^2	7.8×10^{-2}	4.0×10^{-2}	*
10^3	2.5×10^{-2}	1.3×10^{-2}	*
5×10^3	1.4×10^{-3}	6.9×10^{-4}	*
10^4	4.0×10^{-4}	2.1×10^{-4}	1.6×10^{-1}
5×10^4	2.2×10^{-5}	1.2×10^{-5}	7.7×10^{-2}
10^5	5.6×10^{-6}	3.1×10^{-6}	1.8×10^{-3}

* CONTROLLED GRAZING LAND

TABLE XI-P
METEOROLOGICAL CONDITIONS AND DURATION OF EACH
FOR FIVE DAY TEST PERIOD

Time After Blowdown	Meteorological Conditions and Diffusion Parameters Assumed for Calculation of Radiological Dosages	
(Days)	Strong Lapse (N=0.20)	Strong Inversion (N=0.50)
0 through 4	6 hours, wind from 350° to 70°, 4 mph, $C_y = C_z =$ 0.334	18 hours, wind from 110° to 240°, 10% of the time and from 300° to 70°, 90% of the time, 3 mph, $C_y = 0.280$ $C_z = 0.035$

TABLE XI-Q
CORRECTION FACTOR TO ACCOUNT FOR CLOUD MEANDERING OVER A 22-1/2° SECTOR

Distance (meters)	Cloud Meandering Factor (C_m)	
	Lapse Condition N = 0.20	Inversion Condition N = 0.50
10^2	0.95	0.40
5×10^2	0.81	0.27
10^3	0.75	0.22
5×10^3	0.65	0.15
10^4	0.60	0.13
5×10^4	0.51	0.09
10^5	0.47	0.07

The cloud meandering factor is obtained as follows:

$$C_m^1 = \frac{2Q}{\pi u C_y C_z D^{2-n}} \int_{-\infty}^{+\infty} e^{\frac{-y^2}{C_y^2 D^{2-n}}} dy$$

Averaging the result from the above equation over a $22-1/2^\circ$ sector, the following equation is obtained:

$$C_m^1 = \frac{16Q}{\pi^{3/2} U C_z} D^{\frac{4-N}{2}}$$

Since the radiological dosages have been calculated by the use of Sutton's equation (eq 1) for a crosswind distance of zero meters, the cloud meandering factor is then a ratio of equation 4 to equation 1.

$$C_m = C_m^1 = \frac{\frac{16Q}{\pi^{3/2} U C_z} D^{\frac{4-N}{2}}}{\frac{2Q}{\pi U C_y C_z D^{2-N}}} \quad (1)$$

$$C_m = \frac{8C_y}{\pi^{1/2} D^{N/2}}$$

TABLE XI-R

CORRECTION FACTOR TO ACCOUNT FOR WIND VARIATION DURING THE
FIVE DAY RELEASE PERIOD

Wind From	Sector Dose Occurs In	Wind Frequency Based on Winter Time Conditions (Figs. VI-6 & -7) (percent)	Wind Frequency Modified to account for Conditions Specified in Table XI-P (percent)	Atmosphere Conditions Considered
N	H	6.5	17.11	Lapse
NNE	I	10.5	27.63	
NE	J	13.0	34.21	
ENE	K	8.0	21.05	
ESE	M	2.0	0.93	Inversion 10% of Time From 110°-240°
SE	N	1.0	0.47	
SSE	O	1.5	0.70	
S	P	4.0	1.86	
SSW	A	6.0	2.78	
SW	B	4.0	1.86	
WSW	C	3.0	1.40	
WNW	E	3.0	4.64	Inversion 90% of Time From 300°-70°
NW	F	6.5	10.1	
NNW	G	13.0	20.2	
N	H	13.0	20.2	
NNE	I	10.0	15.5	
NE	J	8.0	12.4	
ENE	K	4.5	6.96	

TABLE XI-S

CORRECTION FACTOR - PRODUCT OF WIND PERSISTENCY AND CLOUD MEANDERING FACTORS.

Wind From	Sector Dose Occurs In	Correction Factor for Inversion Conditions (N = 0.50) and Following Distances(m)						
		10^2	5×10^2	10^3	5×10^3	10^4	5×10^4	10^6
SSW	A	1.1×10^{-2}	7.4×10^{-3}	6.1×10^{-3}	4.1×10^{-3}	3.5×10^{-3}	2.4×10^{-3}	1.9×10^{-3}
SW	B	7.4×10^{-3}	5.0×10^{-3}	4.4×10^{-3}	2.8×10^{-3}	2.4×10^{-3}	1.6×10^{-3}	1.3×10^{-3}
WSW	C	5.6×10^{-3}	3.7×10^{-3}	3.1×10^{-3}	2.1×10^{-3}	1.8×10^{-3}	1.2×10^{-3}	9.7×10^{-4}
WNW	E	1.9×10^{-2}	1.2×10^{-2}	1.0×10^{-2}	6.9×10^{-3}	5.9×10^{-3}	4.0×10^{-3}	3.2×10^{-3}
NW	F	4.1×10^{-2}	2.7×10^{-2}	2.2×10^{-2}	1.5×10^{-2}	1.3×10^{-2}	8.7×10^{-3}	7.0×10^{-3}
NNW	G	8.1×10^{-2}	5.4×10^{-2}	4.5×10^{-2}	3.0×10^{-2}	2.6×10^{-2}	1.7×10^{-2}	1.4×10^{-2}
N	H	8.1×10^{-2}	5.4×10^{-2}	4.5×10^{-2}	3.0×10^{-2}	2.6×10^{-2}	1.7×10^{-2}	1.4×10^{-2}
NNE	I	6.2×10^{-2}	4.1×10^{-2}	3.4×10^{-2}	2.3×10^{-2}	2.0×10^{-2}	1.3×10^{-2}	1.1×10^{-2}
NE	J	5.0×10^{-2}	3.3×10^{-2}	2.7×10^{-2}	1.8×10^{-2}	1.6×10^{-2}	1.1×10^{-2}	8.6×10^{-3}
ENE	K	2.8×10^{-2}	1.9×10^{-2}	1.5×10^{-2}	1.0×10^{-2}	8.9×10^{-3}	6.0×10^{-3}	4.9×10^{-3}
ESE	M	3.7×10^{-3}	2.5×10^{-3}	2.1×10^{-3}	1.4×10^{-3}	1.2×10^{-3}	8.0×10^{-4}	6.4×10^{-4}
SE	N	1.9×10^{-3}	1.3×10^{-3}	1.0×10^{-3}	7.0×10^{-4}	6.0×10^{-4}	4.0×10^{-4}	3.2×10^{-4}
SSE	O	2.8×10^{-3}	1.9×10^{-3}	1.6×10^{-3}	1.0×10^{-3}	8.9×10^{-4}	6.0×10^{-4}	4.8×10^{-4}
S	P	7.4×10^{-3}	5.0×10^{-3}	4.4×10^{-3}	2.8×10^{-3}	2.4×10^{-3}	1.6×10^{-3}	1.3×10^{-3}
Correction Factor for Lapse Conditions, N = 0.20								
N	H	1.6×10^{-1}	1.4×10^{-1}	1.3×10^{-1}	1.1×10^{-1}	1.0×10^{-1}	8.8×10^{-2}	8.1×10^{-2}
NNE	I	2.6×10^{-1}	2.2×10^{-1}	2.1×10^{-1}	1.8×10^{-1}	1.6×10^{-1}	1.4×10^{-1}	1.3×10^{-1}
NE	J	3.3×10^{-1}	2.8×10^{-1}	2.6×10^{-1}	2.2×10^{-1}	2.0×10^{-1}	1.8×10^{-1}	1.6×10^{-1}
ENE	K	2.0×10^{-1}	1.7×10^{-1}	1.6×10^{-1}	1.4×10^{-1}	1.3×10^{-1}	1.1×10^{-1}	9.9×10^{-2}

TABLE XI-T

FISSION PRODUCTS AND ORGANS CONSIDERED FOR
INHALATION DOSE CALCULATIONS

Isotope	Atomic No.	Mass No.	Organ Affected						
			Bone	Kidney	Liver	Lung	Muscle	Thyroid	Total Body
Ga	31	72	x	x	x	x			x
As	33	77		x	x	x			x
Br	35	82				x			x
Rb	37	86			x	x	x		x
		87			x	x	x		x
Sr	38	89	x			x			x
		90	x			x			x
		91	x			x			x
		92	x			x			x
Y	39	90	x			x			x
		91m	x			x			x
		91	x			x			x
		92	x			x			x
		93	x			x			x
Zr	40	93	x	x	x	x			x
		95	x	x	x	x			x
		97	x	x	x	x			x
Nb	41	95	x	x	x	x			x
		97	x	x	x	x			x
Mo	42	99		x	x	x			x
Tc	43	99m	x	x	x	x			x
		99	x	x	x	x			x
Ru	44	103	x	x		x			x
		105	x	x		x			x
		106	x	x		x			x
Rh	45	105	x	x	x	x			x
Pd	46	107		x	x	x			x
Ag	47	111	x	x	x	x			x
Cd	48	115		x	x	x			x
In	49	115	x	x	x	x		x	x
Sn	50	125	x		x			x	x
Sb	51	125	x		x	x		x	x

TABLE XI-T (Continued)

FISSION PRODUCTS AND ORGANS CONSIDERED FOR
INHALATION DOSE CALCULATIONS

Isotope	Atomic No.	Mass No.	Organ Affected						
			Bone	Kidney	Liver	Lung	Muscle	Thyroid	Total Body
Te	52	127m	x	x	x	x		x	x
		127	x	x	x	x		x	x
		129m	x	x	x	x		x	x
		129	x	x	x	x		x	x
		131m	x	x	x	x			x
		132	x	x	x	x		x	x
I	53	129				x		x	x
		131				x		x	x
		132				x		x	x
		133				x		x	x
		134				x		x	x
Cs	55	135	x	x	x	x	x		x
		137	x	x	x	x	x		x
Ba	56	140	x	x	x	x	x		x
La	57	140	x		x	x			x
Ce	58	141	x	x	x	x			x
		143	x	x	x	x			x
		144	x	x	x	x			x
Pr	59	143	x	x	x	x			x
Nd	60	147	x	x	x	x			x
		149	x	x	x	x			x
Pm	61	147	x	x	x	x			x
Sm	62	153	x	x	x	x			x
Eu	63	155	x	x	x	x			x
Gd	64	159	x	x	x	x			x

Fig. XI-2 - Fission Product Activity as a Function of Time After Shutdown

Fig. XI-3 - Particulate Fission Product Activity as a Function of Time After Shutdown

Fig. XI-4 - Halogen Fission Product Activity as a Function of Time After Shutdown

Fig. XI-5 - Integrated Gamma Dose From the Containment Shell

SECTION XII

SAFETY ANALYSIS OF THE MAXIMUM CREDIBLE ACCIDENT

XII. SAFETY ANALYSIS OF THE MAXIMUM CREDIBLE ACCIDENT

A. Identification of the Maximum Credible Accident

The maximum credible accident (MCA) for the LOFT facility is conceived as a rupture of the primary coolant circuit after the reactor has been operating at full power for 800 consecutive hours. As a consequence of this rupture, it is assumed that both the primary and secondary system coolants are expelled to the containment volume. Because of the margin of safety incorporated in the system design to accommodate multiple blowdowns and the rigid inspection and acceptance tests required prior to nuclear operation, the probability of a system rupture is very low. In addition, this probability is further reduced by the relatively short operating time planned for the reactor.

Since the primary coolant system is not encased in biological shields the effect of unbalanced forces which could be imposed upon the system as a result of the rapid coolant expulsion must be considered. Such unbalances could severely stress the system and conceivably cause missiles to be generated. Although recent investigations and review of reported data⁽¹⁾⁽²⁾⁽³⁾, indicate that the generation of a missile with sufficient energy to completely penetrate the containment is highly improbable, the current design and supporting analysis is not sufficiently developed to allow the conclusions that such a penetration is incredible. For this reason, then, the rupture of the primary system is postulated to result in the generation of a missile which penetrates the containment barrier, leaving a hole of six square inches in cross sectional area. As the

-
- (1) "Final Hazards Summary Report, Humboldt Bay Power Plant, Unit Number 3", Pacific Gas and Electric Company, September 1, 1961.
 - (2) SRIA-2, "Containment of Fragments from a Runaway Reactor", N. R. Fabel, October 1, 1958.
 - (3) SRIA-6, "Containment of Fragments from a Runaway Reactor", R. W. White, N. B. Botsford, September 15, 1963.

pressure rises in the containment volume, the pressure reduction spray system will be automatically energized to provide a condensing mechanism for reduction of internal pressure in the building and thereby reduce the leakage. With a penetration area of six square inches, the internal pressure will be reduced to atmospheric in approximately 1.5 hours, as shown in Figure XII-1. An analysis of pressure reduction and containment leakage for various hole sizes indicates that for holes having an area considerably larger than six square inches, the pressure reduction is quite rapid and thus total fission product leakage to the atmosphere is less, since it takes approximately ten minutes before any fission products are released from the fuel. For the case of a penetration having a smaller cross sectional area, the pressure reduction is slower, but the total leakage through the smaller hole is still less than that for the postulated penetration. As an example, the curve of pressure decay inside the containment for a penetration of three square inches is shown in Figure XII-2, and the corresponding leakage rate curve in Figure XII-3.

It is further postulated in the identification of the maximum credible accident that the accident occurs during strong inversion conditions with the wind blowing to the northeast. This set of conditions results in the maximum off-site dose.

After the pressure in the containment building has been reduced to approximately atmospheric pressure, the containment atmosphere will be circulated through the exhaust filter system to reduce the fission product concentration. When lapse meteorological conditions are obtained with the wind blowing to the on-site direction, the remaining products in the containment atmosphere will be exhausted through the halogen and particulate removal filters to the stack.

Based upon this postulation of the maximum credible accident, the potential radiological exposures to both on-site personnel and to the general public are presented along with the assumptions used in arriving at these doses.

B. Radiological Analysis

The radiological analysis associated with the maximum credible accident is based upon the analytical models presented in Section XI-A as well as the following basic assumptions:

- (1) Rupture and expulsion of all coolant from the primary and secondary coolant systems.
- (2) Generation of a missile which penetrates the containment vessel and results in a six square inch opening to the external atmosphere.
- (3) Pressure reduction spray system is automatically energized at the first indication of a pressure increase in the containment shell.
- (4) Ten minutes after the coolant systems rupture, 100% of the noble gases, 50% of the bromines, 1% of the iodines, and 1% of the solids are released to the containment building (See Figures XII-4 and XII-5 for fission product activity as a function of decay time).
- (5) 50% of the particulates and halogens released to the containment shell are removed by washout.
- (6) Leakage rate through the six square inch hole is proportional to the square root of the pressure differential as shown in Figure XII-6.
- (7) Release occurs under strong inversion conditions with the wind blowing at 5 mph, $c_y = 0.256$, $c_z = 0.032$.
- (8) Deposition velocity for halogens is 2 cm/sec and particulates for 1 cm/sec.
- (9) Since the particle size and possibly chemical state of the fission products to be liberated from the fuel remain relatively unchanged, a reduction factor of five for the particulates and a halogens has been used for cloud depletion due to fallout for distances greater than six miles from LOFT.
- (10) An analysis of the fission product diffusion through the postulated aperture in the containment building after the internal pressure reaches zero psig indicates the additional contribution to the total dose is insignificant, and therefore, this factor is neglected.

The radiological dosages calculated by the use of the above assumptions and previously mentioned analytical models are presented in Table XII-A.

Personnel located in the immediate vicinity of the LOFT facility may be exposed to direct radiation from the fission products contained in the containment building. However, the levels to which personnel will be exposed will be no greater than those shown in Figure XI-5, and in all probability, will be less, due to the large leak rate and removal effects of the source term by the pressure reduction sprays.

C. Conclusions

The maximum credible accident for the LOFT facility is defined as the complete rupture of the primary and secondary coolant systems resulting in the generation of a missile which penetrates the containment building. The potential radiological dosages resulting from fission product release to the external atmosphere via the penetration are shown in Table XII-A.

The elapsed time between initial fission product release and the period of time required for the first portion of the released effluent to reach the nearest unshielded facility, TSF area, is approximately 25 minutes. Since evacuation of the TAN area would have been completed in this period of time, on-site dosages are not considered for the MCA.

The following exposures have been extrapolated from Table XII-A and are presented for the purpose of comparing the calculated dosages with the recommended values presented in the Federal Register (10 CFR Part 100).

Location Considered	Cloud Dose (rem)	Deposition (2 hr. Exposure) (rem)	Acceptable Whole Body Dose (rem)	Inhalation Dose (rem)	Acceptable Inhalation Dose (rem)
Nearest Inhabited Off-Site Area (7 miles east of LOFT)	22	0.31	25	9.0	300
Nearest town with Population > 3,000 (Idaho Falls, 40 miles southeast of LOFT)	1.1	0.01	25	0.55	300

Based upon the above radiological analysis and corresponding dosages, it is concluded that the radiation exposure to the off-site population will be below the values recommended in 10 CFR Part 100.

TABLE XII-A

RADIATION DOSES FOR MCA

Distance from LOFT (Meters)	Inhalation Dose to the						
	Total Body (rem)	Thyroid (rem)	Muscle (rem)	Lung (rem)	Liver (rem)	Kidney (rem)	Bone (rem)
1×10^2	3.9×10^3	5.4×10^4	7.0×10^0	3.8×10^4	3.2×10^3	3.6×10^3	2.2×10^4
5×10^2	3.5×10^2	4.8×10^3	6.3×10^{-1}	3.4×10^3	2.8×10^2	3.3×10^2	2.0×10^3
1×10^3	1.2×10^2	1.7×10^3	2.3×10^{-1}	1.2×10^3	1.0×10^2	1.1×10^2	7.0×10^2
5×10^3	1.1×10^1	1.5×10^2	2.0×10^{-2}	1.1×10^2	8.9×10^0	1.0×10^1	6.5×10^1
1×10^4	7.4×10^{-1}	1.1×10^1	1.4×10^{-3}	7.3×10^0	6.3×10^{-1}	7.2×10^{-1}	4.5×10^0
5×10^4	5.9×10^{-2}	7.9×10^{-1}	1.3×10^{-4}	5.7×10^{-1}	5.6×10^{-2}	6.2×10^{-2}	3.9×10^{-1}
1×10^5	1.9×10^{-2}	2.5×10^{-1}	4.4×10^{-5}	2.0×10^{-1}	2.0×10^{-2}	2.1×10^{-2}	1.4×10^{-1}

Distance from LOFT (meters)	External Gamma Dose (rem)	
	Cloud Gamma	Deposition Gamma (2 hr. Exposure)
1×10^2	5.0×10^4	2.5×10^3
5×10^2	4.2×10^3	2.2×10^2
1×10^3	1.4×10^3	7.5×10^1
5×10^3	8.3×10^1	6.2×10^0
1×10^4	2.6×10^1	3.8×10^{-1}
5×10^4	1.6×10^0	1.3×10^{-2}
1×10^5	4.2×10^{-1}	3.0×10^{-3}

Fig. XII-1 - Pressure Decay in the Containment Building With Six Square Inch Penetration

Fig. XII-2 - Pressure Decay in the Containment Building With Three Square Inch Penetration

Fig. XII-3 - Containment Leakage Through Three Square Inch Penetration

Fig. XII-4 - Fission Product Activity as a Function of Time After Shutdown

Fig. XII-5 - Halogen Fission Product Activity as a Function of Time After Shutdown

Fig. XII-6 - Containment Leakage Through Six Square Inch Penetration

SECTION XIII

POST-TEST OPERATIONAL SAFETY CONSIDERATIONS

XIII. POST-TEST OPERATIONAL SAFETY CONSIDERATIONS

The safety analysis of the post-test operations is primarily concerned with the hazards associated with the removal of the reactor primary system from the reactor building and its transportation to the hot shop for detailed examination. The radiological hazards that have been evaluated for these operations include the direct radiation from the reactor core to the personnel in the shielded locomotive during removal of the primary system, the effect on the operation of other TAN facilities from direct radiation or contamination during transport of the primary system to the hot shop, and the possibility of secondary criticality during and following these operations.

A. Radiological Safety Analysis

1. Direct Radiation

a. Reactor Removal - LOFT Facility

The radiation dose to personnel located in the shielded locomotive during the removal of the reactor primary system has been evaluated. This analysis is based on a decay time of 5 days following the loss of coolant test and the conservative assumption that the core has melted through the reactor vessel bottom head and is located in the molten fuel container provided under the bottom of the reactor vessel. The total radiation dose to the locomotive personnel is calculated to be 0.7 mrem/hr from this source.

b. Reactor Transport to Examination Area

The gamma dose rate as a function of distance from the railroad dolly while in transit to the hot shop is shown in Figure XIII-1. This curve is based on the assumption that the reactor fuel is contained in the reactor vessel which is shielded with a primary shield containing 3 inches of lead and 45 inches of water. If during the loss of coolant test the fuel penetrates the reactor vessel and is collected in the molten fuel container, the levels shown in the figure will be increased by approximately a factor of 3×10^4 .

During the transport of the test package to the hot shop, personnel will not be allowed within 300 feet of the railroad tracks. In the event there is an indication that the radioactive fuel has entered the unshielded molten fuel catch basin, this exclusion area will be extended to 2000 feet. In either case it may be necessary to evacuate personnel from the unshielded areas in Building 607 when the package approaches the building. Access to the exclusion area will not be permitted until it has been cleared by the Health Physicists in charge.

c. Examination Area

The gamma radiation at the external surface of the hot shop walls with the LOFT core inside is calculated to be less than 1 mrem/hr.

2. Contamination

a. Reactor Removal

To reduce the spread of contamination to a minimum during transportation of the nuclear primary system to the examination area, the interior walls of the containment building and the exterior of the test package will be decontaminated with water. Access to the containment building will be prohibited until the atmosphere has been recirculated through the halogen and particulate removal filtering system for a sufficient period of time to reduce the activity levels to acceptable limits for release to the atmosphere. Subsequent to purging the building volume to the atmosphere and establishment of residual activity the railroad door will be opened to permit entry of the shielded locomotive and the Roving Manipulator System (MOBOT). The reactor primary coolant system will be isolated by valves provided upstream of the rupture mechanism and the MOBOT deployed for close television examination of the primary system. Preliminary cleanup of loose material lying on the floor or reactor dolly tracks will be accomplished with the MOBOT.

b. Transportation to the Examination Area

Although the fine water sprays provided prior to movement of the reactor primary system will remove much of the externally deposited fission products, contamination of the railroad right-of-way during transport is a possibility. Therefore, a distance of 300 feet on each side of the tracks will be an exclusion area until it has been surveyed by Health Physics and decontaminated. To further reduce the possibility of contaminating the right-of-way, the reactor package will not be moved during adverse weather such as wind or rain.

c. Examination Area

Contamination of the hot shop from residual fission products and plutonium during disassembly and study of the reactor primary system is anticipated. However, the hot shop is designed to handle high levels of radiation and contamination and these operations will not present an uncontrollable hazard. See Section VII-B on the capability of the hot shop to handle relatively unshielded experimental nuclear assemblies.

B. Secondary Criticality

1. Removal and Transportation to Examination Area

At the conclusion of the loss of coolant test any water remaining in the primary system will be drained. With the water moderator removed a critical configuration of the core is not possible. However, neutron detectors will be provided in the containment building, the hot shop, and on the locomotive to monitor the reactor core before, during, and after the transfer operation.

2. Examination Area

To preclude a secondary criticality during the reactor disassembly operations in the hot shop, all water will be excluded from the area. The reactor vessel and primary system will not be flooded for decontamination purposes until the entire core, intact and melted, has

been removed. Detailed procedures will be written for the safe dis-assembly of the core and handling of the intact and melted fuel during the post-test examination.

Fig. XIII-1 - Direct Radiation From the Reactor Vessel

SECTION XIV

CONCLUSIONS

XIV. CONCLUSIONS

The LOFT reactor experiment and facility are designed specifically for conducting a test program which culminates in a modeling of a loss of coolant accident for the general class of pressurized-water, UO_2 fueled, power reactors. This accident results in violation of the integrity of two of the three containment barriers. For this reason, emphasis in this safety analysis has been placed on evaluating the ability of the containment building to maintain the fission product release to the environment within acceptable levels. In the case of LOFT, the limits of radiological exposure resulting from such a release are dependent upon whether the release is planned or accidental. The two conditions for which regulations or limits must apply are as follows: one for the test conditions which are dictated by the regulations⁽¹⁾ covering normal or planned releases to the atmosphere and one for the postulated MCA which is dictated by the regulations⁽²⁾ covering accidental releases to the atmosphere.

The radiological doses to on-site personnel and the general public during the loss of coolant test are based on realistic assumptions, supported by experimental evidence, of the fission product behavior and meteorological conditions during the five day test period. This approach utilizes the experimental fission product release data⁽³⁾ to determine the radioiodine release from the fuel as a function of time after coolant blowdown and plateout data⁽⁴⁾ to determine the plateout in the containment building. Credit is also taken for the diffusion of radioiodine from the reactor vessel to the containment building. The analysis also employs the conservative approach that fission product release is proportional to the square root of the pressure and credit is not taken for cloud depletion due to fallout. More detailed discussions of the assumptions are presented in Section X.

-
- (1) AEC 0524-01 "Standards for Radiation Protection".
 - (2) Code of Federal Regulations, 10CFR 100, USAEC.
 - (3) ORNL-3319, "Nuclear Safety Program Semiannual Report for the Period Ending June 30, 1962", August 24, 1962.
 - (4) J. F. Croft, et al, "Experiments of Surface Deposition of Airborne Iodine of High Concentration", AEEW-R 265, June 1963.

From this analysis the following potential doses during the test are obtained:

<u>Location</u>	<u>Whole Body Dose (rem)</u>	<u>Thyroid Dose</u>	
		<u>Inhalation (rem)</u>	<u>Ingestion (rem)</u>
TAN Complex	1.7×10^{-2}	1.6	0
NRF	9×10^{-4}	4×10^{-2}	0
Site Boundary	8×10^{-4}	0.43*	2.0*

*Doses to a Child

If credit is taken for the affect of monitored penetrations, the calculated doses will be considerably below the values presented. In addition, radiological measurements will be made both inside and external to the containment building throughout the test period. If it appears that the permissible off-site exposure may be exceeded, the test will be immediately terminated. In addition, certain additional procedures are available to insure that permissible exposures are not exceeded. These include the purchase of milk, limitations on cattle grazing, and evacuation of people in the downwind areas.

Direct radiation from fission products contained in the containment building is calculated to be negligible at the TAN complex which is the closest area to the LOFT site.

It is therefore concluded that the loss of coolant test can be conducted without undue hazard to on-site personnel or to the general public.

The maximum credible accident involving the LOFT program is identified as the loss of coolant accident accompanied by a missile which results in a six square inch penetration of the containment building. In addition, the assumption is made that the accident occurs during adverse meteorological conditions (strong inversions) with the wind blowing toward the nearest site boundary. It is also assumed that at the time of the accident the pressure reduction sprays are turned on and the containment pressure is reduced to atmospheric pressure in approximately 1.5 hours. Further details on the analytical models and specific assumptions used are discussed in Section XII.

The potential radiological doses to the general public at the nearest site boundary in the event of this postulated MCA are as follows:

<u>Location</u>	<u>Whole Body Dose*</u> <u>(rem)</u>	<u>Thyroid Dose Inhalation</u> <u>(iodine)(rem)</u>
Site Boundary (nearest)	22	9
NRF	3.9	1.6

*Whole body dose is the sum of the direct gamma radiation from the cloud during its passage and from a two hour exposure to fission products deposited on the ground.

It is postulated that approximately 25 minutes is required following the accident to provide a significant fission product leakage from the containment building. In this period of time personnel at all TAN facilities can be evacuated, thus eliminating excessive exposures to on-site personnel.

The post-test operation, although treated briefly in this analysis, indicates that with proper administrative control the transport and examination of the experimental assembly can be accomplished without significant radiological hazard.

On the basis of this preliminary safety analysis, it is concluded that the LOFT facility as presently conceived can be constructed at the TAN site of the NRTS for the express purpose of conducting a loss of coolant test program. It is further concluded that the experimental assembly, in conjunction with this facility, can be operated at power for the required period of operation without undue risk to on-site personnel or to the general public.