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FEASIBILITY AND CONCEPTUAL DESIGN
FOR
THE STEP LOSS OF COOLANT FACILITY

MASTER

T. R. Wilson, O. M. Hauge, and G. B. Matheney

**PHILLIPS
PETROLEUM
COMPANY**

ATOMIC ENERGY DIVISION

NATIONAL REACTOR TESTING STATION

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Atomic Energy Division

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SUMMARY

The Safety Test Engineering Program has been established by the Commission as part of its nuclear safety program for the purpose of conducting large scale engineering safety tests which will demonstrate and provide information regarding the behavior and predictability of complete nuclear systems under actual accident conditions. As a part of this program, a series of safety tests will initially be conducted to provide information and data and to demonstrate the integrated effects of a loss of coolant accident in a typical pressurized water power reactor.

The purpose of this report is to present a summary of studies conducted on the loss of coolant accident and propose an experimental safety program. The various phenomena involved in the loss of coolant accident, related research and development programs, assumptions currently used to predict the various physical phenomena and the general approach to be used in conducting the safety tests are discussed.

In order to accomplish the loss of coolant experimental safety program, a dry containment test facility is proposed for construction at the Test Area North of the National Reactor Testing Station in Idaho. The site selection utilizes existing support facilities ideally suited for performing nuclear safety tests requiring experiment assembly areas and post-test analytical examination of the irradiated nuclear components.

FEASIBILITY AND CONCEPTUAL DESIGN
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I. INTRODUCTION

A. Purpose

The Safety Test Engineering Program (STEP) has been established by the 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 a nuclear accident. This effort is part of the safety program sponsored by the Nuclear Safety Engineering and Test Branch of the Division of Reactor Development. Phillips Petroleum Company has been assigned the responsibility for conducting the experimental program which will be carried out at Test Area North (TAN) of the National Reactor Testing Station using existing facilities and new facilities to be constructed.

Because of prominence of water-cooled and -moderated reactors in the United States power program, the initial experimental investigations will be directed toward providing safety information regarding the major problems of this type reactor, in particular the loss of coolant accident. Preliminary studies^(1,2) have investigated the feasibility of conducting these tests on a typical water-cooled power reactor. A test program has been formulated and the design criteria for the test facility have been established. The purpose of this report is to present the information developed by these studies.

B. Review of Reactor Hazards and Safety Analysis

1. General

The probability and possible consequences of major reactor accidents have been subject of extensive concern and study since the

origin of the nuclear industry. These studies⁽³⁾ indicate a low probability of occurrence of severe accidents in nuclear power reactor plants that are constructed and operated in accordance with existing practices. Nevertheless, the possibility exists that major releases of fission products from a nuclear reactor could occur and that a serious threat to the health and safety of people over large areas could ensue.

In order to minimize the potential hazards of a major reactor accident, the approach in reactor design and safety evaluation has necessarily been to employ wide margins of safety. This approach recognizes that the magnitude of the accident probability is not known and that the reactor accident consequences cannot be confidently predicted except for an upper limit. The competitive position of nuclear power and its growth are affected by plant siting considerations and safety features of plant design. In this regard the basic questions facing the nuclear industry as the reactor designer and the AEC as the safety evaluator involve the estimation of the degree of safety achieved by the engineering barriers provided in the reactor plant design and the estimation of radiological hazards subsequent to a fission product release.

2. Reactor Accidents

Before the general course to be taken by further nuclear safety research programs can be formulated, the kinds of nuclear accidents, the manner in which accidents can occur, and the information needed to estimate the accident consequences must be examined. The way in which most reactors can malfunction and lead to the escape of fission products may be classified as follows:

- (1) super-critical nuclear excursion or nuclear runaway, and
- (2) meltdown of reactor components, even with the chain reaction shutdown, from the delayed heat produced by radioactive fission products and the sensible heat contained in the nuclear fuel and coolant system.

Accidents of the first type involve a reactivity disturbance and may be caused by such things as equipment malfunctions, operator error, instability, etc. In accidents of this type, events are determined by the response characteristics of the nuclear system which are a combination of both non-nuclear features and inherent nuclear characteristics. In the most extreme nuclear runaway, the inherent nuclear characteristics will determine the results. During an extreme excursion, partial meltdown of the core may occur accompanied by chemical reactions or steam explosions of sufficient magnitude to break the primary coolant system, thereby creating the possibility of releasing large quantities of fission products to the surrounding structure.

Accidents of the second type include the loss of coolant. Equipment failures resulting in rupture of the primary coolant system followed by depressurization and loss of coolant can lead to melting of the fuel elements and the release and transport of fission products into the surrounding structure. Melting of the fuel may be accompanied by chemical reactions. The loss of coolant accident is postulated to be the "maximum credible" accident for most water-cooled and -moderated power reactor systems, for example, Shippingport⁽⁴⁾, Yankee⁽⁵⁾, Big Rock Point⁽⁶⁾, Bonus⁽⁷⁾, and Dresden⁽⁸⁾. The accident sequence is similar for all pressurized and boiling water reactor systems; however, the predicted potential hazards are generally estimated to be somewhat greater for a large pressurized water system than for a boiling water reactor of equivalent size because greater potential energy is stored in the pressured water than is stored in an equivalent volume of steam.

A study of the chain of events leading to and accompanying a major accident immediately discloses the complexity of estimating or calculating the consequences of an accident.

Nuclear safety research has been aimed at providing information and data of importance for understanding each of the many phenomena having a bearing on safety. These studies have contributed significantly to understanding of safety problems. However, the present information does not permit reactor designers or safety evaluators to

estimate with the desired degree of accuracy the accident consequences and the safeguards required or the degree of safety achieved by various engineered safety devices.

Criteria⁽⁹⁾ for reactor siting and a guide⁽¹⁰⁾ for safety analysis have been established. The lack of experimental information has caused conservative assumptions to be used in this document. A review of these assumptions is necessary to assess the reactor safety program, and to delineate problem areas presented in the guide. The assumptions are as follows:

- (1) pressurized water type reactor,
- (2) release of 100 percent of noble gases, 50 percent of halogens, and 1 percent of the solids in the fission product inventory,
- (3) 50 percent of the iodines released to the reactor building remain available for release to the outside environment,
- (4) release of available (airborne) radioactivity from the reactor building to the environment occurs at a constant rate of 0.1 percent per day,
- (5) atmospheric dispersion of material occurs under inversion type weather conditions, and a constant wind direction prevails for the duration of the fission product leakage,
- (6) cloud depletion due to particulate fallout will not occur, and
- (7) credit is not taken for applicable shielding in calculating the direct gamma dose.

The assumptions made in preparing the guide are recognized to provide conservative results in determining suitable reactor siting. Furthermore, an acknowledged possibility exists for reducing the conservatism in these assumptions. Estimated reductions in the assumed variables from 500 to 10^6 may be attainable when substantiated by sufficient reliable experimental information; thereby, the siting distance requirements could be relaxed. For example, reduction factors

of various magnitudes can be expected in the following areas: quantity of fission products released in an accident, amount of iodine available for leakage (such as removal by absorption, settling, washdown, filtration, etc.), the leakage rate (by reduction of pressure in the reactor building), calculated direct gamma dose, and cloud depletion. The safety program developed herein is designed to provide a reduction in the uncertainties of the criteria and guide. Attainment of this information can also be expected to reduce capital and operating costs for providing adequate safety in nuclear power plants.

C. Approach to Nuclear Safety

The broad objectives of the Commission's nuclear safety program are to provide information which will assist reactor designers to minimize fission product release problems, and which will assist safety evaluators to assess the degree of safety achieved. Attainment of the objectives requires numerous and varied safety programs including the STEP program which will integrate many of the features of existing nuclear safety experiments. In examining what information has been provided and what information remains to be obtained, it will be useful to describe some general characteristics of accidents.

For this description, it is convenient to view a reactor accident as having three phases: initiation, response, and consequences⁽¹¹⁾. Initiation may be defined as mechanisms which may bring about abnormal conditions such as component or subsystem failure, operator error, inherent instability, etc. Response is defined as the behavior of the reactor system as a result of the occurrence of an abnormal condition. The overall response may be influenced by both the nuclear and non-nuclear characteristics of the system, that is, by both the core and associated subsystems such as control system, coolant system, etc. In the consequences phase important factors are the disintegration of the fuel material by nuclear heating and the dispersal of radioactive materials. The consequences of an accident may also be influenced by both the nuclear and non-nuclear characteristics of the reactor plant and by specific appendages such as containment and safety devices.

Specific safety programs needed and their objectives when classified in this way are not readily apparent without further classification as to the type of information needed. As in any industrial research program aimed at discovering, developing and testing a complex process, fundamental laboratory research, semi-scale component or subsystem studies and full-scale prototype tests are frequently required. In the case of nuclear safety research, because of the many classes of reactors and the variations in nuclear and non-nuclear characteristics within a given class of reactors, the type of information needed may be divided into four classes. These are: fundamental (basic) research, component or semi-scale subsystem studies, class prototype tests and reactor prototype tests.

Fundamental research provides information aimed at understanding the physical laws or phenomena involved in a process and is approached by conducting experiments which separate the effects of each variable. Examples of basic research are: detector research and development, fuel materials research, chemical reaction research, criticality studies, dynamics of bubble formation, etc. Data provided by basic research are generally applicable to many nuclear systems and are ultimately essential to a complete understanding of a process and to development of analytical models for predicting the behavior under a wide range of environmental conditions.

Component or semi-scale subsystem studies provide information and data regarding complete components or subsystems and frequently involve investigation of the combined effect of several independent variables on the overall behavior of component or system. Examples are: in-pile chemical reaction studies, fuel element hydraulic and heat transfer studies, semi-scale containment studies, control mechanisms studies, etc. Information provided is generally applicable to several classes of reactors, but may be confined to a single class or reactor prototype. The information provided by studies of this type is essential to verification of the extrapolability of basic research data and the accuracy of analytical models.

Class prototype and reactor prototype tests provide quantitative and qualitative information on the integral behavior of reactor plants

under actual operating (accident) conditions and assist in evaluating the performance or combined effect of many physical phenomena occurring in each of the subsystems. Class prototype tests are defined here as tests conducted on reactor systems typical of a class of reactors such as pressurized water, sodium-cooled, gas-cooled, etc., while reactor prototype tests are tests conducted on a reactor system identical in all respects to a proposed or existing reactor. Examples of these classes of safety research are: class prototype--Borax I Destructive Test, SPERT I Destructive Test; reactor prototype--SNAP 2/10A Safety Tests.

Large-scale engineering or prototype tests are frequently employed by both industry and government in developing complex processes or systems, for example, ballistic missiles and aircraft. Such tests are necessary because the complex interaction of the many phenomena cannot be confidently represented or interpreted for semi-scale tests. Also, analytical models formulated for the individual phenomena frequently involve empirical parameters which cannot be scaled up for full-scale situations by simple scaling laws. In addition, when time is a major factor in the development of a complex system, both basic research and subsystems testing are minimized in favor of full-scale prototype tests, although the long range benefits of basic and subsystem research required for complete understanding are readily apparent. The information provided by full-scale testing is therefore essential to verification of the extrapolability of basic research and semi-scale subsystems information and to evaluation of the overall behavior of complex systems.

If the same philosophy is applied to nuclear safety research as applied to similar complex industrial and government research and development programs, large-scale engineering tests are needed to complement previous and proposed research and development programs. The nuclear safety research program has thus far provided valuable basic research and semi-scale subsystem safety test information on plate type, heterogeneous, water-moderated reactor systems. Other programs planned or in progress such as containment vessel investigations and fission product release studies will assist in estimating and evaluating the consequences of some aspects of an accident involving

water-moderated reactor systems. Large engineering safety tests supplementing these research efforts are needed to demonstrate the overall consequences of a major reactor accident and to provide data needed to verify methods used to assess safety and confidently predict the behavior of a pressurized water reactor system under actual accident conditions.

Practical considerations have oriented the nuclear safety research programs to date along lines of investigating reactor systems of particular, immediate importance to the reactor industry such as water-cooled and -moderated reactors and fast reactors. Other reactor concepts of interest such as gas-cooled and organic-cooled reactors have to date necessarily received somewhat lesser emphasis in the nuclear safety research because of their slower development in the United States.

The prominence of water-cooled and -moderated reactors in the United States power program dictates that the initial engineering and safety test be conducted on this class of reactors. It is generally agreed that the loss of coolant accident will result in the greatest radiological hazard for this class of reactors. Therefore, a review has been conducted of the hazards accompanying the loss of coolant accident in a pressurized water reactor system, the current techniques used in safety analysis, and the results of related nuclear safety programs. The results of the review are summarized in the following section.

II. REVIEW OF THE LOSS OF COOLANT ACCIDENT

A. Safety Analysis and Related Nuclear Safety Programs

Loss of coolant from a water-cooled nuclear system was separated earlier into the initiation phase, the response phase and the consequences phase of the accident. Included in the initiation phase are the primary system rupture and coolant blowdown stages. The consequences phase consists of the core meltdown, and fission product release and transport stages. Each of these stages involves many individual processes or phenomena such as fluid mechanics, thermodynamics, heat transfer, mass transport, etc., which may take place simultaneously and thus interact. The assumptions and postulations generally employed to assess the hazards attendant to a loss of coolant are reviewed in this section.

1. Accident Initiation and Coolant Blowdown

Initiation of a loss of coolant accident is generally assumed to occur by an instantaneous offset shear (brittle fracture) of either an inlet or outlet primary coolant pipe. Rupture of the pipe causes rapid expulsion of two-phase coolant from both ends of the pipe, with the core bared of water in less than 30 seconds for most reactor systems.

Assumptions are required to estimate the coolant expulsion rate and the time required to uncover the core, since precise calculational methods for determining the two-phase flow rate are not available. The series of events which occur following the rupture and which may affect the expulsion rate are as follows. After rupture occurs, the coolant system remains solid with liquid while coolant pressure decreases to the saturation pressure which corresponds to the bulk coolant temperature. This subcooled blowdown period is of short duration, generally less than one second. At this time, a reactor scram is initiated by means of either a low system pressure signal or a high containment pressure signal. After reaching the saturation pressure, the escaping coolant becomes a two-phase mixture of steam and water. The quality of the steam in the expulsion stream will continue to increase as the

system pressure decreases until only pure steam emerges from the system. The steam expulsion will continue until a pressure balance is reached between the primary coolant system and the containment volume surrounding the coolant system. Because of the continuous change in the steam quality and system pressure and the possible existence of a meta-stable condition in the system, calculations of the expulsion rate are subject to considerable error.

An important phenomenon associated with the loss of coolant is the pressure buildup inside the containment vessel. In order to estimate the pressure transient, assumptions must be made pertaining to the decay heat rate of the core, latent heat distribution, convection heat transfer, expulsion flow rate, steam condensation, etc. Many of the parameters affecting these processes are generally unknown or difficult to predict with a reasonable degree of confidence in the absence of experimental data.

2. Core Meltdown

Several analytical models have been developed to calculate the extent of core (fuel and clad) melting following a loss of coolant. Frequently, models assume that the accident occurs at the end of core life when the fission product inventory and the decay heat flux are at a maximum.

Since the parameters affecting the heat transfer processes are uncertain during and after coolant blowdown, most analytical models are based on the assumption that no heat is lost from the core after the primary coolant has been ejected from the system. With this assumption, the entire core, clad and fuel would melt. Because of more rigorous assumptions, other models predict that only a fraction of the clad and fuel will melt.

Safety devices are frequently designed and incorporated into the system for removal of the decay heat from the core following a loss of coolant which will prevent or reduce the magnitude of fuel melting. Often limited credit, or no credit, is obtained for the heat removal capabilities because of the possibility that these safety devices may become inoperative at the time of the accident.

3. Fission Product Release and Transport

The degree of confidence with which the radiological hazards associated with the maximum credible accident can be calculated is dependent on knowledge of the fission product inventory at the time of the postulated accident, the quantity and type of fission products released to the containment vessel, and the quantity and type of fission products leaking from the containment system to the atmosphere. An accurate knowledge of the meteorological history at the reactor site must also be available.

The fission product inventory as a function of operating time for a given reactor can generally be determined. Frequently, a fission product inventory based on the normal operating reactor power level at the end of core life is used in safety analysis. Since the meteorological history at many proposed reactor sites is usually not well known, conservative conditions are assumed.

Because of the lack of sufficient experimental data, the major source of uncertainty in determining the consequences of a loss of coolant accident is the degree of confidence with which the magnitude of the fission product release to the containment vessel can be determined. The range of assumptions made for the fission product release for most safety analysis work is as follows:

	<u>Halogens</u> <u>(I, Br)</u>	<u>Noble Gases</u> <u>(Xe, Kr)</u>	<u>Solids</u>
Release to containment vessel	8.6%-100%	8.6%-100%	0.1%-1%

Since issuance of the site criteria and guide, the assumptions employed in these documents have been widely followed. The site criteria and guide are based on the information and data developed by the nuclear safety program. The results of several of the safety programs related to the loss of coolant accident are summarized below.

B. Related Safety Programs

The experimental work conducted thus far as part of the Commission's nuclear safety program is directed primarily at providing information which will assist in understanding the various processes and phenomena that are involved in a loss of coolant accident. These safety programs were concerned with the following events:

- (1) coolant blowdown,
- (2) core meltdown,
- (3) fission product release, and
- (4) containment of coolant and fission products.

The purpose of this section is to describe the results of the experiments and to review the effect that these experiments had in: (1) understanding the loss of coolant accident, and (2) assessing reactor hazards.

1. Coolant Blowdown

The experiments involving coolant blowdown were generally conducted for the purpose of establishing the actual flow model which occurred during the expulsion of coolant from the reactor system. With the establishment of the actual flow model, analytical determinations could be made of flow rates and blowdown times, thereby reducing the design pressure requirements for containment.

A series of tests⁽¹²⁾ were conducted in 1959 to demonstrate that the specific application of the mathematical model used to describe the reactor blowdown is conservative. These tests were conducted in a systematic manner to provide information regarding the effects which the following variables have on expulsion flow rate:

- (1) coolant temperatures--tests were performed at coolant temperatures of 400°F, 500°F, and 600°F,
- (2) rupture size--large and small ruptures were simulated using six different sizes of orifices, and

- (3) rupture position--orifices were positioned at top and bottom of the vessel to simulate ruptures of inlet and outlet piping.

No report was found which evaluated the results of these tests. However, these tests provided valuable data with regard to relationship of coolant temperatures, rupture size, and rupture position on the blowdown process.

Another series of tests^(13,14) were performed to establish whether conservatism in design and hazards evaluation could be maintained if flashing flow were assumed. These tests constituted the discharge of water through orifices of the same diameter, but of different lengths. The results of these tests indicated the following:

- (1) The actual flow rates were from 40 to 60 percent lower, depending on the length of orifice, than the rates predicted by the Burnell correlation which is based on a flashing flow model.
- (2) The observed flow rates were lower (by as much as 50 percent for the longest orifice) in all cases than would be predicted using a frictionless flashing flow model based on thermodynamic equilibrium, a flow coefficient of 0.60, and the measured subcooling at the orifice entrance, but were greater than those flow rates predicted assuming saturated conditions at the orifice inlet.

Analysis of these results indicated that the calculated flow rates are conservative by as much as 80 to 100 percent.

Another experimental program⁽¹⁵⁾ was conducted to evaluate and demonstrate a different type of containment concept. These tests, in addition to their primary purpose, provided information concerning flow rates during a loss of coolant accident and which substantiated the conclusions of the previous test. The observed flow rates varied from 80 percent (for a slow leak) to 40 percent (large rupture) of the flow rates predicted from calculations using a flow coefficient of 0.61 and the density of saturated liquid.

The above tests indicate that the prediction of flow rates by the existing acceptable analytical techniques are conservative estimates.

Reasonable proof can be provided by a full scale test consisting of a reactor system subjected to an actual large rupture accident.

2. Fuel Meltdown and Chemical Reactions

The review of the experimental work accomplished revealed that limited programs have been conducted to investigate the heat transfer characteristics of a bare core located in a reactor vessel following a loss of coolant incident. However, some effort had been made with respect to fuel melting characteristics and a much more significant and comprehensive undertaking had been made to examine metal-water reactions of the fuel and clad material with the water vapor in the systems.

a. Core Meltdown

An experiment⁽¹⁶⁾ was made in 1958 to evaluate the magnitude of fission product escape and distribution under conditions of simulated core meltdown. In addition, the mode of meltdown was to be studied which would include the quantity of melt produced, the characteristic formation of the melt and the amount of hydrogen produced from the zirconium-water reaction. Full-scale plate-type fuel sub-assemblies of zirconium clad uranium were used, and an induction heating device was used to simulate the predicted decay heat. These tests showed that sections of the core could be deposited in the bottom of the reactor vessel either from melting or from the metal-water reaction. With the center of the subassembly in a molten condition and the outer walls at less than the melting temperature, the tests demonstrated the possibility of completely plugging the fuel plate channels. Furthermore, when melting occurs, the molten metal was released from the subassembly in the form of streams rather than droplets. The value of these results was reduced since the heating was terminated when melting started instead of perpetuating as in an actual reactor decay heat situation.

The data showed that at least 20 percent by weight of the core could be consumed in the metal-water reaction and that the value could be higher in an actual reactor incident. This reaction would result in

the production of copious quantities of hydrogen which might cause localized concentrations leading to a potential hydrogen explosion.

b. Metal-Water Reactions

Experimental studies have been conducted in the chemical-reaction field to investigate the characteristics of metal-water reactions. The major portion of the in-pile experimental work has been done at the TREAT reactor facility. The in-pile experiments^(17, 18) conducted in this facility involve the melting of fuel specimens in water and measurement of the quantity of hydrogen produced by the metal-water reaction. Both cylindrical and rod type fuel specimens have been tested. The type of specimens include metal core (clad and unclad), oxide core (metal), and cermet core (clad and unclad).

Analytical studies pertaining to the kinetics of metal-water reactions are presented in references 19, 20, and 21.

Work has been done in this field to allow a prediction of the metal-water reactions that may be associated with a core meltdown. Although metal-water reactions have been investigated, the mechanisms and effects--such as, kinetic reactions, energy release, temperature, and rate of formation--are not thoroughly understood. Furthermore, the progression of decay heat and associated core melting are transient parameters which preclude an accurate prediction of the magnitude of core melting and metal-water reactions during a loss of coolant accident.

3. Fission Product Release

The release of fission products from nuclear fuels has received considerable attention. Several experimental nuclear safety programs^(16, 22, 23, 24, 25) have been conducted or are in progress to provide information of importance for predicting the fission product release under reactor accident conditions. The programs are designed to investigate the effects of:

- (1) release mechanisms (high-temperature diffusion, oxidation, melting and vaporization),
- (2) atmosphere on release,

- (3) fuel burnup,
- (4) quantity of fuel,
- (5) center melting (nuclear heating) as compared to external melting,
- (6) environment: metal cladding, fuel cluster, etc.

An experimental program⁽¹⁶⁾ was accomplished with induction heating of zirconium clad fuel plates in a meltdown chamber which simulated a reactor vessel. Results were as follows:

- (1) Release of xenon from the subassembly ranged from 42 to 58 percent and was relatively independent of the meltdown characteristics.
- (2) Iodine release from the subassembly ranged from 1.3 to 7.6 percent under conditions of low total heat input and from 7.9 to 25.8 percent under conditions of high total heat input. However, iodine release from the meltdown chamber, which would simulate the reactor vessel, was much lower and ranged from 0.5 to 2.0 percent of the total in the subassembly (i.e., approximately 15 percent of the fission products released escaped from the chamber).
- (3) The quantity of other fission products released was lower and ranged from 0.04 to 2.0 percent depending on the isotope and heat input rate.

The experimenters concluded that, if the core should remain in a molten state, larger quantities of fission products may be released in an actual incident than indicated by experimental results. This conclusion was made from the observation that the quantity of fission products released was dependent upon the time at temperature and degree of melting.

Other experiments⁽²²⁾ were conducted to investigate the percentage of fission products release from UO_2 fuel as a function of temperature, fuel burnup, and atmosphere. The results revealed the following releases:

Irradiation Level (MWD/Ton)	Temperature (°C)	Percent of Individual Fission Products Released		
		Xe-Ke	I	Cs
Trace	1400°	0.8	4.0	0.02
4,000	1400°C	6.1	23.0	21.0
Trace	Molten Fuel	99.5	89.7	91.3
11,000	Molten Fuel	100.0	99.7	98.7

From these results it appears that at low temperature the release is greatly dependent on fuel burnup and the release from molten fuel is relatively independent of fuel burnup.

The observed concentrations of fission products released from molten fuel are similar to the guidelines used in safety analysis of power and test reactor sites set down in TID-14844. These guidelines assume a pressurized-water type reactor for which the maximum credible accident will release 100 percent of the noble gases (xenon and krypton), 50 percent of the halogens (iodine and bromine) and 1.0 percent of the solids in the fission product inventory. This release represents approximately 15 percent of the gross fission product activity.

Experiments⁽²⁵⁾, conducted in Great Britain to investigate the behavior of iodine released in a reactor containment vessel, demonstrated that from 40 to 80 percent of the iodine released from the reactor system was deposited on surfaces inside the containment vessel. These experimental data are in general agreement with the assumption of TID-14844.

4. Containment

To understand the effectiveness of the reactor containment, several experiments were run to investigate the ability of the containment devices to reduce pressures inside the containment vessel and contain the fission products released from a simulated accident.

a. Fission Product Containment

Some early tests⁽²⁶⁾ were made to simulate the fission product retention and leakage characteristics of a PWR-type containment structure. Later tests⁽²⁵⁾ were conducted to investigate the behavior

of iodine in a reactor containment shell. These experiments demonstrated the release, escape and deposition characteristics of fission products. However, the amount of fission product leakage from the containment shell after the fuel meltdown, of importance to reactor siting, was not sufficiently demonstrated to allow conclusions to be made. Thus, in order to attain maximum credit for the containment, additional tests involving a full-scale reactor accident are suggested to further evaluate the fission product retention qualities of a representative containment vessel.

b. Pressure Reduction Devices

The development and testing of devices to reduce the pressure in the containment building began initially in 1956 with an experiment⁽²⁷⁾ designed to investigate the effects of heat absorption in the containment vessel and the influence of heat dissipation on the pressure time relationship inside the vessel. These tests demonstrated that the heat loss through a bare steel shell would result in reducing the calculated maximum pressure obtained within the vessel (assuming no heat transfer during the loss of coolant) by approximately 60 percent. If the same steel shell were lined with concrete, the pressure would be reduced by approximately 30 percent. These tests also indicate that the pressure rise would practically be eliminated by releasing the coolant into a water bath or through a water spray.

Other tests⁽²⁸⁾ were performed whereby the coolant was injected into a water pool. These tests demonstrated that no appreciable pressure rise would occur if the steam and boiling water were expelled into pools of cold water.

A device was developed and tested using the pressure-suppression concept⁽¹⁵⁾. This device constitutes a small structure which serves both as a water pool and a pressure containment. Primary coolant was released from a simulated reactor vessel through ducts into the pressure-suppression chamber which contained a pool of cold water. The interchange of heat between the expanding steam and the cold water effectively reduced the primary coolant pressure. Therefore, expelled coolant and the associated high pressure was contained within a relatively small chamber without the requirement of a conventional, expensive dry containment shell.

5. Conclusions

The research and development programs carried out thus far in the field of nuclear safety have provided, and will continue to provide, information of importance to the long range understanding of the complex phenomena associated with the loss of coolant accident in water-cooled and -moderated power reactors. Basic research programs aimed at investigation of the effect of each of the independent parameters associated with the various loss of coolant phenomena are particularly essential to the development of reliable analytical models for predicting accident consequences which are applicable to many reactor systems. Nevertheless, the nuclear safety program cannot rely solely on basic research because of the inherent limitations of separate phenomenological studies for providing information on the interrelation between the several phenomena involved during a loss of coolant and the ultimate need for integrated proof-of-principal tests to demonstrate the reliability of experimental results obtained by basic research studies.

A test program is proposed which utilizes a full-scale nuclear system. The program is intended to provide the engineering and safety information needed by the nuclear industry at an early date and to assist in identifying those phenomena which are most significant with respect to safety during a loss of coolant. The proposed program is described in the following section.

III. PROPOSED EXPERIMENTAL PROGRAM

A. Introduction

The proposed loss of coolant test program includes the testing of a complete nuclear system, typical of pressurized water power reactors, under conditions that simulate as closely as possible those conditions which are present in an actual accident. The broad objectives of the program are:

- (1) to demonstrate the consequences of a loss of coolant accident in a pressurized power reactor,
- (2) to provide information on the behavior of integrated components and interacting processes which is needed to answer specific questions regarding accident hazards and safety assessment,
- (3) to determine which phenomena associated with the loss of coolant accident are of principal importance in predicting or evaluating the consequences of such an accident and thereby provide information needed to guide subsequent follow-on programs directed at further studies of these phenomena, and
- (4) to provide information needed to complement research programs aimed at understanding the basic theory of the individual phenomena.

To achieve these objectives, a stepwise test program is proposed that will provide information on all phases of the loss of coolant accident, i.e., coolant blowdown, core meltdown, fission product transport, and radiological hazards.

The test program is separated into three stages in accordance with the test sequence. The stages are: (1) Stage I - Coolant Blowdown Test with a Nonradioactive Core, (2) Stage II - Static Physics Measurements and Reactor Operation, and (3) Stage III - Loss of Coolant With a Radioactive Core.

The first part of the program, designated as Stage I, will consist of performing several preliminary blowdown experiments as a function of system temperature, pressure, and rupture size and location. These

preliminary tests are performed to investigate the containment building and primary coolant system response, to evaluate the mechanisms employed to initiate the simulated primary coolant pipe rupture, to gain some early information pertaining to fission product transport and plateout behavior inside the containment building, and to evaluate the test instrumentation and testing techniques prior to actual core meltdown test. In addition, some of the data pertaining to mass flow rates will assist in evaluating the analytical models used for predicting coolant blowdown rates and contain vessel pressures.

Following the preliminary Stage I tests, the core will be installed and low power and high power physics and engineering measurements needed to determine the reactor characteristics will be performed. The reactor will be operated for approximately 850 MWD to buildup the fission product inventory and to provide sufficient after-shutdown heat flux to cause core melting. This part of the program is identified as Stage II.

At the end of the extended power run, a pipe rupture will be initiated, the coolant will be expelled from the system, and the core will be permitted to undergo melting. During this stage of the program, identified as Stage III, the coolant blowdown characteristics, extent of core damage, and the space-time behavior of the fission products will be determined.

Upon completion of the experiment the complete nuclear system will be removed to the hot shop, and the extent of core damage and fission product release will be determined.

At the conclusion of the program discussed above, depending upon the results, it may be desirable to evaluate the effectiveness of several safety devices in reducing the consequences of a loss of coolant accident. A description of such a program is presented under "Follow-On Programs".

The nuclear system to be used for this program will be pressurized water system containing a low enriched, rod type, oxide core of conventional design, having a power capability of approximately 50 Mw(t). The entire nuclear system will be mounted on a railroad dolly to facilitate cleanup and movement from the test building to the examination area (hot shop) following the final test. During the performance of the

test, the dolly containing the nuclear system will be housed in a containment vessel capable of withstanding the pressures associated with the coolant expulsion and of limiting the fission product leakage to the atmosphere.

The general description and objectives of each stage of the experimental program, proposed follow-on programs, and required development programs are presented below.

B. Stage I - Coolant Blowdown Test With a Nonradioactive Core

This stage of the experiment is performed to investigate the coolant blowdown characteristics and consequences as a function of rupture size, location, and coolant temperature and pressure conditions. These tests are suggested prior to the installation of the core to provide information substantiating, (1) the techniques used to initiate a rupture, (2) the ability of the containment vessel to withstand the transient pressure pulse, (3) experimental instrumentation and techniques, (4) the rupture location that introduces the maximum concentration of radioactive iodine to containment vessel, and (5) the analytical models used in predicting the coolant expulsion rates and the containment vessel pressure. In addition, this data will provide some early information pertaining to fission product transport and plateout inside the containment building as well as supplying data needed to extrapolate the test results from this primary coolant system to systems operating at different temperatures and pressures.

The tests will be performed on the complete nuclear system, except for the core. A dummy core, identical in size, mass, and structural rigidity to the actual core will be installed in the system to provide the same flow restriction qualities that will prevail during the actual core meltdown test. It should be recognized that the results of preliminary tests using a dummy core will not necessarily duplicate the results of tests using a live core.

This stage of the program may be separated into two steps: (1) coolant blowdown, and (2) fission product transport phenomena. The general test procedure and data to be collected are presented below.

1. Coolant Blowdown

A limited number of pipe ruptures will be performed up to 2500 psi and 600°F to investigate the blowdown characteristics as a function of temperature and pressure for rupture sizes of about 6 in., 12 in., and 20 in. in diameter. The location of the ruptures will be on the outlet and inlet main primary coolant lines at positions as near as possible to the 84 in. inside diameter reactor vessel. The effect of the primary coolant pump operation on the system hydraulics during blowdown will be examined in this stage of the experimental program.

The number of rupture locations and sizes identified are considered to be a minimum acceptable number for obtaining sufficient data to evaluate the analytical models used by industry to predict the expulsion rates and the consequent pressures on the containment vessels. The expulsion rate data may also be useful in evaluating the hydraulic behavior of the coolant system during blowdown for various blowdown rates.

The experimental data to be obtained from each blowdown test are:

- (1) Coolant expulsion rate as a function of time--expulsion mass flow rate and the coolant quality at the rupture.
- (2) Pressure inside the coolant system and containment vessel as a function of time.
- (3) Temperature as a function of time inside the coolant system, at the rupture location, at various positions on the inner and outer surface of the containment vessel, at associated equipment inside the containment vessel, and in the containment volume.
- (4) Strain at various positions on the containment vessel and coolant system.

It should be noted that reliable techniques are not available for initiating ruptures larger than 6 in. in diameter and for measuring the mass flow rate and coolant quality at the rupture. A development program will be required to investigate these items prior to the performance of this test program.

2. Fission Product Transport Phenomena

During several of the ruptures, Iodine-129 containing a trace of Iodine-130 and a small quantity of radioactive krypton will be released inside the reactor vessel and their transport characteristics will be observed. The isotopes will be contained inside a simulated fuel rod located at the center of the core mockup. They will be released at a time after the rupture that is representative of the theoretical time at which core melting starts. The specific purpose of these experiments is to provide information regarding the rupture location that provides the maximum fission product leakage to the containment vessel. These tests will also provide an opportunity for evaluating the experimental techniques for measuring iodine transport and plateout behavior as well as evaluating the decontamination techniques for iodine removal.

At least one rupture incorporating a fission product release will be initiated in conjunction with this stage of the test program for the purpose of providing early information on the effectiveness of the pressure reduction spray system and fission product filtering systems in reducing the iodine leakage from the containment vessel. This system is designed to reduce the leakage, following the actual core meltdown, to a nonhazardous level in the event a change in meteorological conditions and normal leak rate present radiological hazards to on-site or off-site personnel.

The measurements to be made during this phase of the test are:

- (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.

C. Stage II - Static Physics Measurements and Reactor Operation

This stage of the test program includes all low power and high power experimental measurements that are required to provide a thorough understanding of the core nuclear, hydraulic and heat transfer parameters. An understanding of these parameters is essential to interpretation and analysis of experimental results.

The fuel and clad temperature as a function of time after the pipe rupture is dependent on the reactor power behavior following the initiation of the rupture. In order to provide information on the power behavior or inherent shutdown mechanisms due to void formation, the reactor will not be scrammed at the time of the system rupture. The control rods will be maintained at their operating position until such time as the reactor power has been reduced several decades. The rods will then be manually scrammed.

The magnitude and distribution of the after-shutdown decay heat flux are required to evaluate the analytical models for predicting the extent of core melting following a loss of coolant. The magnitude of the average afterheat flux can be predicted with reasonable accuracy by existing techniques provided the power and flux distribution and the operating history of the core are known; thus, the variable to be determined from this experiment is the flux distribution which will be determined by extensive neutron flux measurements in the core.

The magnitude and distribution of the fission products are also directly proportional to the power history and flux distribution in the core. Thus, they will be determined by techniques described above. This information is required to determine the magnitude of the fission product release from the damaged fuel rods. A continuous operating period of at least 400 hours at 50 Mw(t) prior to the test will be sufficient to produce an adequate heat flux and fission products inventory. The after-shutdown heat flux for the first four hours following shutdown will be approximately 85 percent of the heat flux following an infinite operating period at the same power level. The fission product inventory will be:

<u>Isotope</u>	<u>Percent of Equilibrium @ 50 Mw(t)</u>
I-131	76.1
I-132	100
I-133, 134, 135	100
Xe-133 m	99.3
Xe-133	88.8
Xe-135 m, 135	100
Kr (all isotopes except Kr-85)	100
Kr-85	0.3

The additional data to be collected during the power run are:

- (1) continuous recording of reactor power level, coolant temperature and pressure, and control rod position,
- (2) fuel rod clad and fuel temperature, and
- (3) flux profile distribution periodically throughout the power run with telflex and miniature chamber insertion.

D. Stage III - Loss of Coolant With Radioactive Core

This stage of the experimental program is intended to demonstrate the loss of coolant accident on a complete nuclear system containing a core with sufficient decay heat to cause fuel rod melting. Prior to the performance of the test, extensive calculations will be made to predict the extent of core damage and the potential radiological hazards associated with loss of coolant. The data obtained during Stage III will be used to evaluate these analytical techniques.

The proposed tests to be performed may be separated in accordance with the sequence of events that occurs following the system rupture, such as: (1) coolant blowdown, (2) core meltdown, (3) fission product release, transport, and plateout, (4) radiological hazards, (5) facility decontamination, and (6) post-test examination. The specific purpose of each test and the measurements to be made are described below.

1. Coolant Blowdown

While the nuclear system is operating at the normal power [50 Mw(t)] operating conditions, a primary coolant line rupture will be initiated. The size of the rupture will be equivalent to the inside diameter of the coolant pipe and the location will be as determined by analysis and by Stage I experimental results. The core will be allowed to shut itself down by the inherent negative temperature and density coefficients of reactivity.

The specific purpose of initiating the rupture and allowing the coolant to flow from the system is to demonstrate and measure the effects of rapidly removing the coolant from the reactor system in a manner that is representative of the most severe pipe break accident. The coolant expulsion segment of this experiment and the data to be collected will be identical to that described in Stage I for the maximum size rupture, except that the test measurements will be made over a longer period of time. The mass flow rates or vapor currents through the rupture opening and appropriate temperatures and pressures are expected to be measured for several days.

A complete history of these parameters will aid in describing the transport mechanisms for transport of fission products from the release point to the containment vessel. In addition, the data will be most useful in determining the effects of heat generated in the core on the coolant expulsion phenomena.

The measurements proposed for this part of the test are enumerated below:

- (1) core power,
- (2) primary coolant flow, temperature, and pressure inside the coolant system and on each side of the rupture opening,
- (3) coolant expulsion rate at the rupture opening, mass flow rate and coolant quality,
- (4) vapor current velocity and temperature at the rupture opening after the coolant has been ejected,

- (5) temperature at various positions on the inner and outer surface of the containment vessel, at associated equipment inside the containment vessel, and in the containment volume,
- (6) pressure inside the containment vessel,
- (7) material strain at various positions on the containment vessel and coolant system,
- (8) in-core temperatures,
- (9) strain on core fuel rods, and
- (10) moisture content of air within containment vessel as a function of time and position.

2. Core Meltdown

Following a rupture in the primary coolant system of a pressurized water reactor, many complex and interrelated phenomena occur in the core region which ultimately determine the extent of core meltdown and quantity of radioactive materials released. Since these phenomena have a direct effect on the radiological hazards associated with a loss of coolant accident, the ensuing consequences must be surveyed to assist in evaluating the analytical models and assumptions used to predict the extent of core meltdown and the quantity of the fission product release. The areas which require special attention are as follows:

- (1) measurement of the fuel and cladding temperatures at several positions, both axially and radially, in the core,
- (2) determination of the extent of core damage after the test has been terminated, and
- (3) investigation of the presence and magnitude of a metal-water reaction.

Items (2) and (3) will be investigated after the test has been terminated and the entire nuclear facility has been removed to the hot shop for examination.

The majority of the power reactor hazards analyses are based on the assumption that the entire core, fuel and clad, melts. An actual determination of the fraction that undergoes damage may provide a basis for a more realistic assumption which will reduce the postulated hazards associated with a loss of coolant accident.

3. Fission Product Release, Transport and Plateout

The fission product behavior, such as: (1) the amount of fission products release from the core, (2) the transport behavior inside the reactor vessel and containment vessel, and (3) the plateout behavior, needs to be better understood. An experimental determination of the fission product behavior for the actual accident conditions should provide a basis for reducing some of the conservatism which is employed in the siting limitations for nuclear power plants. This part of the experiment is expected to provide the greatest gain toward the reduction of the calculated radiological hazards which are presently incorporated in hazards analyses.

a. Release

Before the radiological hazards can be accurately evaluated, the concentration of the various fission products that are released from the core must be known. By knowing the fission product inventory in the core and the concentration remaining in the fuel after core meltdown as determined by post-meltdown examination, the amount released can be determined.

b. Transport and Plateout

The fission products transport and plateout behavior will be determined as a function of time after core meltdown. Experimental information to determine these phenomena are needed to assist in evaluating the radiological hazards.

The fission product concentration in vapor and on the material surfaces will be measured as a function of time at a number of positions inside the containment vessel. Because of the large size of the containment vessel, some compromise must be made between the number of

samples that are desirable and the number that can be taken. To provide a complete coverage with a minimum number of samples, the container volume and material surfaces will be separated in zones with reference to rupture location. In each zone measurements will be made to determine the identity of the fission products and their concentrations as a function of time. The summation of the concentrations in each zone will then be considered as the actual inventory of each isotope inside the containment vessel.

The transport behavior will be evaluated from air volume samples of both the free gas and particulate matter in each zone of the containment volume. This sampling will also indicate the type of fission products and their concentrations that are available for leakage from the containment vessel as a function of time.

The plateout behavior will be determined from plateout samples located on machinery and wall surfaces in each zone. These samples will be identical, in material and surface coating, to the surface areas being monitored. Thus, by knowing the number of curies of a specific isotope on a sample of given surface area, an estimate of the plateout on the entire surface area for each zone can be made. In addition, plateout concentration as a function of position inside the reactor vessel and primary coolant lines and the concentration collected in the building sump tank will also be determined.

No attempt will be made to mockup the extensive surface areas that exist in conventional nuclear power plants because (1) it does not appear feasible to simulate or mockup the large surface areas and transport paths in such a manner that the fission product plateout data obtained will be applicable to all reactors, (2) it appears more desirable to provide "upper limit" information regarding fission product leakage from the containment building, and (3) it appears possible to predict the quantity of fission products that would be plated-out on various surfaces by measuring the plateout on prepared representative samples, of known surface area, as a function of time and position.

The specific measurements to be attempted are enumerated below:

- (1) fission product identification and concentration as a function of position on the inner surfaces of the coolant pipes, reactor vessel, and structural materials inside the reactor vessel,
- (2) fission product identification and concentration in liquid waste tanks,
- (3) plateout on machinery and containment shell surface areas as a function of time, and
- (4) fission products in containment air volume as a function of time.

4. Radiological Hazards

Radiological measurements will be attempted after the core meltdown to determine the radiation dose rates resulting from fission product leakage from the containment vessel and the gamma dose rates from the direct radiation emanating from the building. These measurements are required to assess the hazard associated with the experiment to both on-site and off-site personnel. With the knowledge of the meteorological conditions, the measured dose rates downwind can be used to evaluate the analytical models used for these predictions.

The measurements to be made are enumerated below:

- (1) meteorological conditions--wind speed, atmospheric temperature versus height above the ground, air trajectory studies, and cloud size,
- (2) total external whole body dose from gamma and beta rays originating in the cloud versus distance from the building,
- (3) inhalation dose from iodine versus distance downwind,
- (4) total external whole body dose from radioactive fallout versus distance downwind,
- (5) direct gamma dose rate from containment building as a function of distance and time,
- (6) direct gamma and beta dose rates inside the containment vessel versus time, and
- (7) fallout particle size.

5. Decontamination

At the conclusion of the test and prior to gaining access to the building, the airborne fission products and surface contamination within the building and on the experimental equipment will be reduced to a non-hazardous condition. This will be accomplished by remote decontamination using the wall and dome spray systems to deluge the building interior with appropriate chemical solutions and flush water.

The remote decontamination of the complete facility is not specifically intended to provide data to the nuclear industry. However, the effectiveness of the procedures and materials used may well be of general benefit to the Commission and industry in preparing procedures to cope with emergency situations.

After the outer surfaces of the nuclear system have been decontaminated sufficiently for removal from the containment vessel, the dolly containing the entire system will be removed to the examination area.

6. Post-Test Examination

The examination of the nuclear system is designed to provide the information required to evaluate the extent of core damage, the fission product dispersal, and the physical damage to the coolant system resulting from the loss of coolant. A complete understanding of these parameters will aid in the overall evaluation of the consequences of the loss of coolant accident.

The core remains will be disassembled and each individual fuel rod will be examined for clad rupture, clad melting, and fuel melting. Representative samples of the fuel rods that have undergone the various degrees of damage will be chemically analyzed to determine the identity and quantity of fission products remaining in the fuel. All loose fuel found in the system will be examined to determine whether or not it had undergone melting and to determine the quantity of fission products remaining.

The data to be obtained from the post-test examinations are:

- (1) the number of fuel rods that had both clad and fuel melting, plus the extent of melting for each rod and its location in the core,
- (2) the number of fuel rods that had only clad melting, plus the extent of melting for each rod and its location in the core,
- (3) the number of fuel rods that had only clad rupture and their location in the core,
- (4) identification and concentration of fission products remaining in the fuel as a function of fuel rod damage and location in the core,
- (5) identification and concentration of fission products remaining in the reactor vessel and coolant pipes,
- (6) physical damage to the nuclear system caused by the loss of coolant, and
- (7) extent of metal-water reaction.

E. Follow-On Programs

One of the objectives of the test program is to determine which phenomena associated with the loss of coolant accident are of importance in evaluating or predicting the consequences of such an accident. It is anticipated that several follow-on programs to further study these phenomena will be initiated as a result of this program and it is expected that some of these programs could be carried out within the LOFT facility.

In addition three follow-on programs which could also be conducted in this facility are proposed for evaluating the effectiveness of special safety devices which have been suggested for minimizing the radiological hazards associated with a loss of coolant accident. The three safety devices are the spray system, the filtering system, and the foam system. A separate program is proposed for each device; therefore, three destructive tests are required. These additional destructive experiments will primarily test the effectiveness of the particular safety device; furthermore, more reliability will be added to the results from the initial loss of coolant experiment.

The first approach to this series of experiments will be to test the devices, namely, the containment vessel spray system and the filtering system, that are incorporated in the original LOFT facility design. Following these tests the foam system, which is presently in the development stages, could be tested with only slight alterations to the presently designed facility.

Each safety device tested will require a coolant rupture on an identical nuclear system containing the same type core with a comparable operating history as the nuclear system described in the preceding experimental program. The experimental procedure would also be identical to that previously described except for Stage I, which will be completely eliminated, and Stage III, which will be changed to incorporate the use of the safety device being testing. A general discussion of the actual safety device tests is presented below.

1. Spray System

A system rupture will be initiated as described in Stage II of the experimental program, and the core will be allowed to melt as described in Stage III. Radiological measurements to determine the fission product inventory in the containment volume will then be made until a relationship is established between the original core meltdown test and this test. At this time the containment volume sprays will be initiated, and again the fission product inventory will be determined as a function of time. In addition, radiation measurements will be made to evaluate the radiological hazards from gamma rays emanating from the containment vessel, and from fission product leakage from the building. From this data and data obtained during the post-test examination, an efficiency factor for the spray systems can be determined.

2. Containment Gas Filtering System

This test will be identical to that outlined for the spray system except that a recirculating filtering system, as described in Section VI-B-1-f, will be used to reduce the fission product concentration.

3. Foam System

A foam approach, as developed by Professor Leslie Silverman, Harvard University, is intended to encapsulate released fission products by means of a high-expansion foam. The foam that has been used for development experimental purposes is a high-expansion (1000 to 1) lauryl sulfate solution to which various iodine reactants have been added. Preliminary tests indicate that as much as 90 to 95 percent of iodine can be removed from the atmosphere with this system. Further studies using Iodine-131 are currently underway. If these studies also indicate that this is a promising technique, a test will be performed to determine its effectiveness under actual loss of coolant conditions. The test procedure will be the same as outlined for the spray system except that the foam system will be used.

In addition to evaluating the effectiveness of the safety devices, the data obtained during coolant blowdown, core heatup and meltdown, and post-test examinations will tend to increase the reliability of the data obtained during the first core meltdown when no safety devices are used.

F. Development and Testing Programs

Development and test programs paralleling the design effort appear to be required to develop some of the testing techniques and equipment that are required in order to carry out the proposed program. The development and test programs visualized are as follows:

- (1) rupture disk testing,
- (2) two-phase flow measurement,
- (3) instrumentation development, and
- (4) decontamination techniques.

This test program requires that the rupture opening for the primary coolant expulsion simulates a pipe shear and be of such geometry that its area can be expressed analytically. Some development work has been done⁽²⁹⁾; however, this work has been limited to openings less than 6 in. in diameter. It is recommended that this program be extended to cover

ranges of openings up to 20 in. in diameter. Furthermore, a limited number of experiments have been performed to determine expulsion rates from small systems. These systems were designed such that the expulsion rate was determined by weighing the entire system during coolant blowdown. Even though this technique may be applicable for the LOFT facility, additional development on this concept is required.

To supplement possible system weight change techniques in order to ascertain the time history development of coolant expulsion from the ruptured pipe, with attendant two-phase flow in core, vessel and piping, special detectors must be developed for measuring the resulting transient flow rate, mass-flow rate, and coolant density in the various regions.

Techniques using drag disks, anemometry, ultrasonics, x-ray absorption, high frequency dielectric measurements and the magneto hydrodynamics principally should be explored.

In order to gain admittance to the containment vessel after the completion of the test program, some decontamination of equipment and structural surfaces must be performed. Several techniques appear promising, however, they have not been used under full-scale field conditions. Thus, it is necessary to evaluate these techniques prior to the performance of the experimental program.

G. Roving Manipulator System

There exists within the facilities of the STEP program a remotely-controlled roving manipulator system which will be used in the LOFT program to effect removal of rail obstructing debris, spot decontamination of the test facilities as required after general flushing of the building, visual aid, and recovery of near field radiological instrumentation.

The Mobot, which can be controlled by radio or via a tri-axial cable as convenient, is equipped with two closed-circuit TV cameras on positioning arms, 100 lb manipulating arms, jib crane and a fork lift, all of which can be operated from the remote console in the reactor control room. The Mobot is capable of climbing over a 6 in. rail and up a 35% slope.

Inasmuch as remote operation is time consuming and somewhat limited in scope, the Mobot will only be used when radiation or other hazards exist to preclude the presence of personnel from the area.

IV. REACTOR AND EXPERIMENT DESIGN CONCEPTS

A. Discussion of Core Concepts

The loss of coolant accident experiment must provide the nuclear industry with engineering information capable of extrapolation to typical power reactors. To assist in attaining this objective, the criteria for the conceptual design of the core states that the design must reflect reactor design practice currently being employed in industry. In this regard, the core is to contain slightly enriched uranium dioxide in the form of pressured and sintered cylindrical pellets stacked in stainless steel tubes. The core power must be kept small enough to minimize heat dissipation equipment costs and yet large enough to insure that melting of a significant amount of the core will be attained. The power density was chosen, consistent with current commercial power designs, as approximately 55 kw/l of core. In addition, it is necessary to consider the selection of a core utilizing fuel rods that have undergone high burnup (~ 7000 MWD/T) or a core with non-irradiated fuel rods. A core containing high-burnup fuel will release a higher concentration of fission products. However, existing data and analytical techniques are available for determining the fission product inventory as a function of fuel burnup, and experiment data⁽²²⁾ are available for predicting the fission product release from UO_2 fuel as a function of fuel burnup. Thus, the overall concentration of fission products released from a low-burnup core can be extrapolated with reasonable accuracy to that released from a high-burnup core.

The high-burnup core introduces many handling problems such as (1) fuel acquisition from an operating power reactor, (2) transportation of the highly radioactive core to the NRTS, (3) instrumenting of the fuel elements to obtain meaningful information, and (4) the additional equipment required to load the reactor. In addition, the operation of a high-burnup core at the necessary heat flux would require additional heat dissipation equipment, resulting in excessive facility costs.

The possibility of using a partially spent core, seeded with new elements, was also considered in order to reduce the overall reactor power requirements and size of the core. However, to insure melting

of the spent fuel without damage to the new fuel, a complicated calandria type core is necessary with auxiliary cooling required in the fuel tubes housing the new fuel. The results of a loss of coolant accident with a calandria geometry and associated reactor internal equipment would not provide information usable in typical power reactor technology.

B. Nuclear Characteristics

1. General

The first nuclear reactor system for the LOFT facility is a pressurized water system designed to operate continuously for 850 MWD at a maximum power level of 50 Mw(t). Although detailed reactor engineering calculations are necessary to determine final core size and characteristics, it is felt that the following conceptual design characteristics are sufficient to allow preliminary sizing of the vessel and primary system and to demonstrate the capability of sustained power operation with a smaller size core.

The fuel rods or tubes in this design are bundled into 24 fuel assemblies which are assembled to form a reactor core approximately 45 in. in diameter and 36 in. in height. The reactor core in the conceptual design contains nine cruciform-type boron-stainless (1.5 w/o B¹⁰) control rods. Each control rod is 7.865 in. wide, 0.265 in. thick, and 36 in. long. The stainless steel control rod followers, which are attached to the bottom of the control rods, are of the same dimensions as the control rods.

The reactor core is surrounded by a form-fitting baffle that divides the coolant flow between the fuel bearing and the reflector region. The reflector region is considered as that region between the baffle and the reactor thermal shields.

The water coolant flows through a nozzle near the bottom of the reactor vessel, upward through the core and reflector and exits through the upper reactor vessel nozzle. After leaving the reactor vessel, the coolant passes through the primary coolant system described in Section IV-E.

A schematic of the reactor core and control rod arrangement is presented in Fig. IV-1.

For the purpose of this report, machine calculations were made to determine the power level, size and critical mass of a uranium oxide core. Upper limit design parameters, such as operating temperatures, heat flux, etc., for typical pressurized water power reactors were used to develop the design criteria along with a reasonable power level to attain a measurable quantity of fission products and a decay heat flux that approaches the decay heat flux for a typical pressurized water power reactor at the end of core life.

A summary of the important design features of the core is presented in Table IV-a. It is to be expected that the final design will result in some modifications in the above characteristics. Certain assumptions have necessarily been made in the present report which need calculational confirmation. For instance, burnup calculations should be made to determine more exactly the reactivity needed for 400 hours of operation. In addition, the effect on reactivity of decreasing the reflector thickness (assumed effectively infinite) was not evaluated in this study. However, even though some changes may be made as a result of additional calculations which will be carried out for a final design, present calculations indicate that the basic LOFT conceptual design as presented here can meet the requirements of the loss of coolant accident.

From the standpoint of gaining sufficient reactor control, the only approach investigated was that of increasing the vane length of the cruciform control rods. While it was shown that this method is feasible, other schemes could be used. For example, it might be possible and advantageous to increase the B^{10} enrichment in the boron stainless steel, and thereby retain smaller control rods.

2. Reactor Physics

Reactor physics calculations for the LOFT reactor were based exclusively on four-group diffusion theory in two dimensions. Experience with similar reactor systems, i.e., water moderated cores employing low enrichment UO_2 pin type fuel elements, has shown that

TABLE IV-a

LOFT CORE DESIGN FEATURES

Length (inches)	36
Number of fuel assemblies	24
Fuel assembly size (inches)	7.641 by 7.641
Fuel element	
Fuel pellet diameter (inches)	0.294
Fuel rod diameter (inches)	0.340
Gap between clad and fuel (inches)	0.004 (nominal diameter cold)
Clad thickness (inches)	0.021
Rod spacing (inches)	0.422 (center to center)
Clad material	Stainless steel
Average fuel temperature	950°F
Average clad temperature	545°F
Fuel enrichment	~ 4.2%
Control rods	
Number	9
Type	Cruciform
Material	Boron-stainless (1.5 w/o B ¹⁰)
Follower	Stainless steel
Normal operating power [Mw(t)]	50
Normal operating heat flux	
Average	86,000 Btu/hr/ft ²
Maximum (nominal)	218,000 Btu/hr/ft ²
Excess reactivity (cold)	14.5%
Excess reactivity (hot)	3.5%
Shutdown reactivity (cold)	5%
Peak-to-average power distribution	
Axial	~ 1.4
Radial	~ 1.8
Core lifetime	400 efph
Core area	1400 in ² (see Fig. IV-1)
Reflector	H ₂ O
Coolant temperature (average)	516°F
Δ T	31°F
Coolant pressure (psig)	2500

this approach is quite satisfactory for conceptual design purposes. In particular, the results of identical calculations previously carried out for the MARTY critical experiment⁽³⁰⁾ and the SPERT-I UO_2 core⁽³⁰⁾ were found to be in good agreement with experiment.

a. Thermal Constants

Thermal constants for neutron energies from 0 to 0.625 eV were obtained using the 650 SOFOCATE⁽³¹⁾ code. For the core, an initial average Wigner-Wilkins flux spectrum was calculated by SOFOCATE with input atom densities homogenized on a volume fraction basis. Constants for the UO_2 , stainless steel cladding, and surrounding water moderator were then arrived at by averaging their energy dependent cross-sections over the homogenized Wigner-Wilkins spectrum. With these constants, a P-3 spherical harmonics program⁽³²⁾ was used to calculate the fine flux profile over a unit cell (see Fig. IV-2) which allowed determination of flux disadvantage factors for the UO_2 and stainless steel cladding. Finally, a second Wigner-Wilkins spectrum was calculated with the atom densities adjusted by these flux disadvantage factors. The resulting core constants made it possible to treat the core as homogeneous with respect to thermal neutrons in the diffusion theory calculations.

Constants for the control rods, control rod followers and flow skirt were obtained by averaging over the core spectrum. In the case of the control rods, the Blackness Theory⁽³¹⁾ was applied.

b. Fast Constants

The fast group constants were arrived at using the GAM-1⁽³³⁾ code. This code takes into account Doppler broadening of the U-238 resonances and the geometric effects of fuel lumping. As in the thermal case, constants for the control rods, rod followers, and flow skirt were determined by averaging over the core spectrum.

c. Diffusion Theory Calculations

The four group diffusion theory calculations were done entirely with the PDQ-2 program⁽³⁴⁾. All problems contained 4900 mesh points and represented one quadrant of the core in X-Y geometry with

essentially infinite water reflection. The group independent buckling (0.001) used in the problems was taken to be equal to the geometric vertical core buckling plus 10 cm reflector savings.

Similar calculational methods previously applied to the MARTY critical experiment resulted in calculated eigenvalues which were 2.1% high. Based on these results, all eigenvalues predicted for LOFT are normalized by reducing the calculated values 2.1%.

d. Fuel Enrichment

With the core geometry as described previously, the U-235 content of the UO_2 may be varied to obtain the necessary reactivity in the cold core. This reactivity must be sufficient to maintain criticality as the reactor temperature and power are raised to operating conditions. Thereafter, the core must have sufficient remaining reactivity to override equilibrium xenon and samarium poisoning and sustain fuel burnup for a desired 400 hours of operation.

To arrive at the required reactivity, and hence fuel enrichment, it was tentatively assumed that the reactivity necessary to override equilibrium xenon and samarium poisoning is 2.51%. It is also assumed that fuel burnup requirements in LOFT will not exceed $1\% \Delta K/K_1 K_2$. Based on these assumptions, the LOFT core when first brought to full power should have a remaining reactivity of 3.51%. To determine the fuel enrichment necessary to give this amount of reactivity, two PDQ-2 problems were run for fuel enrichments of 3.44% and 5%, respectively. Input constants were obtained for an average coolant and reflector temperature of 516°F, and an average UO_2 temperature of 945°F. This corresponds to conditions of full power operation. In addition to these two problems, four problems were run with various fuel enrichments for the cold (68°F) reactor case. The results of all problems are shown graphically in Fig. IV-3. From curve 2 of the figure, it is found that the necessary fuel enrichment is 4.2%, i.e., this is the fuel enrichment which will provide the LOFT reactor when initially brought to operating temperature and full power with a remaining reactivity of approximately 3.5%. Predicted LOFT eigenvalues as read from the curve and the resulting average temperature coefficient of reactivity are listed in Table IV-b.

TABLE IV-b

LOFT EIGENVALUES AND AVERAGE TEMPERATURE
COEFFICIENT OF REACTIVITY FOR A 4.2% UO₂ ENRICHMENT

K_{eff} (cold)	1.101
$\rho_C \equiv K_{\text{ex}}/K_{\text{eff}}$ (cold)	.091
K_{eff} (hot-full power)	1.036
$\rho_H \equiv K_{\text{ex}}/K_{\text{eff}}$ (hot-full power)	.035
$\Delta\rho = \rho_C - \rho_H$	.056
Average Temperature Coefficient of Reactivity $\Delta\rho/F$ Deg.	-1.2×10^{-4}

3. Reactor Control

As shown in Fig. IV-1, the proposed LOFT geometry would employ nine cruciform control rods. Boron stainless steel tentatively enriched to 1.5% by weight B¹⁰ was chosen as the control rod material with rod follower sections composed of pure stainless steel. The width of the cruciform vanes is 0.265 in., while the length of the vanes may be regarded as a variable to obtain the necessary rod worth. As a starting point in control rod worth calculations, a vane length of 3.9325 in. (or overall cruciform width of 7.865 in.) was assumed.

It is desirable to have enough control rod worth to render the cold reactor at least 5% subcritical under normal operating conditions. Also, in the event of a stuck rod accident, it is necessary that any eight of the nine rods must have sufficient worth to keep the reactor subcritical. The worth of nine control rods was calculated with PDQ-2 using as a base problem the cold, 5% enriched fuel case. The results were as follows:

K_{eff} (cold - rod follower sections in - no flow skirt)	1.144
K_{eff} (cold - all nine rods in - no flow skirt)	.986
ΔK_{eff}	.158
Control rod worth ($\Delta K_{\text{eff}}/K_1 K_2$)	.140

If one applies this reactivity swing to the 4.2% enriched fuel case, the results for LOFT would be:

K_{eff} (cold - rod follower sections in - with flow skirt)	1.101
K_{eff} (cold - all nine rods in - with flow skirt)	.954
ΔK_{eff}	.147
Control rod worth ($\Delta K_{\text{eff}}/K_1 K_2$)	.140

The calculated K_{eff} of .954 may be viewed with some uncertainty because no transport theory corrections were applied in obtaining the fast group diffusion theory constants for the control rods. Due to this effect, the rods would be worth slightly less than the calculated 14%. On the other hand, the control rods would actually be worth somewhat more than 14% in the softer energy spectrum of the 4.2% enriched fuel environment. The first of the two effects may prove to be predominate, so that further calculations, which are being made, might indicate the necessity of increasing the vane length.

To investigate the effect of extending the vane length, two additional problems were run with all nine rods in but with increased vane lengths. The problem results are shown graphically in Fig. IV-4. From the figure, it is seen that considerable added shutdown reactivity can be obtained by this method. Of course, there are limitations to the shutdown reactivity that could be added in this way due to mechanical difficulties. For instance, the rods must pass through the end restraining grids so that a maximum extension is obviously impossible. Another effect of adding shutdown reactivity by vane extension is that the fuel enrichment must be increased. This increase of enrichment is

necessary since the correspondingly larger follower sections would displace fuel with stainless steel, resulting in a loss of core reactivity.

In the event of an accident in which a single control rod sticks, it must be possible for the remaining eight rods to shut the reactor down. Because the center rod is worth more than each of the other rods, the shutdown reactivity was determined for the case where the center rod is completely withdrawn and remaining rods inserted. The relative worth of the central rod was determined with PDQ-2 by rerunning a problem with only the eight outer rods in. The central position was occupied by a follower section. The resulting eigenvalue was .9779, whereas the calculated eigenvalue for all nine rods in was .9056. The rods of both problems had 1.45 inch vane extensions. Comparing these eigenvalues to the base problem with rods out ($K_{\text{eff}} = 1.144$), the central rod is responsible for about one-third (.36) of the total reactivity swing due to all nine rods.

If this result is applied to rods having a 14.0% reactivity swing, the eight outer rods alone would be insufficient to make the LOFT reactor subcritical. The estimated eigenvalue for this case is 1.002. Therefore, it will be necessary to design the outer rods to control a greater shutdown reactivity. One method of obtaining greater shutdown reactivity is to extend the vanes of all the rods. Table IV-c lists some estimated eigenvalues resulting from the insertion of control rods with different vane extensions. The estimated eigenvalues are based on the assumption that the relative fractional worth of the total reactivity swing controlled by the central rod is not a function of the vane length.

The peak-to-average power density in the X-Y plane of the hot LOFT core is approximately 1.8. A typical power traverse on a 45° diagonal (through the position of peak power density) is shown in Fig. IV-5. The peak-to-average power density is only weakly a function of enrichment, being 1.84 and 1.81 for the 3.44% and 5% fuel enrichments, respectively. Any vane extension would, of course, tend to raise these ratios somewhat.

TABLE IV-c

LOFT CORE EIGENVALUES FOR VARIOUS VANE EXTENSIONS

K_{eff} (cold - no rods)	1.10
K_{eff} (cold - all nine rods in - vane extension = .45 in.)	.92
K_{eff} (cold - eight outer rods in - vane extension = .45 in.)	.98
K_{eff} (cold - all nine rods in - vane extension = 1.0 in.)	.89
K_{eff} (cold - eight outer rods in - vane extension = 1.0 in)	.96

No axial calculations were made, although it is expected that a peak-to-average power density in the axial direction will be approximately equal to that of a cosine distribution (1.4). However, at the beginning of full power operation when the control rods are partially inserted, the axial peak-to-average power ratio will be somewhat higher than 1.4.

Table IV-d presents a summary of all pertinent PDQ-2 problem results (normalized to MARTY critical experiment calculations) which were run for the proposed LOFT reactor (Fig. IV-1). All problems, except where noted, had a 0.25 in. form-fitting stainless steel flow skirt and 0.265 in. by 7.865 in. control rod followers. All fuel enrichments are given in atomic percent of U^{235} .

C. Reactor Vessel

The reactor pressure vessel assembly consists primarily of a cylindrical shell with top and bottom hemispherical closures, plus appropriate core structure and a thermal shield. The 84 in. internal diameter cylindrical shell will be approximately 6.5 in. in thickness if a single-shell design is incorporated or approximately 5 in. in thickness if a welded, multi-layer design is incorporated. The average thickness of the top and bottom forged hemispherical heads will be

approximately 5.5 in. with adequate reinforcement for top control rod penetrations and top and bottom coolant nozzle attachments. The cylindrical shell and hemispherical closures are constructed from SA-212B carbon-silicon steel or ASTM A-302B carbon-molybdenum steel, both being proven high-pressure, high-temperature reactor vessel materials. The reactor vessel will have an overall height of approximately 23 feet. The total weight will be approximately 80 tons for the single-shell design and approximately 60 tons for the welded, multi-layer design.

TABLE IV-d

SUMMARY OF LOFT CORE EIGENVALUES

<u>Description</u>	<u>K_{eff}</u>
Cold - 3.4% fuel enrichment	1.050
Cold - 4.0% fuel enrichment - no flow skirt	1.095
Cold - 6.0% fuel enrichment - no flow skirt	1.180
Cold - 5.0% fuel enrichment - no flow skirt	1.144
Cold - 5.0% fuel enrichment	1.140
Hot - full power 3.44% fuel enrichment	.975
Hot - full power 5.0% fuel enrichment	1.085
Cold - 5% fuel enrichment - rods in (overall vane length = 7.865 in.) - no flow skirt	.986
Cold - 5% fuel enrichment - rods in (overall vane length = 10.765 in.) - no flow skirt	.906
Cold - 5% fuel enrichment - rods in (overall vane length = 15.102 in.) - no flow skirt	.837
Cold - 5% fuel enrichment - 8 outer rods in (overall vane length = 10.765 in.) - no flow skirt	.978

The pressure vessel will be designed for 2500 psig pressure at 668°F temperature and will be tested at 1-1/2 times the design pressure, or at 3750 psig. The ASME code will be used for the vessel design; however, the stringent thermal and pressure cycling stages proposed for the LOFT experimental program will prevent the possibility of

meeting all operating conditions of the ASME code. It must be emphasized that no personnel will be endangered during the blowdown test operations since the facility will be operated remotely. The choice between the single-shell and multi-layer shell will be primarily based on economy and convenience, not for the sake of meeting the requirements of any established design or safety code.

The LOFT reactor vessel contains numerous penetrations of various sizes and configurations to accomplish the loss of coolant test program. There are eight 2-in. flange nozzles equally spaced in a plane above and below the core for instrumentation leads into the core region. In order to photograph the reactor core during the meltdown phase, small viewing windows are incorporated into nozzles diagonally in line with the core. Because of the high radiation field in the containment building during normal operation and after core meltdown, all cameras must be installed and removed immediately prior to and after the loss of coolant test. The Architect-Engineer will provide a means of installing and removing this equipment.

Since fission product plateout is a function of surface temperature, the reactor vessel is insulated to obtain meaningful test results. The insulation will also minimize thermal stress in the heavy wall vessel during the loss of coolant tests.

The determination of type and quantity of fission product deposition on the vessel walls before the detailed examination of sub-systems and components in the hot shop would assist in establishing the data more quickly and serve as an additional control. Therefore, vertical nozzles are located on the top head within the inside periphery of the vessel walls, from which are suspended metal coupons. These coupons representing system materials can be retrieved for radiochemical and metallurgical analysis.

Nozzle penetrations are also provided for the experimental neutron and gamma detectors, pressure and temperature transducers, and for the instrumented fuel assemblies. These nozzles, located in the upper vessel wall, are oriented to facilitate recovery of this equipment from within the vessel without removal of the top head assembly.

In addition, nozzles are required for insertion of the fission and ion chambers. The associated thimbles and drive mechanisms will be designed and furnished by the operating contractor responsible for nuclear design.

Nine control rod extensions penetrate through the hemispherical top closure. Since the control rod drive mechanism has not been selected, the location and mounting details will be furnished to the Architect-Engineer by Phillips Petroleum Company. The top head penetrations are constructed in accordance with the latest ASME code consistent with the reactor vessel design. See Fig. IV-6.

D. Biological Shield

The Architect-Engineer will provide a water shield surrounding the reactor vessel to prevent neutron activation of the components inside the containment building and to attenuate the gamma rays emanating from the reactor vessel. The water shield will be mounted on the dolly and will be sufficient to:

- (1) reduce the thermal neutron flux emanating from the shield to less than 10^5 neutrons/cm² -sec⁻¹ during power operations at 50 Mw, and
- (2) reduce dose rate from fission product and activation gammas to less than 200 mr/hr at the surface of the shield at 15 minutes after shutdown. This level will be based on the assumptions that the reactor had been operating at 50 Mw for 400 hours.

Provisions will be made for cooling the primary water shield to a maximum temperature of 150°F.

E. Coolant Systems

1. Reactor Primary System (see Fig. IV-7)

The reactor primary system consists of a pressurizer, a bypass demineralizer system and a heat removal loop containing a heat exchanger, two primary pumps, an emergency shutdown pump, and a number of fill, drain and vent lines. The design criteria for the reactor primary system are summarized in Table IV-e. Primary coolant enters

and leaves the reactor vessel through single 20-in. nozzles at the base and the top of the vessel, respectively. The coolant path is through the single-pass primary heat exchanger to the two primary pumps in parallel, through the pumps to the primary flow control valve and flow measuring station, and is then returned to the base of the reactor vessel. Within the reactor vessel, 12,000 gpm flows up through the core, and the remaining 2000 gpm is diverted to the reflector and thermal shield regions.

TABLE IV-e

REACTOR PRIMARY SYSTEM OPERATING CONDITIONS

Reactor power	50 Mw (t)
Reactor maximum power density (nominal)	8500 Btu/hr-in ³
Reactor inlet temperature	497°F
Reactor exit temperature	528°F
Maximum clad surface temperature	610°F
Reactor inlet pressure	2500 psi
Core pressure drop	5.0 psi
Average coolant velocity in core	5.5 ft/sec
Core flow rate	12,000 gpm
Total reactor flow rate	14,000 gpm
Number of primary coolant pumps	2
Pump horsepower	200
Number of heat exchangers	1
Total heat exchanger area	2010 ft ²

Pressure in the reactor vessel and coolant loop is maintained and controlled by a pressurizer system. This system consists of a pressurizer vessel connected to the reactor outlet piping by a 4 in. static water leg. The pressure is regulated by automatically controlled immersion heaters which maintain a constant pressure steam dome in the pressurizer vessel. The immersion heaters have a combined output of 200 kw and are wired for staged operation. A pressure control valve

plus two safety relief valves located on the pressurizer serve to protect the primary system from excess pressure. The control valve is also used to bleed non-condensable gases from the vessel during startup and operation, and to reduce the system pressure during shutdown.

The primary coolant system is initially filled with demineralized water by means of the demineralized water service pump located in the utility building. After initial filling, makeup is supplied by means of a small capacity high pressure makeup pump (5 gpm, 5750 ft-hd, 10 hp). The excess water during heatup is discharged through a control valve in the reactor inlet piping to the drain header. The addition of makeup water and blowdown of excess water are controlled from the pressurizer liquid level controller in order to maintain a near constant volume of water in the primary system.

The two primary pumps are canned rotor, single stage centrifugal pumps rated at 7000 gpm at a temperature of 497°F with a suction pressure of 2500 psi and a discharge pressure of 2535 psi. Each pump is driven by a 200 hp motor. A single emergency pump (800 gpm, 25 ft-hd, 7-1/2 hp) is connected in series with the primary pumps to provide emergency cooling. The emergency pump operates during reactor operation and is connected to the diesel power system to provide cooling in the event of a commercial power failure.

The primary heat exchanger is a single pass shell and tube exchanger with fixed tube sheets. Differential expansion between the shell and tube side is provided by an expansion diaphragm located in the shell. Primary heat exchanger characteristics are listed in Table IV-f.

Since the primary system is to be constructed of carbon steel, satisfactory operation can be obtained only by maintaining the primary coolant at a high pH. In addition, a method of removing fission products from the primary coolant is needed in the event of a premature fuel element failure. A bypass demineralizer system is provided for this purpose. Primary coolant at 497°F is diverted from the primary header and routed through a booster pump (120 gpm, 230 ft-hd, 1 hp). Flow is routed through a regenerative heat exchanger and a secondary ion exchange system heat exchanger where the temperature is reduced to

130°F. Three 14-in. diameter mixed bed lithium hydroxide ion exchange columns are provided giving a maximum combined output of 120 gpm. From the ion exchange columns the water is returned to the primary system through the shell side of the regenerative heat exchanger where the temperature is increased to 350°F before mixing with the primary coolant. The pH of the system is thus maintained between 9.0 and 10.0. The ion exchange columns are designed for remote charge and discharge of resin.

TABLE IV-f

PRIMARY HEAT EXCHANGER CHARACTERISTICS

Heat transfer area	2010 ft ²
Tube side fluid	Demineralized water
Shell side fluid	Dowtherm A
Tube length	10 ft
Tube outside diameter	3/4 in.
Tube wall thickness	14 BWG
Tube configuration	Triangular
Tube pitch	1 in.
Tube side and channel design pressure	2500 psig
Tube material of construction	304 stainless steel
Shell inside diameter	42 in.
Shell design pressure	50 psig
Baffle type	Segmental
Shell material of construction	Carbon steel

Carbon steel is selected as the material of construction for the primary system. This selection is made on the approximate equal ultimate strength of carbon steel versus stainless steel at 500°F, short life of the system, lower initial cost and proven decontamination methods for carbon steel.

Nozzles for conducting the loss of coolant tests are located on the inlet and outlet primary headers adjacent to the pressure vessel.

The nozzles are designed for the insertion of variable sized rupture disks up to an area equivalent to the internal area of the largest primary pipe (20 in.). The exact method for initiating a rupture will be determined from the results of the proposed development program.

Although the rupturing devices will be developed by Phillips Petroleum Company, the resultant forces from coolant blowdown will be negated through design by the Architect-Engineer.

2. Reactor Secondary and Tertiary Systems (see Fig. IV-7)

Heat generated in the reactor is transported from the primary heat exchanger through the secondary Dowtherm loop to the tertiary coolant system containing the cooling tower where it is released by evaporation to the atmosphere. Utilization of the secondary Dowtherm system permits a compact primary system completely mounted on the railroad flatcar, plus an intermediate temperature reduction step.

The total Dowtherm coolant flow is 16,000 gpm. Dowtherm circulates on the shell side of the primary heat exchanger at an inlet temperature of 200°F and an outlet temperature of 250°F. The heat in the Dowtherm is dissipated to the tertiary water system in the two Dowtherm heat exchangers. Characteristics for these heat exchangers are listed in Table IV-g.

Two single stage centrifugal pumps (8000 gpm, 60 ft-hd, 200 hp) connected in parallel are used to circulate the Dowtherm through the loop. An emergency Dowtherm circulating pump (900 gpm, 40 ft-hd, 15 hp) is installed in series with the main Dowtherm circulating pumps to provide shutdown cooling. The emergency pump operates during reactor operation and is connected to the diesel power system to provide cooling in the event of a commercial power failure. The Dowtherm loop is maintained at 15 psig by means of a pressurized nitrogen blanket above the surge tank level.

TABLE IV-g

SECONDARY HEAT EXCHANGER CHARACTERISTICS

Heat transfer area	2100 ft ²
Tube side fluid	Dowtherm A
Shell side fluid	Raw water
Tube length	18 ft-4 in.
Tube outside diameter	3/4 in.
Tube wall thickness	20 BWG
Tube configuration	Triangular
Tube pitch	1 in.
Tube side and channel design pressure	50 psig
Tube material of construction	Seamless carbon steel
Shell inside diameter	26 in.
Baffle type	Segmental
Shell material of construction	Carbon steel

The total tertiary flow is 7000 gpm which includes 1000 gpm of utility cooling water. Tertiary coolant flows from the Dowtherm heat exchanger to the cooling tower where the on-tower and off-tower temperatures are 140°F and 85°F, respectively. Two vertical turbine pumps (3000 gpm, 125 ft-hd, 150 hp) circulate the tertiary water to the shell side of the Dowtherm heat exchangers.

An induced draft redwood cooling tower consisting of three cells is provided for final heat dissipation to the atmosphere. Each cell is 30 ft long, 36 ft wide and 40 ft high. Each cell contains a fan driven by a two-speed 50 hp motor. The fans are reversible in low speed to permit deicing operations during the severe winter weather. The tower was selected on the basis of the following parameters:

7000 gpm maximum flow
 2×10^8 Btu/hr heat load
65°F maximum wet bulb
85°F off-tower temperature
140°F on-tower temperature
17°F approach

The utility and emergency cooling (UCW) pump (1000 gpm, 125 ft-hd, 40 hp) is used for emergency tertiary heat removal from the Dowtherm heat exchanger and for final heat removal by the cooling tower for the reactor auxiliary systems. These include flash tank cooling and tertiary coolant to the secondary ion heat exchanger. The UCW pump operates during reactor operation and is connected to the diesel power system to provide cooling in the event of a commercial power failure. The cooling tower makeup requirements are based on maintaining four cycles of concentration* in the cooling tower water. MTR-ETR experience has indicated that seven cycles of concentration are satisfactory at a water temperature of 110°F. However, the LOFT tertiary on-tower water temperature is 140°F, and it is deemed advisable to limit the maximum cycles of concentration to four, thereby minimizing scale formation in the heat exchangers.

Makeup water treatment is required to inhibit corrosion, scaling, growth of slime and algae and for the degeneration of the cooling tower redwood. When the reactor is operating at full power, it is estimated that 370 lbs of 66° Be sulfuric acid, 15 lbs of dianodic (a corrosion and scale inhibiting chemical), and 1.5 lbs of chlorine are required per day.

* "Cycles of concentration" is the term employed to indicate the degree of concentration of the circulating water as compared to the makeup water.

F. Emergency Coolant Systems

1. General

Three emergency cooling systems are provided to dissipate the nuclear heat following a commercial power failure or an unanticipated loss of coolant. The coolant systems represent typical methods of supplying emergency coolant to the core region on various reactor systems presently in existence.

The emergency cooling systems are (1) Emergency Water Spray System, (2) Emergency Steam Spray System, and (3) Decay-Heat Removal System. Systems 1 and 2 will be primarily used for follow-on loss of coolant accident studies where the effectiveness of in-tank spray systems are evaluated in controlling the magnitude of core meltdown. Hopefully, the water spray or steam spray systems will limit the amount of fuel damage, potential metal-water reactions and the subsequent release of fission products to the primary system volume.

2. Water Spray System*

The emergency water spray system consists of a sparger ring located within the pressure vessel above the core with nozzles for directing a water spray onto the core. Demineralized water is furnished to the sparger ring from the 150 gpm, 80 psi plant demineralized water system. The automatic control system is interlocked during normal operation with the reactor vessel water level and pressure instrumentation to preclude using this system until initiation of a loss of coolant accident.

* The water spray and steam spray systems will not be installed in the reactor vessel for the initial experiment. However, headers for each system will be supplied to the coupling station and capped for future emergency systems.

3. Steam Spray System*

The emergency steam spray system includes a high pressure steam header located below the core with nozzles for directing steam onto the core. High-pressure steam (125 psi) is furnished to the steam header from the two plant 5000 lb/hr water tube boilers. The system is also interlocked with the reactor vessel water level and the pressure instrumentation to preclude using the system until the core is entirely void of water.

4. Decay Heat Removal System

The decay heat removal systems are in integral part of the three reactor heat dissipation systems. The primary and Dowtherm emergency pumps are connected in series with their respective main circulating pumps. Each emergency pump operates during reactor operation and is connected to the diesel power system to provide cooling in the event of a commercial power failure.

A 1000 gpm utility and emergency coolant (UCW) pump provides emergency cooling of the secondary Dowtherm heat exchanger in the event of a commercial power outage. Normally the UCW pump provides cooling water to all the reactor auxiliary systems and equipment, i.e., refrigeration, compressed air, thermal sleeves, diesel generator, etc. Power is supplied to the 60 hp UCW pump motor by the diesel generated power system.

* The water spray and steam spray systems will not be installed in the reactor vessel for the initial experiment. However, headers for each system will be supplied to the coupling station and capped for future emergency systems.

G. Instrumentation

1. General

The reactor control, experimental and process instrumentation responsibility will of necessity be divided between the Architect-Engineer and nuclear designers of the loss of coolant experiment (Phillips Petroleum Company). It is presently envisioned that Phillips will be responsible for the experimental instrumentation with the exception of the building monitoring and sampling systems. Therefore, in the following discussions, experimental instrumentation descriptions are intentionally brief and are presented for information purposes only.

The reactor control system to be designed by the Architect-Engineer is typical of systems used in the standard power reactor facilities with the exception that automatic power control will be less elaborate and automatic program sequencing equipment will be added for initiating the experimental rupture and the control of appropriate photographic and experimental data recording equipment.

2. Reactor Control

Either or both of two types of reactivity control may be used, i.e., mechanical control rods, which may be withdrawn from or inserted into the core, and neutron absorbing chemicals, which may be dissolved in the cooling water in variable concentrations. The discussion which is to follow is applicable to either type of reactivity control.

Reactor control may be discussed in three phases; (1) manual control to accomplish startup, criticality, and operational power level, (2) servo control for automatic adjustment of reactivity to compensate for short term fluctuations at operational power level, and (3) safety system control to minimize damage to the facility resulting from operator and equipment malfunctions.

a. Manual Control

A reactor control console will be provided with the necessary control features, limit, position, and magnitude indicators to

permit a trained operator to control independently each of the reactivity control devices. Increase of reactivity by manual control will be subject to interlocks requiring certain monitoring instrumentation to be operating on scale in proper ranges. Manual control will override the automatic servo control which will be discussed below, but will be subject to being overridden by the safety system, also discussed below. Manual control will be used to accomplish all long term reactivity adjustments, unless dissolved chemical poison is chosen as the sole means of control, using a servo system similar to that proposed for the Advanced Test Reactor (ATR). In a design using shim rods and regulating rods, only the regulating rods would be servo operated to provide automatic regulation of power level.

b. Servo Control

Servo control will be provided for operation at power. The range of automatic control may be limited to account for little more than the short term fluctuations and fuel burnup expected in an eight hour period. As the precise control normally used in test reactors and on-line power reactors is not required here, an on-off type controller design should be adequate for maintaining constant power level. The operator will effect selection of "control region" within the core.

c. Safety System Control

The safety system will monitor independently all system parameters which relate to safety of personnel and the facility, and automatically take appropriate action when required. Nuclear safety parameters may include: power level, period of e-fold power multiplication, core temperature, and fuel element integrity. Process parameters may include: coolant temperatures, coolant pressures, coolant flow rates and coolant composition. Action of the safety system will be appropriate to protect the facility as completely as possible against all eventualities except the "planned accidents" for which it is designed. This reactor is to be typical of power reactors which are generally difficult or impossible to restart without refueling if a shutdown occurs in an advanced stage of core burnup. For this reason, the "appropriate" action taken upon the sensing of a potentially hazardous condition is not

arbitrarily to "scram" the reactor but to provide maximum opportunity for correction of the condition without affecting operation of the reactor. Thus, existence of all potential hazards will first be manifested by audible and visible alarms to alert the reactor operator. If a potential hazard should develop into an actual hazard, then the safety system must take additional action to reduce reactor power to a level which may still be considered safe. Depending upon the extent of a hazard, the safety system action may be a "setback", i.e., automatic lowering of the set point of the power level servo control system, a "reverse", i.e., power reduction by automatic full speed operation of all reactivity controlling mechanisms, or a "scram", i.e., sudden complete shutdown by auxiliary safety rod insertion and/or chemical poison injection.

3. Reactor Instrumentation

Instrumentation shall be provided to monitor reactor power over the entire range of operation from source level to well above the expected operating level of the reactor for four modes of control; i.e., startup, manual operation at power, servo operation at power, and safety system operation. All operational instrumentation will be energized by failure-free power sources in such a manner that failure of commercial power will not interrupt operation of the instrumentation. Redundancy will be provided so that at least two independent indications are available for each measured parameter.

a. Startup Range

The startup instrumentation consists of two low-level log count-rate meter circuit channels. Monitoring of the neutron flux will be possible in the cold, clean reactor at source power. A minimum count rate is required prior to withdrawing the control rods for reactor startup. These channels range from source level to within four decades of the power range, in adjustable increments of four decades. The control rods are interlocked to prevent operation unless one channel is on scale.

Provisions will be made in the water filled shield tank for inserting the startup detectors. A re-entrant tube into the reactor as well as a remote withdrawal device may be required for one startup channel.

b. Intermediate Range

The intermediate range instruments also employ duplicate independent channels and monitor neutron flux in the intermediate portion of the flux range. These channels provide coverage for approximately seven decades of power level and overlap the upper portion of the source range and lower portion of the power range. A boron-lined compensated ionization chamber is used for the detection of thermal neutrons. The log current and period computer convert the current from the ionization chamber to a voltage proportional to the logarithm of the current. Differentiation of this voltage provides a signal inversely proportional to reactor period. Log current and period signals are transmitted to indicators and recorders in the control room. Placement of intermediate range channels will be similar to startup channel placement.

c. Power Range

This system is composed of three identical channels that monitor the neutron flux level from 1% to 300% of full power. At least two of the three channels must be in operation or scram action is initiated.

The power range circuit receives a signal proportional to the thermal neutron flux from the uncompensated ion chamber. The uncompensated ion chambers are located in the water filled shield tank.

Calorimetric instrumentation will provide the primary measurement of reactor power. Coolant flow is monitored together with reactor vessel inlet and outlet temperatures. From this information, power level is continuously computed by electronic equipment and will be used to calibrate neutron flux instrumentation.

Provisions also will be made for additional flux measurements at various locations within the reactor core by means of small re-entrant tubes through which miniature ionization chambers may be inserted or

coolant water may be quickly sampled to measure flux level using N_{16} gamma decay techniques.

All of these indications as well as level and period information from the safety instrumentation will be at the disposal of the console operator to facilitate startup, criticality, and attainment of operational power levels.

d. Servo Control

Inherent rapid response and directness of measurement provided by the neutron flux measurement instrumentation at operating power level dictate that the servo control system be operated from these measurements to reduce overshoot and instability problems. The set point of the servo control will be adjusted manually on the basis of calorimetric power measurements.

e. Safety System

The safety system incorporates instrumentation to provide nuclear information, process information, and miscellaneous information independent from all other monitoring systems.

Neutron pulse chambers and gamma-compensated ionization chambers provide signals to the safety system to indicate neutron flux continuously over the entire range from source level of the cold clean reactor to 300% of design power.

Power level information is provided to the safety system by microammeter amplifiers, the outputs of which are proportional to the logarithms of the ionization chamber currents. Period information is provided by additional circuits in these same amplifiers whose outputs are proportional to the derivatives of the logarithmic outputs. Trip set points are provided to give alarms, setbacks, reverses, and scrams at various unsafe levels of power and period.

Fuel element temperature information will be provided to the safety system by appropriate thermocouples.

Process information will be provided by instrumentation to monitor coolant temperatures, coolant pressures, fission products resulting from fuel element failure, coolant flow rate, etc. with trip circuits included to activate alarms and power reductions when set points are reached indicating unsafe operating conditions.

Miscellaneous safety equipment will also include appropriately placed and guarded manual scram buttons in the facility which will permit personnel to shut down the reactor in event of an emergency.

4. Experimental Instrumentation

a. Temperature

Fast response thermocouples and resistance thermometry techniques are used for measurement of fuel rod surface and meat temperatures, coolant channel temperatures, expulsion region temperatures, and piping temperatures in the region of rupture. In addition, temperature measurements are made within the containment vessel, including the wall surfaces, to obtain a spacial temperature distribution.

b. Pressure

Sufficient pressure detectors, internal to the core, internal to the pressure vessel, and external to the pressure vessel, are employed to measure space and time gradients of acoustic and afterflow pressure generation. The detectors are of a type and so protected as to be insensitive to transient temperature and radiation effects. In addition, the detectors respond to a broad frequency range for indication of shock generation or buildup. Within the core the pressure detectors are located to aid in determining directional effects for correlation to disassembly and meltdown patterns.

c. Radiation Detection

Standard fast response, thermal and fast neutron, and gamma detectors are employed to measure the gross nuclear time-history as seen external to the vessel. In addition, high level miniature flux monitors are placed within the core for power distribution measurements.

d. Expulsion Flow Rate

Fast response drag disk and air anemometry techniques for mass flow indication, and impeller and ultrasonic techniques for velocity indication are employed for qualitatively determining the character and time-history magnitude of coolant expulsion. Glass ports in certain portions of the piping also permit high-speed photography to reveal quality and velocity characteristics of the coolant upstream from the rupture.

e. Strain Gauges

Strain gauge instrumentation is applied to the fuel rods for internal rod pressure measurements, to core and structure components for thrust and acceleration measurements, and to the vessel for strain energy absorption correlation with the acoustic and impact-flow pressure measurements. In addition, strain instrumentation is applied to the primary piping for overall system displacement and strain energy absorption indication.

f. Dynamic Effects

Accelerometers and fast response load cells are utilized for determining overall system displacement effects.

g. Photography

Photographic techniques are employed to study rupture generation and artificial safety or quenching equipment time-history behavior.

h. Water-Steam Measurements

Selected portions of the vessel and piping are used to accommodate ultrasonic techniques for determining the time-history of coolant phase change characteristics.

5. Process Instrumentation

Since remote operation of the reactor from the control and utility building is essential, it is required that the plant instrumentation be designed to permit operation of certain plant equipment and control of certain process variables from the process control room. The equipment requiring remote operation is primarily that which is mounted on the railroad flatcar or located in the test building. Process variables are recorded and displayed in the appropriate locations. Since this instrumentation is fairly standard and straightforward as related to reactor operation and control, it is not detailed here.

In addition to the process instrumentation there will be fission break monitoring, stack gas monitoring and health physics instrumentation. The fission break monitor measures direct radiation increase from a side stream of the primary coolant. It is expected that a technique which measures increase in count rate is satisfactory for this measurement as compared to the integrating technique used on the stack. The stack monitor utilizes scintillation counters with totalizers to record both gas and particulate activity.

The health physics instrumentation consists of constant air monitors which have sampling stations in critical regions throughout the reactor area. These stations alarm at the remote locations and indicate as well as alarm at a central station. Other health physics instrumentation, such as the portable instruments to measure various radiation types, are also available. There will also be remote area radiation monitors to measure and indicate the radiation intensities at strategic locations throughout the reactor area.

The Architect-Engineer shall provide for the process instrumentation as set forth above. Although Phillips Petroleum Company will be responsible for design of the above experimental instrumentation the Architect-Engineer shall provide the necessary services to assure a complete operable system. Such services shall consist of junction boxes, conduit, pressure connectors, power and such other items as may be required to integrate the Phillips Petroleum Company design into the completed facility.

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Fig. IV-1 - Core Cross Section

Fig. IV-2 - Fine Flux Variation Within Fuel Pin

Fig. IV-3 - Calculated K_{eff} versus UO_2 Enrichment

Fig. IV-4 - Eigenvalues

Fig. IV-5 - Power Traverse

V. EXPERIMENT CONCEPTS FOR OTHER REACTOR SYSTEMS

A. General

Safety testing of other reactor types has to be considered prior to establishing criterion for a facility to accommodate the loss of coolant accident studies. Since the engineering and safety test program is not limited to any one type of nuclear system or a single accident, the facility to be constructed must be extremely flexible and adequately designed to allow testing of other reactor concepts of interest in the Commission's reactor power program. It must be recognized, however, that appurtenances, auxiliary systems, and services unique to a selected nuclear system will necessarily be required at the time the specific experiment is authorized. For example, should it be expedient to investigate the consequences of an accident associated with a sodium-graphite reactor system, sodium storage and transfer systems would be required at the test facility or in the Examination Area to allow safe handling, charging of the nuclear system, and subsequent disposal following the destructive test.

A technical feasibility study for conducting field-scale engineering tests on other reactor class prototypes⁽³⁵⁾ will be published in June of 1963. The preparation of the feasibility studies proceeded concurrently with the development of the loss of coolant experimental program and facility conceptals to insure siting, building and system compatibility with other nuclear systems and postulated accidents. Experimental safety programs which appear to be needed for gas-cooled, fast, sodium-graphite, and organic-cooled and -moderated power reactor systems were developed after a review of existing safety analysis reports. The proposed programs were then analyzed to determine the feasibility of conducting the experiments at the NRTS in the proposed LOFT facility. Analysis of all the reactor concepts and experimental programs indicates the most severe facility requirements and radiological hazards are attendant to the loss of coolant accident on a pressurized, water-cooled and -moderated power reactor system. Therefore, with the exception of auxiliary systems and probable coolant handling systems, the proposed LOFT test facility will accommodate the prominent reactor technology in the present United States power program.

For the purposes of this report, the various reactor types and the experimental programs which might potentially be performed in the proposed LOFT facility are summarized in the following sections.

B. Gas-Cooled Reactor System

The status of the gas-cooled reactor systems in the United States can be described as advancing in technology and development. As yet, the United States has not operated a gas-cooled reactor system for other than experimental or demonstrative purposes. The early British Calder Hall type of gas-cooled commercial nuclear power plant had low power densities and was so large that the United States found them economically unfeasible. Recent improvements in power densities, resulting from development of higher temperature fuels and claddings, has made the gas-cooled reactor much more attractive, and the United States has accelerated the developmental program for this type reactor system.

The reactors that have been and are being developed are designed for two different nuclear power applications: commercial power stations and special purposes such as aircraft propulsion, ship power, and portable electrical power.

The gas-cooled reactors which are being developed for central power station applications have several distinct characteristics which result largely from the economic requirements necessary for competitive nuclear power. The fuel is of low enrichment, resulting in relatively low power densities, and the use of graphite moderator is common. In order for the power output to be significant, large volumes of the low heat capacity gas coolant must be circulated through the reactor. The overall result is that the gas-cooled reactor systems for commercial power applications are very large compared to a water system.

The special-purpose gas-cooled reactor systems are much smaller than those designed for commercial power stations. They are constructed of highly enriched fuel, clad with high-temperature materials and do not use graphite for a moderator.

Because of the two distinct applications of gas-cooled reactors which result in the reactor systems being quite different, the major accidents will be summarized separately.

The use of gas-cooled reactor systems for commercial power applications has only recently progressed past the conceptual design phase in the United States. The Experimental Gas-Cooled Reactor (EGCR), designed by Kaiser Engineers and Allis Chalmers Manufacturing Company, is being constructed at Oak Ridge National Laboratory (ORNL). The High Temperature Gas-Cooled Reactor (HTGR) designed by General Atomics has been authorized, and construction has started at Peach Bottom, Pennsylvania. Both of these systems use a helium coolant and graphite moderator. The EGCR, which is rated at 84.3 Mw(t), will have a UO₂-stainless steel clad fuel element and the HTGR, rated at 112.5 Mw(t), will have a uranium carbide and thorium carbide fuel in a graphite can.

Preliminary hazards summary reports have been issued for these installations, both of which employ total reactor containment. A review of these preliminary reports showed that the major credible accidents were reactivity accidents and accidents induced by system component failures. The reactivity accidents associated with these reactors were: (1) control rod accidents, (2) steam or water entering the core, (3) control rod falling out of the core, (4) cold coolant, and (5) improper fuel handling accidents. The associate component failures were: (1) loss of blowers, (2) steam tube rupture, (3) pressure relief valve failure (primary or secondary system), (4) primary coolant system rupture, and (5) fuel element failure. In both cases the maximum credible accident was a result of a system depressurization caused by a rupture of the main coolant system piping.

The basis for design of the containment shells is a hypothesized maximum credible accident. For the EGCR the primary coolant system rupture was further complicated by a rupture in the secondary system, resulting in the expulsion of the steam and water in one steam generator and the cladding rupture of one fuel element. The theoretical pressure peak attained in the containment shell would be 9 psig for this accident. The possibility of exothermic chemical reactions aggravating the accident was not considered credible for the EGCR.

The HTGR containment shell was designed to contain a maximum credible accident which involved a series of failures. The accident involves the release of the primary system helium followed by a rupture of one steam generator tube.

This event is accompanied by the immediate total chemical reaction of the released water and steam with the reactor core graphite assuming the complete conversion to carbon monoxide and hydrogen, which are then released at 660°F to the containment. The accident is further complicated by subsequent chemical reactions of the remaining water, steam, oxygen and carbon dioxide in the container atmosphere with the core graphite at a rate dictated by natural convection through the reactor vessel. The highest peak containment pressure resulting from the multiple accident and based on the conservative assumptions was 8.0 psig at 150°F, which was used as the containment shell design pressure.

The concept of utilizing gas-cooled reactors for special purposes was first utilized in the now-defunct Aircraft Nuclear Propulsion (ANP) program. This was followed by the development and operation of the Army's Gas-Cooled Reactor Experiment (GCRE) and the prototype Mobile Low-Power Reactor (ML-1) by Aerojet General Corporation, the development of the Maritime Gas-Cooled Reactor (MGCR) by General Atomics Corporation, and the recent adaptation of the ANP HTRE-3 experimental reactor to a maritime application concept, designated the 630-A, by General Electric Company.

The GCRE [2.2 Mw(t)] and ML-1 [3.3 Mw(t)] reactors use a nitrogen coolant, water moderator and Hastelloy-X clad UO₂ plate type fuel. The MGCR [74.4 Mw(t)] uses a helium coolant, BeO moderator, and Hastelloy-X clad UO₂-in-BeO pellet fuel, and the 630-A [72 Mw(t)] has an air coolant, water moderator and nichrome clad UO₂-in-nichrome fuel.

The only hazards report reviewed for the special purpose reactors was the report concerned with the GCRE reactor. However, an early report of the hazards associated with the MGCR is available.

The MGCR report considered (1) startup accident, (2) cold coolant accident, (3) flooding the core with water, (4) loss of coolant accident, and (5) fuel element failure. This report concluded that the reactor is inherently safe provided that the reactor has a shutdown margin larger than the reactivity increase introduced by flooding the core and that emergency cooling could be initiated within 17 minutes to prevent core meltdown.

The GCRE hazards summary report considered (1) control system failure, (2) accidental flooding of the core, (3) addition of fuel accidentally, (4) fluid temperature change, (5) loss of coolant flow, and (6) loss of coolant. The report stated that the maximum credible accident would be initiated by a rupture of the inlet or outlet plenums of the primary coolant lines accompanied by a complete control rod failure at a cold near-critical condition. The postulated accident would culminate in a nuclear excursion and metal-water reactor (from melting and vaporizing fuel) resulting in an energy release equal to approximately 38 lbs of TNT. The GCRE does not use the total containment concept.

Because no one type or size of gas-cooled reactor has been used for other than experimental or demonstrative purposes, it is difficult to select a particular reactor on which to perform safety tests. However, certain constituents or applications of the reactors provide an insight into the types of tests which might be necessary in the future.

The commercial power reactors which use a graphite moderator might require a safety program in which a loss of coolant accident would be performed in order to prove the relatively low radioactivity and energy release from the coolant itself, and to investigate the possibility and extent of the exothermic graphite, cladding, or fuel oxidation. In addition, an investigation of the steam-graphite reaction would be of interest.

The LOFT containment vessel appears to be capable of containing the coolant and energy release from a maximum credible accident of a 50 Mw(t) commercial type gas-cooled reactor system. However, both the containment vessel's aperture and the flatcar would be too small physically for this reactor system. In order for a system of this type to be installed on the flatcar and housed inside the LOFT containment vessel, a scaledown factor of approximately five of either the EGCR or HTGR systems would be necessary. A detailed analysis entailing rigorous nuclear and heat transfer calculations is necessary to determine a more exact scaledown factor for these reactors.

The special purpose reactors, according to the hazards reports, offer virtually no hazard in a loss of coolant accident resulting from the coolant release and chemical reactions. However, the possibility of water entering and flooding the reactor is credible and could effect a reactivity accident if the shutdown margin is not adequate. A loss of coolant accident could be performed which would incorporate water flooding of the core. This test can be performed in the LOFT facility provided: missile protection is installed to contain the explosive potential of a power excursion, and a device is installed for flooding the reactor. The nuclear package can easily be installed on a flatcar.

Exothermic reactions between the moderator or cladding and air or other emergency coolants are not considered to be hazardous because of the selection of materials for the special purpose type reactors. However, a loss of coolant will result in core melting and a consequent fission product release. An additional hazard associated with the loss of coolant accident for this type reactor, specifically with maritime application, is the possibility of water entering the reactor and causing a reactivity accident. Thus, it may be of interest to incorporate the flooding of the reactor in a test program designed to demonstrate the effect of a loss of coolant accident.

In conclusion, the LOFT facility design appears to be adequate for conducting safety tests with gas-cooled reactor systems provided some scaledown of the reactor systems are feasible.

C. Fast Reactor System

Only two fast power reactors are being built in the United States at the present time, the Enrico Fermi reactor and the Experimental Breeder Reactor II. These two reactors are similar in design, and the hazards analyses for both reactors were considered for determining the hazards of the proposed prototype tests in the LOFT facility.

The fast reactor proposed for the LOFT tests is similar in basic design to the EBR-II and Enrico Fermi reactors. It will be an unmoderated, sodium-cooled, fast reactor and may include the breeding concept. Other types of fast reactors are being investigated, but they are as yet in the experimental stages.

Accidents which might occur from a fast reactor system come under two general categories: (1) nuclear accidents, and (2) system-failure accidents. Nuclear accidents are those associated with reactivity insertions such as: startup transients, accidental control rod motion, step reactivity insertions, introduction of moderator into the core, gas entrainment, and meltdown criticality. System-failure accidents include loss of flow, blocking of flow, and loss of coolant. Other accidents include sodium coolant reactions with either water or air caused by leaks in the primary system.

The accident recognized as the most hazardous for both power reactors is the meltdown accident caused by the loss of coolant. In this accident, it is assumed that part of the center of the core becomes molten and flows toward the bottom of the core. The top of the core then drops causing a prompt critical nuclear excursion, which may be followed by a violent sodium-air reaction if the primary system is breached.

The containment vessel for LOFT is similar to that for the EBR-II, and is adequate containment for the combination of the nuclear excursion and sodium fire. The off-site radiological hazards from this type of release should not be significantly different than the release from a pressurized water reactor. However, this is being investigated.

The proposed STEP tests on the fast reactor system are to test the nuclear and system-failure aspects of the system and to determine the hazards associated with sodium coolant reactions and their effects on fission product release and dispersal. The first series of LOFT tests should involve only the nuclear and system-failure tests with no sodium-air reaction. Because of the cleanup problems associated with a sodium-air reaction, the characteristics of this phenomenon should be investigated by small-scale or laboratory type experiments. A full-size prototype fast reactor can be tested in LOFT, provided power level is limited to approximately 50 Mw(t).

The core meltdown resulting from a system failure is considered to present the maximum energy and fission product release to the containment shell. From the evaluation reports for EBR-II and Enrico Fermi, it is concluded that this test can be performed safely in the LOFT

facility by providing adequate protection against missiles. Static, kinetic, and transient tests simulating different type accidents will be performed. During some of these tests, partial melting of some fuel elements may take place, thus releasing fission products to the primary coolant. However, this release should not exceed that postulated for the core meltdown tests.

A reactor facility equivalent to either EBR-II [63 Mw(t)] or Enrico Fermi [300 Mw(t)] can be installed on the railroad dolly and is compatible with the LOFT containment and hot shop facilities. Remote handling techniques and remote coupling, cleanup, and decontamination equipment proposed for LOFT will be adequate for the tests.

The LOFT facility as presently proposed is capable of handling the fast reactor test series provided some additional equipment can be installed. The presence of large amounts of liquid sodium metal constitutes a serious fire hazard both in the facility and in the hot shop. A fire protection system to extinguish large sodium fires will be necessary in both of these locations. Water will not be allowed in the vicinity of the reactor if a large amount of sodium is present. If a sodium-air reaction were to occur, a vacuum might be formed in the containment vessel. To prevent damage to the vessel, a relief valve may be installed on the dome to relieve the vacuum. Additional storage and handling equipment for liquid sodium must be installed along with a heating system to keep the sodium coolant in the liquid phase. All handling of sodium coolant will have to be done under the cover of an inert gas to prevent contact with air. An inert gas system will then be necessary both in the LOFT facility and the hot shop.

A fuel handling system for remotely removing and replacing irradiated fuel in the reactor must be designed and installed in the hot shop. This equipment must also operate under an inert gas blanket. Hot shop inspection equipment must be modified to allow fuel inspection under these conditions. Equipment for cleaning sodium metal from irradiated fuel must be designed, and a method of storing this fuel must be found.

In summary, the LOFT facility appears to be adequate for conducting the series of fast reactor tests, including the meltdown excursion.

D. Organic Reactor System

The organic reactor concept has received attention only recently and, therefore, has been subject to less development than other major reactor programs from the viewpoint of both commercial power application and safety aspects. At the present time, the sole power reactor is the 45.5 Mw(t) Piqua Organic Moderated Reactor (Piqua OMR), which is a small prototype power reactor. Consequently, the source of hazards evaluation for organic-type power reactors is exclusively the hazards analysis which was conducted by Atomics International for the Piqua OMR.

The Piqua OMR incorporates the organic-cooled, organic-moderated concept. Other combinations of organic as either the coolant or moderator material have been suggested, particularly by the Canadian and Euratom Atomic Energy organizations, with the Canadian organic cooled, heavy water moderated reactor now under construction.

The major credible accidents which are hypothesized for an organic-type reactor system are as follows: (1) startup transient, (2) rod withdrawal at power, (3) step-function reactivity insertion, (4) loss of coolant pressure, (5) loss of coolant flow, (6) water leak into organic-coolant system, (7) organic-coolant system leak, (8) rupture of steam system, and (9) organic reactions.

From the above list of credible accidents, the maximum credible accident (MCA) is designated as that incident which presents the greatest level of associated radioactivity. The MCA for Piqua OMR is the uncontrolled reactor startup transient. This hypothesized incident is initiated by the maximum permissible uncontrolled withdrawal rate of control rods from zero reactor power. The subsequent events involve a power excursion and core heatup. Ultimate termination of this incident is meltdown of a small portion of the core so that a sub-critical assembly prevails. The associated release of core fission products is circulated and maintained within the primary coolant system.

The proposed STEP test program for organic-type reactor systems is twofold. One program involves the simulation of loss of coolant pressure, loss of coolant flow, and startup-transient accidents utilizing an integrated organic-cooled, organic-moderated reactor system. This

reactor system can be installed on a railroad flatcar, and the tests can be conducted in the LOFT facility. The other program involves the combustion of irradiated organic coolant to determine the radioactivity release behavior of burning organic material. This combustion program will require a remote fire-box installation, which does not demand elaborate design features or connection with the LOFT facility.

No serious hazards are evident in connection with the proposed organic-type reactor test program. Only the uncontrolled startup-transient test will involve a possible release of fission products to the organic coolant; however, if core destruction and release of fission products occur, the radioactivity will be maintained within the primary coolant system. The calculated energy release for the uncontrolled startup-transient test should not create a condition which seriously endangers the integrity of the primary coolant system and pressure vessel assembly. Even in the remote possibility of a primary system rupture, the pressure buildup within the containment shell will be negligible due to high flash-point of terphenyl coolant. Although no serious contamination problem is evident, the radiation directed from the primary coolant system and pressure vessel can be substantial and may require remote operating and maintenance procedures for uncoupling, stabilizing, and transporting the reactor system. Another potential danger is that the core may be dismantled to such a degree that control rods can not be re-inserted following the uncontrolled startup-transient test. In the event that control rods can not be re-inserted into the core and that partial core meltdown accomplishes only a sensitive subcritical configuration, the subsequent cooldown of the in-core coolant with its inherent negative temperature coefficient of reactivity may initiate criticality. Although certain hazards are conceivable in connection with the proposed reactor test program, an elaborate assessment of all potential hazards and incorporated safeguards should preclude any serious consequences.

The major additional LOFT and TAN hot shop facility requirements for an organic-type reactor system are an electrical supply for primary system heating and a water fire-protection system to safeguard against organic fires. No elaborate decontamination equipment will be required

since any fission products release will remain within the primary system and since manual methods can be applied to organic leaks. No facility provisions are recommended for drainage of organic coolant since the terphenyl material becomes a solid at normal room temperatures and therefore could cause trouble to any non-accessible drainage system. Removal of coolant from the primary system and reactor vessel prior to post-test examination in the hot shop can be accomplished by step-wise drainage into barrels; thereby, disposal of the radioactive source can be achieved in a normal fashion.

In conclusion, the proposed safety test program for an organic-type reactor system can be conducted with the assurance of ample safety to operating and environmental personnel and without any major alteration to the proposed LOFT facility and TAN hot shop other than addition of a water fire-protection system and electrical-heating system.

E. Sodium-Graphite Reactor System

The sodium-graphite power reactor concept has been advanced in this country by Atomics International. The Hallam Nuclear Power Facility (HNPF), a 240 Mw(t) large-plant power prototype, is scheduled to commence operation in 1963. Considerable information has been obtained since 1957 from the Sodium Reactor Experiment (SRE), a 20 Mw(t) experimental power reactor. Very little information is available concerning the Russian 50 Mw(e) sodium-graphite power reactor, Volga.

As a result of limited development, the evaluation of safety aspects which are associated with the sodium-graphite concept has received less attention than the comparable evaluation for water-type reactors. The hazards analyses conducted for Hallam NPF and the experimental data attained from SRE are the exclusive sources of information for assessment of sodium-graphite reactor hazards.

The major credible accidents which are hypothesized for sodium-graphite reactors are as follows: (1) fuel-handling incident, (2) rupture of control-rod thimble, (3) meltdown of in-core fuel element, (4) sodium spill, (5) sodium fire, (6) sodium-water reaction, (7) startup transient, and (8) step-insertion of reactivity. From the above list of credible accidents, the maximum credible accident (MCA) is designated

as that incident which presents the greatest level of associated radioactivity. For example, the MCA for Hallam NPF is assumed to be a fuel-handling incident. This hypothesized incident involves the dropping of a spent fuel element from the fuel-handling machine onto the floor of the reactor building.

However, a fuel-handling incident is associated directly with the fuel-handling equipment design and operating procedures, not with a particular reactor classification. Likewise, the rupture of a control-rod thimble which could release fission gases to the reactor building and the meltdown of an in-core fuel element as the consequence of foreign matter in the primary coolant are accidents which are not related directly with an particular reactor type.

Two independent STEP test programs are proposed for assessment of the more important, unknown safety aspects associated with the sodium-graphite reactor concept. One program involves the combustion of radioactive sodium to determine the radioactivity separation and release behavior of burning sodium. This program will entail a firebox facility which can be remote from the LOFT complex. Special sodium handling and storage equipment will be required to facilitate test operations and to ensure safety for test personnel. As previously stated, the prime purpose of the proposed combustion program is to resolve the uncertainties of radioactivity dissociation and transport from burning sodium. No attempt is proposed for the study of pressure and temperature transients resulting from a sodium spill since this feature has been field tested by Atomics International. The water-sodium reactions also have been analyzed extensively in laboratory tests.

The other more sophisticated test program involves the simulation of the startup transient and the step-insertion of reactivity accidents incorporating an integrated sodium-graphite reactor system. This reactor system can be installed on a railroad flatcar, and the testing can be conducted in the LOFT facility. The prime purpose of this proposed test program is to study the inherent shutdown mechanisms of a representative sodium-graphite reactor. These shutdown mechanisms are defined as the innate reactor characteristics which remain operative in the event that all exterior controls fail.

No serious hazards are foreseen in connection with the proposed sodium-graphite reactor test program other than the hazards which are normally associated with the notorious sodium coolant. The primary coolant system will contain the fission products released as the result of core melting. The compatibility of the sodium coolant with the fuel, moderator, and structural material precludes the possibility of chemical energy release. In the event of a primary system rupture and no fire, the pressure buildup within the containment shell will be negligible due to the high flash-point of sodium and the low operating pressure of the sodium coolant. If a sodium fire occurs, the pressure buildup and associated hazards will not be worse than those discussed for the sodium-cooled fast reactor. Although no serious contamination problem is foreseen, substantial radiation directed from the sodium coolant and suspended fission products may require remote operating procedures for uncoupling, stabilizing, and transporting the reactor system following destructive testing. Another potential danger which will prevail at all times, particularly during post-test examination in the TAN hot shop, is the fire hazard of sodium. Although potential hazards are evident in connection with the proposed sodium-graphite reactor test program, predetermined evaluation and safeguards should prevent any serious accident.

A major LOFT and TAN hot shop facility requirement which is associated with the sodium-graphite reactor system is an appropriate fire-protection system to safeguard against sodium fires. Bulk calcium carbonate in boxes should be located in strategic positions for the purpose of extinguishing small fires. Met-L-X extinguishers should be provided to subdue large sodium fires.

In conclusion, the proposed safety test program for a sodium-graphite reactor system can be conducted with ample safety if proper precautions are utilized. No major alterations are needed for the proposed LOFT facility and TAN hot shop if the size of the reactor system does not exceed 50 Mw(t).

VI. FACILITY CONCEPTUAL DESIGN

A. Facility Siting

1. General

The Test Area North (TAN) facilities are located within the National Reactor Testing Station (NRTS) boundary and are approximately 27 miles north-northeast of the Central Facilities area. The TAN area, formerly known as the Aircraft Nuclear Propulsion (ANP) area, is located at a latitude of $43^{\circ} 50' N$, longitude $112^{\circ} 41' W$, and at an elevation of 4790 ft⁽³⁶⁾. The TAN area is composed of five different facilities, one of which is the proposed Loss of Flow Test (LOFT) facility (see Figs. VI-1 and VI-2). The other four facilities are listed in Table VI-a along with their direction from LOFT. The LOFT facility, located 1-1/2 miles west and 1/2 mile south of the Lithium Cooled Reactor Experiment (LCRE) facility, consists of a containment test building, an underground shielded control building, and a process building (see Fig. VI-3). The control area and supporting utilities and services are located adjacent to the test building to eliminate remote operation on systems outside the test building. Since sustained reactor power operation is contemplated in the LOFT test program to accomplish fission product inventories, the control building will be manned at all times including during the destructive test series.

TABLE VI-a

TAN FACILITIES

	<u>Location</u>	<u>Direction & Distance from LOFT</u>	
a.	Lithium Cooled Reactor Experiment (LCRE), formerly Flight Engine Test (FET)	Northeast	1 mile
b.	Low Power Test (LPT)	East	4 miles
c.	Technical Support Facilities (TSF), formerly Administration & Maintenance Area (AMA)	East-northeast	3 miles
d.	Initial Engineering Test (IET)	Northeast	3 miles

The surface of the area is nearly flat with an elevation of 4800 ft at the LOFT facility. Surface drainage is good and studies have shown there are no important impermeable boundaries existing within the TAN area⁽³⁷⁾.

2. Geography

a. Topography

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 it is 34 miles long, and from east to west it is 29 miles wide⁽³⁸⁾ (see Fig. VI-4).

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⁽³⁹⁾ the mountains rise to an elevation of 10,000 feet. The elevation is lower along the Snake River to the east and southwest except for a gentle 600 ft rise midway between the site and the river. This rise runs parallel to the Snake River⁽³⁸⁾.

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⁽³⁹⁾. The mountains encircle the Snake River Plain and 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⁽³⁸⁾ (see Figs. VI-5 and VI-6).

b. Population Distribution

(1) On Site. The working force at each of the NRTS facilities as shown in Table VI-b is variable, depending on the construction work in progress. The closest facility to TAN is NRF which is 22 miles south-southwest and has a daytime population of 1400 with a nighttime force of 200. The projected daytime population of TAN is approximately 300 with 50 nighttime personnel.

TABLE VI-b
POPULATION DISTRIBUTION OF NRTS⁽³⁹⁾

<u>Location</u>	<u>Day</u>	<u>Night</u>	<u>Distance from TAN in Miles</u>	<u>Direction from TAN</u>
TAN Area	300	50		
NRF	1400	200	22	SSW
MTR-ETR	700	150	25	SSW
CPP	300	20	25	SSW
EBR-I and BORAX V	100	2	30	SSW
Central Facilities	800	100	27	SSW
OMRE-EOCR	250	10	27	SSW
SPERT	140	5	26	SSW
Army Reactor Area	100	5	27	SSW
TREAT and EBR-II	200	5	22	SSW
Total	4290	547		

(2) Off Site. The towns with unlisted populations in Table VI-c are unincorporated and no census figures are available. For location of towns see Fig. VI-7.

3. Meteorology

The meteorological conditions at NRTS are described in detail in previous reports^(41,42). These reports are used as a basis for the following data:

a. Temperature⁽⁴³⁾

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

TABLE VI-c

OFF-SITE POPULATION DISTRIBUTION⁽⁴⁰⁾

<u>Location</u>	<u>Distance from TAN in Miles</u>	<u>Population</u>	<u>Direction from TAN</u>
Mud Lake	10-15	187	East
Monteview	10-15	*	NE
Berenice	10-15		West
Terreton	10-15	*	East
Howe	15-20	25	WSW
Hamer	20-30	144	East
Winsper	20-30		NNE
Roberts	20-30	422	ESE
Small	20-30	10	NNE
Dubois	30-40	447	NE
Menan	30-40	496	ESE
Lewisville	30-40	385	ESE
Idaho Falls	30-40	33,161	SE
Arco	30-40	1,562	WSW
Moore	30-40	358	WSW
Darlington	30-40	10	West
Atomic City	30-40	141	South
Humphrey	40-50	25	NNE
Spencer	40-50	100	NE
St. Anthony	40-50	2,700	ENE
Parker	40-50	284	ENE
Plano	40-50		East
Salem	40-50		East
Sugar City	40-50	584	East
Rexburg	40-50	4,767	East
Lorenzo	40-50	100	ESE
Sunny Dell	40-50		ESE
Rigby	40-50	2,281	ESE

TABLE VI-c - Continued

<u>Location</u>	<u>Distance from TAN in Miles</u>	<u>Population</u>	<u>Direction from TAN</u>
Ucon	40-50	532	ESE
Iona	40-50	702	SE
Ammon	40-50	1,882	SE
Shelley	40-50	612	SE
Basalt	40-50	275	SSE
Firth	40-50	322	SSE
Goshen	40-50		SSE
Blackfoot	40-50	7,378	SSE
Moreland	40-50	320	SSE
Riverside	40-50		SSE
Rockford	40-50		South
Martin	40-50		WSW
Grouse	40-50	58	West
Mackay	40-50	560	ESE

* The rural population in the area surrounding Mud Lake, Montevieu, and Terreton is approximately 1,000.

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 1.73 in. in June, 1954, and that of snow, 8.5 in. in January, 1957. For the TAN area in particular, the annual amount of moisture is 7.35 in., with a maximum rainfall per 24 hours of 1.33 in. recorded in July, 1953.

c. Adiabatic, Lapse, and Inversion Conditions

Normal weather conditions at NRTS develop lapse conditions during daylight hours with inversion conditions readily 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, which is greater than the undisturbed or night surface speed. Should the surface winds maintain a speed greater than 15 miles per hour through the night, they will frequently prevent the formation of an inversion. The season with the least number of inversions is spring with inversions occurring on 92% of the days. During the winter season, inversions are of a longer duration than those during the summer months due to the longer nights in the winter. From Fig. 30 of IDO-12015⁽⁴²⁾, inversions may be expected 92% to 98% of the nights of the year, and an inversion of at least 10-hours duration may be expected on more than 61% of the nights of the year.

Measurements of the thermal gradient at the NRTS have been underway since 1950, and the data⁽⁴²⁾ lead to the following conclusions:

- (1) When long durations of inversions are considered (> 15 hours), Fig. 30⁽⁴²⁾ shows that during the summer months the duration of such an inversion has only a 0.2% probability of occurring.
- (2) During the spring, fall, and winter seasons the probability of an inversion of long duration increases considerably. Lapse conditions where $n = .25$ or lower exists over 50% of the time for these months at the NRTS.

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. 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 greater than 60 consecutive hours.

The winds at the NRTS show a seasonal variation with the principal contrast being between winter and summer as shown in Figs. VI-8, 9, 10, 11, 12, and 13. 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. A correlation between precipitation and wind direction indicated that a precipitation wind rose did not vary significantly from a surface wind rose.

For the purpose of establishing the upper limit of environmental hazard resulting from an activity release at LOFT, strong inversion conditions with Sutton's stability constant $n = 0.5$ are used. Lapse conditions, represented by the Sutton constant $n = 0.25$ are used as the most probable average condition existing at the site.

4. Geology

The NRTS is at the central northern edge of the semi-arid Snake River Plain in southern Idaho, adjacent to the southern foothills of the Lemhi and Lost River ranges. The 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⁽³⁸⁾.

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⁽⁴⁴⁾.

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⁽⁴⁴⁾.

The altitude of the water table ranges from 4580 ft above sea level in the northern part of the station to about 4400 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⁽³⁷⁾.

5. Seismology

The NRTS site is located in a region which "The Pacific Coast Uniform Building Code", (1949) designated as a zone 2 area⁽⁴⁵⁾. 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 epicenters 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 completely safe⁽³⁸⁾.

Although the lava plain of the Snake River is geologically young, the surrounding mountains are mostly of great age⁽³⁸⁾. 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.

Fig. VI-1 - Plot Plan - Test Area North

Fig. VI-2 - LOFT Proposed Plot Plan

Fig. VI-3 - LOFT Pictorial

Fig. VI-4 - National Reactor Testing Station Layout

Fig. VI-5 - NRTS and Southeastern Idaho

Fig. VI-6 - Topographical Cross Section Through NRTS

Fig. VI-7 - National Reactor Testing Station and Vicinity

WINTER

SPRING

SUMMER

FALL

TOTAL

Fig. VI-8 - TAN 20 Foot Level Lapse Wind Roses
November 1952 Through May 1962

Fig. VI-9 - TAN 20 Foot Level Inversion Wind Roses
November 1952 Through May 1962

WINTER

SPRING

SUMMER

FALL

TOTAL

Fig. VI-10 - TAN 20 Foot Level Wind Roses
November 1952 Through May 1962

Fig. VI-11 - TAN 150 Foot Level Lapse Wind Roses
May 1956 Through May 1962

Fig. VI-12 - TAN 150 Foot Level Inversion Wind Roses
May 1956 Through May 1962

Fig. VI-13 - TAN 150 Foot Level Wind Roses
May 1956 Through May 1962

6. Technical Support Facilities

The Test Area North (TAN) of the National Reactor Testing Station (NRTS) has been selected by the Commission as the area in which to conduct the Safety Test Engineering Program (STEP). Phillips Petroleum Company, the Commission's designated nuclear safety contractor, has already been assigned the Initial Engineering Test (IET) facility, a part of the Low Power Test (LPT) facility, and approximately 20,000 sq ft of technical and administrative office space in the Administration Building 602. The Technical Support Facilities (TSF), which includes the hot shop, assembly areas, hot cells, metallurgical and analytical laboratories, radioactive component storage areas, and other miscellaneous service buildings, has also been assigned to Phillips Petroleum Company. However, these service facilities are administered through the operating branch for the purpose of providing services to all TAN contractors of which the STEP Program, administered through the Phillips technical branch, is a part. Other Commission activities located at TAN include the Fast Reactor and Maritime Programs (General Electric Company), located in part of the LPT, the Experimental Beryllium Oxide Reactor (EBOR - General Dynamics), located in the former Shield Pool Test Facility (SPTF) and the Lithium Cooled Reactor Experiment (LCRE - Pratt and Whitney), located in the former Flight Engine Test (FET) facility. Administrative offices for all the contractors are located in Building 602.

The TAN area is ideally suited for furthering the study of consequences of major reactor accidents. The integrated field-scale testing of nuclear reactors subject to destructive excursions requires the use of large hot shop facilities to perform post-test disassembly and detailed analytical examination of individual components. Mobility of the STEP large reactor experiments also appeared prudent to allow fabrication of the experiment in an area other than the test facility, and to allow the time consuming post-test examination to be performed in an area more suited for analytical work. Such a mobile experiment philosophy is conducive to more effective utilization of the expensive containment type remote operated test facilities such as required in the loss of coolant test program.

The TAN area, originally built for the now-defunct ANP program, was constructed under the same mobile reactor philosophy in order to study the reactor components following extended operation of direct cycle nuclear engines. Consequently, remotely located test facilities were built approximately 1-1/2 miles from the examination area connected by heavy duty 4-rail trackage on which large, relatively unshielded reactor assemblies could be moved with ease. A special manned shielded locomotive was designed and built to move the highly radioactive nuclear packages to the examination area.

a. Examination Area

TAN buildings 607, 657, and 633 are designated as the examination area and comprise an integrated facility for performance of all phases of post-test analytical examination of nuclear systems and subsystems following sustained power operation. The flexibility of the facility and its contained equipment was clearly demonstrated when the SL-1 reactor vessel was remotely disassembled and studied to understand the initiating nuclear excursion mechanism.

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 and control laboratory, inspection and control area, and miscellaneous auxiliary facilities (locker room, offices, shower area, etc.).

The major facilities within the examination area are discussed briefly herein to acquaint the Architect Engineer with the capabilities and services provided.

(1) Hot Shop. The hot shop is an area 160 ft long by 51 ft wide by 67 ft-6 in. high surrounded by reinforced concrete shielding walls to provide a minimum dose rate of 6.25 mrem/hr in the operating galleries located on both sides of the long areas of the hot shop while containing reactor packages emitting 5.7×10^6 R/hr gamma

sources. Oil filled shielding windows located in both the upper and lower operating galleries 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, bi-parting concrete door with a staggered joint. Movement of this door is accomplished 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 essentially 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 of 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. It is used to lift, hold, and transport heavy assemblies, tools, and fixtures, and to retrieve other types of handling equipment should they become inoperative.

The control stations for the heavy crane are located at the hot shop windows and are interlocked so that only one station may have control of the crane at a given time and control cannot be "stolen" from the operating station by a standby station. Each control station contains all controls 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 has been engineered to service a variety of "hot" mechanisms with the greatest possible versatility and to handle future designs with a minimum of modification.

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

The remotely controlled overhead manipulator is a heavy-duty, rectilinear type machine designed to use various types of tools. Mounted on the manipulator trolley are telescoping tubes which may be rotated through an angle of 350° in either direction from their neutral position. The telescoping tubes may be extended or retracted a distance of 25 ft at a maximum speed of 25 fpm. Attached to the telescoping tube is a clevis, arm, wrist, and general purpose hand. With this apparatus it is possible to perform various remote manipulations. A load of 3000 lbs can be lifted with the arm and hand in a straight-down position. A detachable hook on the clevis can lift loads up to 5000 lbs.

The overhead manipulator is controlled from one of the four control consoles located in the control galleries. The consoles are interlocked so that only one has control of the manipulator at a time, and control cannot be "stolen" from one console by another.

The wall manipulators consist essentially of General Mills, Model C rectilinear manipulators mounted on a track and boom system. The track and boom system is designed to operate on the two long walls of the hot shop (two manipulators for each wall) at a maximum distance of 24 ft from the wall. Each manipulator operates on its own boom system, but each pair operating on one wall has in common the horizontal track system for that wall. The bottom of the horizontal boom on which the manipulator operates will clear an elevation of 30 ft from the floor in an "up" position, and in the "down" position will permit the operation

of the manipulator hand at the floor. The boom will swing in a horizontal plane about a pivot to the wall line.

Each wall manipulator and its supporting boom can be controlled from a control console located in any viewing window on the same side of the hot shop.

A pair of Argonne master-slave type manipulators is also installed in the hot shop. These manipulators are mounted in 10 in. tubes passing through the hot shop wall above one of the hot shop windows. These manipulators are designed for a 6 ft-10 in. wall thickness, for work loads in the order of 4 lbs per hand, and for the radioactivity levels encountered in the hot shop. They are provided with positive locks so they may be locked in position as desired by the operator. These manipulators are adaptable for small light work where some sense of feel is necessary. Either the wall manipulators or the heavy overhead manipulator may be used to deliver work to the master-slave manipulators.

The 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.

(2) 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 services 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 shielded from the hot shop. The overhead manipulator and overhead crane may be brought to the services cubicle, decontaminated, and maintained by contact method. A special fixture has been provided so that the General Mills wall manipulators may be remotely removed from the track and boom system and transported by either the overhead manipulator or the overhead crane to the services cubicle.

(3) 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. Objects under this periscope may be viewed with magnifications of 1X, 3X, and 6X powers and may be photographed by use of a camera attached to the periscope.

This area is serviced by four Model 8 master-slave manipulators, two Argonne #6 manipulators, and two General Mills type manipulators. Each master-slave manipulator services only a small area directly in front of its mounting window location, but the bridge mounted General Mills manipulators service the entire cubicle. A 3-ft extension hand has been provided to extend the work volume and usefulness of the General Mills manipulator.

Contact maintenance has been assumed possible in the case of the RML equipment. The design of the inspection cubicle and its equipment is based on the assumption that radioactive materials may be removed from the room, the room decontaminated, and contact maintenance performed. Acid resistant materials have been used in the construction of this equipment to permit decontamination procedures.

The following services are furnished to the RML: acid, high pressure air, vacuum sampling line, raw water, mask air, demineralized water, and instrument air.

(4) 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 area. The power to the crane is on the emergency power circuit and can therefore be controlled from the dispatcher's console. Normally it is controlled from a push-button pendant station on the trolley.

(5) Hot Cells. The hot cell facility, consisting of four hot cells and miscellaneous work areas, provides space for post-irradiation examination of reactor fuel and mechanical components. The cell is divided into four equal areas, the type and degree of contamination that can be handled in each being slightly different.

High density concrete was used in the walls with normal density concrete used for the ceilings. The high density concrete walls between the cells reduce the maximum radiation contribution from one cell to another to 100 mr/hr.

The change room is located along the only passageway from the contaminated area to the clean area with hand-wash fountains and pass-through showers available for personnel decontamination.

b. Radioactive Storage

Within the TSF area is located a radioactive storage yard known as the Radioactive Parts Storage and Security Area (RPSSA). The RPSSA is located 3/8 mile west of the examination area and is used to store large nuclear systems and subsystems that have been previously used in the various experimental programs. The area is served by the 4-rail track system and includes two large high bay Butler type buildings capable of receiving the dolly mounted systems. Presently the HTRE-2 and -3 reactor assemblies are housed in these structures.

c. Hot Waste Storage

Immediately north of the examination area and west of the hot cell extension is located the liquid waste storage area and the evaporator building, TAN-616. Liquid wastes from all the test facilities are received by truck and stored in two 20,000 gallon underground stainless steel storage tanks. Due to the close proximity of the examination area, the radioactive liquid wastes from the hot shop, hot cells and laboratories are pumped directly to the storage tanks. The tank contents are circulated through the evaporators to concentrate approximately 200 gallons/hour to a volume of 12 to 15% of initial volume. The concentrate can be permanently stored in either of two 50,000 gallon storage tanks located in the storage tank area or

shipped by a shielded tank truck to the Chemical Processing Plant (CPP) for ultimate storage. It is intended that highly radioactive liquid wastes from the LOFT facility will be processed in a similar fashion using the existing TSF evaporator facility.

d. Service Facilities

The self-sustaining TSF area accommodates other service activities which are used in support of the examination area and the remote test facilities. These include a warehouse and receiving building, TAN 628, a health physics building, TAN 606, a medical dispensary building, TAN 618, fire station building, TAN 603, and an equipment repair shop building, TAN 604. Those services which are specifically in support of the TSF buildings involve the conventional utility systems such as steam, fuel, water, power, waste, etc.

7. Radiological Factors

The magnitude of the radiological hazards to the general population, as well as to on-site personnel, will be dependent on the quantity and type of fission products released to the atmosphere. For the purpose of the radiological calculations it has been assumed that the following fission products are released to the containment shell:

- (1) 100 percent of the noble gases,
- (2) 50 percent of the halogens, and
- (3) 1 percent of the solids.

Of these quantities, 50 percent of the halogens (iodine) in the containment vessel is assumed to remain available for release to the atmosphere. The remaining 50 percent of the iodine is assumed to adsorb onto internal surfaces of the reactor building or adhere to internal components.

a. Radiological Hazards to On-Site Personnel

The radiological hazards to on-site personnel, as well as the general population, were based on a five-day exposure period from fission products produced from a reactor operating at 50 Mw for

400 hours. The gross fission product inventory of which 15 percent represents the total release to the containment shell, as well as the individual isotopes of 90 decay chains, was computed by the 7090 "Curie" computer program. The results from the "Curie" program were used as input data for the following radiological calculations.

(1) Direct Radiation From the Containment Shell. The direct radiation dose from the containment shell was calculated by the use of the following operation:

$$D = Q \alpha S B \frac{r_1^2}{r_2^2} e^{-\mu r} \int_0^t t^{-.21} dt$$

Q = Initial gamma source in curies, (4.1×10^7 curies)

α = Conversion factor, (1.6×10^{-4} rem/curies-sec) at 1 meter

S = Reduction in dose rate due to shielding by containment building, (.17)

B = Buildup factor, take as $[1 + \mu r + \frac{(\mu r)^2}{3}]$

μ = Air attenuation coefficient, ($7.9 \times 10^{-3} \text{ m}^{-1}$ for 1 Mev gamma)

r_1 = 1 meter

r_2 = Distance to receptor, (meters)

$\int_0^t t^{-.21} dt$ = Correction for radioactive decay (seconds)

t = Exposure time (4.3×10^5 seconds = 5 days)

As shown in Fig. VI-14 the radiation dose to personnel located in the LCRE area, which is the nearest facility to LOFT, is approximately 7 mrem.

(2) Direct Radiation From the Radioactive Plume. The direct gamma radiation dose from the radioactive plume was calculated by the following equation:

$$D = \frac{2Q}{\pi U c_y c_z (r)^{2-n}} (L_r)(C)(E) \int_0^t t^{-.21} dt$$

$$\begin{aligned}
Q &= \text{Initial gamma source in curies } (3.4 \times 10^7 \text{ curies}) \\
L_r &= \text{Leak rate } (.175\%/ \text{day}) \\
C &= \text{Conversion factor, } (.26 \frac{\text{Dis-m}^3\text{-rem}}{\text{sec-curie-mev}}) \\
E &= \text{Average energy of gamma source strength } (\frac{1 \text{ mev}}{\text{Dis}}) \\
U &= \text{Mean wind speed (m/sec)} \\
c_y \text{ \& } c_z &= \text{Sutton's diffusion parameters (m}^{n/2}) \\
n &= \text{Sutton's stability parameter}
\end{aligned}$$

Lapse and inversion atmospheric conditions were assumed for five days following the destructive test to calculate the direct radiation from the cloud at the specified containment leak rate (see Figs. VI-15 and VI-16). A more realistic curve based on a weighted mean of the atmospheric conditions present for the five-day period will be developed for the final hazards report summary on the LOFT facility by using computer codes. However, for this report if the worst possible atmospheric conditions are assumed to be present (inversion conditions, 2.5 mph wind, 0 ft height of release) for the five-day exposure period, the integrated dose at the LCRE facility is approximately 25 rem. The probability of an inversion condition lasting more than 15 hours is only .2%, which provides a direct radiation dose between 50 mrem and 25 rem as shown in Figs. VI-15 and VI-16.

(3) Inhalation Dose to the Thyroid. The inhalation dose to the thyroid was calculated by the use of the following equation

$$D_t = \frac{(2)(R)}{\pi c_y c_z U(r)^{2-n}}$$

$$\sum_{I-131}^{I-135} \frac{(8.54 \times 10^2)}{m} (f_a \bar{E} T_e Q_i) \frac{F_r L_r}{\lambda_r + L_r} [1 - e^{-(\lambda_r + L_r)T_1}] (1 - e^{-\frac{.693T_2}{T_e}})$$

- R = Breathing rate ($3.47 \times 10^{-4} \text{ m}^3/\text{sec}$)
 f_a = Fraction of inhaled material which reaches the thyroid
 $\bar{E}$ = Effective energy absorbed by the critical organ (mev/dis)
 T_e = Effective half life (sec)
 Q_i = Initial number of curies of isotope i present (curies)
 F_r = Fraction of isotope i available for release (.25)
 L_r = Leak rate ($2.06 \times 10^{-8} \text{ sec}^{-1}$)
 λ_r = Radiological decay constant for isotope i (sec^{-1})
 T_1 = Inhalation time ($4.03 \times 10^4 \text{ sec}$)
 T_2 = Time the dose rate is integrated over (1 year)

The curves presented in Figs. VI-17 and VI-18 are for inversion and lapse conditions which are assumed to be present throughout the five-day exposure period. As mentioned previously, more realistic curves will be presented in the final hazards analysis. However, as can be seen in the inhalation dose curves, the dose to the thyroid will be between .4 rem and 75 rem for personnel located in the LCRE area.

b. Radiological Hazards to the General Population

The radiological hazards considered, for the general population, were a direct radiation dose from the radioactive plume and an inhalation dose to the thyroid received as a consequence of inhaling material from the plume.

As shown in Figs. VI-16 and VI-18 the nearest site boundary is approximately 5-1/2 miles, and the radiation dose to individuals located in this area would be 10.5 rem to the thyroid and 3.8 rem from cloud shine. As mentioned previously, the dose is based on the worst possible atmospheric condition which would persist for the five-day exposure period. Therefore, a more realistic value would be between the dosages shown in Figs. VI-15 and VI-17 and the above mentioned values.

If Montevideo, the nearest population center to LOFT, is considered the radiation dose to the general population would be 1 rem direct cloud dose and 3 rem thyroid dose which are above the acceptable exposures presented in the Radiation Protection Guides as recommended by the Federal Radiation Council. Again these doses were based on inversion conditions and a 2.5 mph wind.

With the exception of the direct radiation dose from the containment shell the radiological calculations, for both on-site and general population, were based on a constant leak rate for the five day exposure period. Since the leak rate will decrease proportionally to a decrease in internal pressure, calculations were made to determine the exponential pressure decay as a function of time. If an assumed internal pressure of 24.2 psig exists in the containment shell at some specified time following the loss of coolant accident, the differential pressure after three days decreases to approximately zero psig by heat conduction and losses from the containment shell.

Calculations were also made to determine what length of time would be required to reduce the internal pressure (24.2 psig) to zero psig in the event it were necessary to terminate the experiment following the loss of coolant accident. Using the containment dome spray system which is capable of continuously delivering 150 gpm of water, the time to reduce the building pressure to 1 psig in four hours. Therefore, the radiation doses presented in Figs. VI-15 and VI-16 could be reduced by a factor of .114 while the doses shown in Figs. VI-17 and VI-18 could be reduced by a factor of .1851.

From the preliminary calculations, it can be concluded that no serious radiological hazard should exist to the general population and if proper safety precautions are taken, no serious hazard should be presented to on-site personnel.

Fig. VI- 14 - Direct Radiation from the Containment Shell

Fig. VI-15 - Direct Radiation from the Cloud - Lapse Condition

Fig. VI-16 - Direct Radiation from the Cloud - Inversion Condition

Fig. VI-17 - Inhalation Dose to the Thyroid - Lapse Condition

Fig. VI-18 - Inhalation Dose to the Thyroid - Inversion Condition

B. Facility Description

1. Containment Vessel

a. General

The LOFT test building is a cylindrical steel pressure vessel with a hemispherical top closure and a 2:1 semi-elipsoidal bottom closure. The vessel is designed to withstand and contain an internal pressure of 24 psig. The vessel is 80 ft in diameter and measures 80 ft tangent to tangent. Total height is approximately 140 ft, 65 ft of which is above the top of the earth embankment around the building. The shell material is high strength, high impact, Lukens LT-75 carbon-manganese-silicon firebox quality steel.

To prevent missiles from penetrating the steel containment vessel, the vertical interface of the vessel is lined with concrete 1 ft 0 in. thick and extends from the operating floor to the overhead crane. The containment vessel dome is also lined with 6 in. of shotcrete (see Figs. VI-21 and VI-22). All internal surfaces are sealed with a protective coating to allow decontamination of the building. The Architect-Engineer shall investigate the stress analysis of all construction materials to be used in the reactor vessel and containment shell for shock loading and thermal stress. The results of this investigation will determine the final selection of material to be used.

The 104 ft diameter spherical building shown in Figs. VI-25, VI-26, and VI-27 is an alternate test building configuration that should be optimized by the Architect-Engineer. Both building designs are in agreement with the safety test program, and selection will be determined by economics only.

The containment building volume and pressure design must accommodate loss of coolant experiments with many different primary system volumes. As an upper limit the containment design pressure will be sufficient to withstand the pressure associated with the coolant blowdown from an experimental system containing 2000 cu ft of water at 2500 psig and 668°F. However, for the experimental system described herein (approximately 1300 cu ft) the containment vessel volume will provide an internal

pressure of approximately 24 psig following the coolant blowdown to be representative of existing reactor containment structures.

In order to provide leakage to the atmosphere commensurate with postulated leakages from typical reactor containment vessels, the building design should include means for controlled venting to the atmosphere up to a maximum of 0.175% of the free volume per day under all pressure conditions. This maximum allowable leakage (0.175% of the building free volume per day) must of experimental necessity include the normal or uncontrolled leakage from the building.

Conventional external pressure changes plus the possibility of negative pressure due to atmospheric pressure changes or depletion of the contained oxygen supply due to fire are taken into account in the building design. To insure the integrity of the containment vessel a pressure test will be performed prior to each succeeding loss of coolant test. Since this will be a continued operational requirement the Architect-Engineer will provide the necessary equipment and instrumentation as a permanent part of the facility.

A large number of openings into the building are required for access of the railroad dolly mounted experiment, personnel, equipment, electrical conduits, etc. The largest opening is the 28 ft wide by 50 ft high railroad door opening. This opening is designed to withstand the deflection and bending moments produced by the maximum differential pressure. Buttresses are provided for the external door support and provide a bearing wall for the door. The railroad door is a steel framed door filled with concrete and designed for the maximum differential pressure. Air-locks are not required for the railroad doors and exterior personnel doors; however, a personnel air-lock will be required in the tunnel between the control building and the test building at the boundary of the containment shell. The doors and passageways are designed for simplicity, reliability, and to withstand the maximum predicted test pressure.

Ten fuel-element storage cells are provided in the floor of the test building and extending down into the concrete web between the access and equipment corridors. The stainless steel storage cells measure 12 in. inside diameter x 10 ft deep and are spaced a minimum of 24 in. between holes. Each storage cell drains into the hot waste drain system

to dispose of accumulated liquid wastes. Each cell is equipped with a closure plug 24 in. long with suitably designed offsets to preclude radiation shine. All closure plugs must be made airtight to prevent air leakage through the drain system.

All service connections are sealed for consistency with the pressure and leakage requirements of the building. The air inlet and exhaust connections meet this requirement by quick closing valves which operate upon an increase in building pressure above atmospheric or an increase in radioactivity of the exhaust air.

The 4-rail dolly trackage enters the containment vessel at grade level, 25 ft 6 in. above the lowest point of the semi-elipsoidal bottom closure. The tracks terminate in the building 10 ft from the wall opposite the railroad door.

Lighting within the containment building is maintained at 50 ft-candles at floor level by overhead high bay incandescent luminaries. Supplemental high-intensity fixed focus lighting fixtures are provided overhead and in the vertical building walls to provide 2000 ft candle power around the reactor and at the loss of coolant blowdown nozzle to facilitate high speed photography.

Space is provided in the containment building above either coupling room for temporary storage of the reactor vessel head.

b. Coupling Stations

Two coupling stations are provided on the north-south centerline of the test building, one on each side of the railroad trackage. The face of the coupling station parallel to the railroad trackage is 11 ft (minimum) from the building centerline (Figs. VI-22 and VI-23).

The face of the coupling station has a 7-1/2 in. wide slot designed to accommodate the coupling plug attached to the railroad dolly. Parallel to this wall, 18 in. back, is a second partition suspended from the coupling station ceiling. This partition forms a labyrinth which will reduce direct radiation inside the coupling room.

The coupling room is 10 ft wide by 8 ft high by approximately 22 ft deep. Access to this room can be gained by a shielding door off the

operating floor or a circular stairway from the basement. The side walls and roof are 5 ft 6 in. thick, high density concrete, with lead partitions and doors of equivalent shielding thickness.

A motor-driven lead door is provided to close the slot allowing personnel access to the coupling plug suspended from the railroad dolly. Controls for these doors are located on the face of the coupling station, in the remote control room, and in the control building. Position indicators are located in the remote control room and on the control console in the control building.

Utility and electrical requirements for the reactor package are supplied from one coupling station while the other coupling station is designed to handle the instrumentation and low level signals from the reactor package. All of the utility, electrical, and instrument leads will be equipped with connectors located in the coupling stations, and jumpers will be used between the facility junction boxes and the reactor package coupling plug. The utility lines are connected to the coupling plug by using spool pieces with quick disconnect couplings. The large supply and return tertiary coolant pipe lines exit the building floor and are mated to the reactor package lines by remote operated couplers.

The coupling station will contain a remote controlled decontamination system so that some decontamination can be performed prior to admittance to the area.

c. Overhead Crane

A remotely controlled 30/5 ton polar crane with a 70 ft clear hook height is installed within the test building. Controls for the crane are located in the remote control room (opposite the railroad door). An intercom system with phone jacks strategically located on the operating floor connects the remote control room to obtain audio contact with the crane operator when access to the building is permitted.

d. Earth Embankment

Earth embankments are provided around the test building to reduce the dose rate to 7.5 mr/hr during normal reactor operating

periods. Although the reactor contains a primary water shield, the N^{16} activity in the primary coolant system constitutes a primary radiation hazard. The earth shielding will allow limited access to the grade level entrance of the control and utility building and to the outside entrance of the remote control and sample rooms. For cost purposes, the earth embankment has a 12 ft wide flat top, 39 ft 6 in. above grade, with a 3:1 slope and surrounds the entire test building except at the large railroad dolly access door.

e. Remote Control and Sample Rooms

The remote control and sample rooms are erected opposite the railroad door on the earth embankment surrounding the containment vessel. The two level structure has the sample room, hall, and personnel elevator entrance at the lower level with the remote control room and switchgear area on the second level. Controls for the crane, coupling station doors, railroad door, and manipulator are located in the remote control room.

The concrete block structure measures 28 ft square by 23 ft high. A 5 ft square personnel elevator connects the first floor to the tunnel between the test building and control building. The first floor level is 51 ft above the tunnel entrance door.

The sample room is airtight and measures 28 ft long by 17 ft wide (outside dimensions) and has a 9 ft ceiling. The remote control room is above the sample room and covers the entire level except for the enclosure around the elevator shaft and the stairwell.

The remote control and sample rooms are designed to allow access during periods of high radiation inside the test building. A shielding wall between this building and the test building provides a maximum weekly dose of 100 mrem (assuming 8 hrs/day, 5 days/wk) following a destructive test with 15% of the fission products released to the containment dome. For the purpose of cost estimation, this wall is 5 ft 6 in. thick high-density concrete containing a shielding window designed to provide an overall view of the test building interior. This window measures 5 ft wide by 3 ft high on the inside. The window design is consistent with the leakage requirements for the test building

A small 25 lb wall-mounted manipulator is located inside the test building above the oil filled shielding window. The manipulator will be used to transfer solid samples from the sampling conveyors to the sample transfer tube and vice versa. The tube designed to transfer sample materials between two pressure environments penetrates the containment vessel and adjacent shielding into the sample room.

f. Containment Sampling Systems

The LOFT test building is equipped with sampling systems for solid fission products and halogens released to the interior of the containment building (see Fig. VI-28). Four gas sampling trains for isotopic gas analysis, six high and low volume recirculating air sample systems for gaseous particulate analysis and a solid fission product and integrated dose rate sample recovery system are provided to systematically obtain quantitative information on the release of gaseous and solid fission products as a function of time. In addition, similar sampling systems are included in the building exhaust system for radiological control during emergency and controlled release of the containment volume to the atmosphere.

The four gas sampling trains consist of charcoal iodine traps for particulate and molecular iodine removal, and coagulation delay reservoirs with millipore filters and/or charcoal with CO_2 or N_2 cooled low temperature elements for collection of xenon or krypton samples. A Research and Development program presently being conducted by ORNL in connection with the Nuclear Safety Pilot Plant may introduce new techniques in sampling noble gases and gaseous fission products from contained volumes. It is suggested that the A-E follow this program in order that the latest sampling devices and procedures be incorporated into the LOFT facility design.

High and low air samples are taken at six points in the interior of the building and conveyed by means of vacuum pumps to a filter manifold located in the sample room. The filtered air is then exhausted back into the test building. The filters are an integral part of filter holders which have been equipped with quick disconnect type couplings, in order that the holders may be removed from the manifold and taken to the TAN hot shop for analysis on the mass spectrometer.

Solid fission products, metal foils, plateout samples, etc., are positioned in a pattern within the containment building by being attached to cable-type mechanical conveyors. These zone conveyors are installed at the wall and floor surfaces to allow placement of sample holders at representative locations on the containment surfaces. Recovering of samples as a function of time requires the motorized cable-type conveyors to converge near the remote controlled manipulator to facilitate transfer of the sample holders into the sample room via the installed transfer tube. Six equi-distance vertical wall conveyors rise to the level of the wall manipulator and traverse horizontally to the manipulator location. The continuous fan-shaped conveyor at the main floor level rises along the wall immediately below the sample room shielding window to a level accessible to the wall manipulator.

Once the radioactive samples are transferred into the sampling room, they are loaded into shielding pigs and moved to the hot shop for radiochemical analysis. The shielding pigs will be supplied by PPCo; however, the Architect-Engineer will provide means of transporting the pigs in and out of the sampling room. New sample plates are then transferred from the sample room through the transfer tube into the containment building and mounted in the previously occupied sample holder. The conveyor will then be moved back to its original position to obtain additional fission product deposition as a function of time.

The building exhaust system serves a twofold purpose: (1) to recirculate building gases through the necessary particulate and molecular iodine filters to reduce the level of gaseous and particulate activity within the building during emergency conditions, and (2) to provide some degree of filtration during evacuation of the building to the atmosphere. The system consists of a 35,000 cfm exhaust fan located downstream of a series of roughing filters, absolute filters, and charcoal filters for Iodine-131 removal connected by ducting to the containment building or to the stack. Included are gas sampling trains similar to those installed on the main containment shell, radiation instrumentation, differential pressure instrumentation, and flow control devices.

g. Decontamination Equipment

Permanent decontamination facilities are provided in the LOFT test building for decontaminating the interior wall, ceiling, and floor of the test building (see Fig. VI-28).

The decontamination system includes a 12 ft diameter header with spray nozzles mounted on the under side of the test building dome. The overhead decontamination sprays will also be used for studying pressure reduction effects and fission product deposition when spraying water into the containment shell.

A second header is located below the overhead crane to cover the walls and area near the wall at floor level. This header is approximately 78 ft in diameter with spray nozzles at appropriate intervals. Removable nozzles are also located on risers around the railroad trackage at floor level. These risers are normally capped to reduce obstructions on the floor, and the nozzles will be inserted to the floor decontamination header just before a test. In addition to these systems the Architect-Engineer will make provisions for decontaminating areas below the operating floor that will require personnel admittance soon after the loss of coolant test.

The decontamination solution makeup system consists of an insulated 15,000 gallon stainless steel mix tank with steam heating coils, motorized propeller mixers and a solution supply pump (150 gpm, 200 ft-t_{dh}, 7-1/2 hp). After the mix tank is filled with demineralized water, and heated if necessary, the chemicals are manually introduced to the mix tank. Chemical solutions ranging from a pH of 1.5 to 13 are then pumped to the building distribution systems.

2. Control and Utility Building (See Fig. VI-29)

The half-buried control and utility building is a 98 ft by 117 ft reinforced concrete structure covered by shielding to provide a maximum dose rate of 2.5 mrem/hr following a destructive test with 15% of fission products released to the containment dome. It has a shielded entrance at grade and the shielded roadway entrance at floor level.

The control building houses the control and instrumentation room, experimental laboratory, photographic laboratory, instrument shop, eating area, mechanical work area, offices, counting room, and utility area with an attached turnaround area.

The control and data rooms serve as the operations center for the LOFT facility during all test phases including the destructive tests. Control consoles and process and experimental instrumentation required for recording and reduction of the test data are located in these rooms. All door interlocks, gate control, test warning sirens, and visual and audible warning systems are handled from the control room.

The counting room is formed of 2 ft thick high-density concrete walls and has an entrance labyrinth formed by an inside wall.

Space for one vehicle, for loading of passengers, and space for one standby vehicle is provided in the turnaround area connecting with the shielded roadway.

Filtered air intake and exhaust structures terminate above the earth roof cover. The main filters on the intake are of the high-efficiency type preceded by electrostatic and coarse filter banks.

The control and utility building construction is primarily reinforced concrete on prepared subgrade. The roof contains prestressed concrete beams covered with a concrete slab 18 in thick for shielding purposes. Interior concrete walls are filled and painted to provide a smooth attractive wall surface. The floors contain asphalt tile top set with rubber cove except in the turnaround area, utility and mechanical work areas. Interior walls and partitions are of stud construction covered by gypsum wallboard. Glass windows are used extensively in those areas such as the control room where observers and visitors are restricted from areas of test control. The control building is also designed to serve as a fallout shelter for 20 people for a period of two weeks.

Connecting the control building, the test building and the process building are a series of tunnels which contain cable trays for instrumentation and electrical leads and wall racks for routing the utility and process system piping. Space is also provided for personnel access

to each building. The tunnels are covered by earth shielding to provide a maximum dose rate of 7.5 mrem/hr following a destructive test with 15% of fission products released to the containment dome.

The utility area is a room of not less than 3000 sq ft, 15 ft high, housing the steam boilers, demineralizers, softeners, compressed air systems, emergency power systems, and auxiliary support systems for the nuclear experiment. All services to the test building are to be routed via the connecting tunnel.

3. Process Water Building

The process building is an 80 ft by 40 ft pumice block structure which houses the heat exchangers and equipment for transferring of 50 Mw reactor heat from the secondary to tertiary cooling loops. Three sides of this process building are enclosed in the shielding berm of earth that separates the test building from the process building. The building will be shielded sufficiently to reduce the maximum postulated dose rate to 7.5 mr/hr. A 35 ft tunnel is provided as a pipe gallery from the test building to the process building for the secondary and auxiliary cooling water piping. The process building is connected to the cooling towers by an underground canal for the tertiary water. In addition to the heat exchanger equipment for the secondary and tertiary loops, this building will house the cooling tower turbine pumps, tertiary water treatment system, and all electrical equipment necessary for the heat dissipation systems.

Truck doors are provided on the south wall as well as personnel doors and windows. The roof is of prestressed concrete and the floor shall be designed for 2000 lbs/sq ft loading. Structural steel is designed in accordance with the latest addition of AISC specifications for design, fabrication, and erection of structural steel for buildings. Excavation for column footings have a minimum depth of 5 ft below grade and main load bearing sections are carried on undisturbed soil or on tamped fill sufficient to provide adequate bearing.

Heating is provided by fin tube unit heaters and ventilation is provided by roof type exhaust fans. Fire protection is provided by a wet pipe sprinkler system installed in accordance with the NFPA

Pamphlet No. 13. Adequate lighting is provided for all working areas with fluorescent lighting in ceiling mounted fixtures. Grounding and lightning protection is used throughout the building and all structural steel, equipment, piping, conduits, etc., are grounded and connected to the substation grounding system. Equipment frames and receptacles for general 120 volt single phase supplies are provided at approximate 12 ft intervals around the interior of the building.

4. Central Utilities Area (CUA)

Utilities and power are supplied to LOFT from the Central Utilities Area (CUA) located 5/8 mile west of the LOFT site. The location of CUA permits normal access to the area by maintenance and operation personnel except for a short time following a fission product release at LOFT. The CUA location will also facilitate future test areas with minimum expansion to the installed utilities.

The CUA consists of a utility building, an electrical substation, two water wells with pump houses, and a 500,000 gallon above ground raw water storage tank. The utility building is a 20 ft by 30 ft one-story pumice block structure containing the service water pump, the fire water pumps, the chlorinator, and electrical switchgear. The area is serviced by an access road from the LOFT facility and an emergency road connecting Highway 22.

Fig. VI-19 - LOFT Plot Plan

PLOT PLAN

Fig. VI-20 - LOFT Plot Plan and Elevation View

Fig. VI-21 - LOFT Test Building - Section A-A

Fig. VI-22 - LOFT Test Building - Section B-B

Fig. VI-23 - LOFT Test Building - Section C-C, Floor Plan Above Grade

Fig. VI-24 - LOFT Test Building - Section D-D, Floor Plan Below Grade

Fig. VI-25 - Alternate LOFT Test Building - Section A-A

Fig. VI-26 - Alternate LOFT Test Building - Section B-B

Fig. VI-27 - Alternate LOFT Test Building - Floor Plan

Fig. VI-28 - Containment Building - Sampling, Exhaust, and Decontamination Schematic

Fig. VI-29 - LOFT Control Building - Floor Plan

5. Transportation

a. Trackage

New dolly trackage is provided from the turn into the existing LCRE site to the LOFT site. This trackage is of the same design as the existing 4-rail trackage in the TAN complex. All fill and subgrade is compacted to 95% maximum density at optimum moisture content, and the trackwork is designed according to the American Railway Engineering Association's standards. The minimum radius of curvature of this trackage is 1000 ft, and the maximum grade is 1%.

b. Turntable

Another existing component of the railroad system that is used in the support of the LOFT facility is the 90 ft diameter turntable located just west of the main TAN hot shop. This turntable is equipped with television camera, indicators, and controls for remote operation by a dispatcher located 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 built-up plate main girder spans 42 ft from the spindle to the carriage rail.

Calculations indicate the turntable is not capable of supporting the experiment design load of 580 tons and the locomotive load of 215 tons concurrently.

Perusal of turntable complexities indicated the most feasible method of reinforcing the turntable to handle the loads involved in the forthcoming STEP tests is to construct an additional 42 ft diameter circumferential track complete with carriage assemblies (see Fig. VI-30). The carriage assemblies are duplications of the existing carriages located under the main girders. The new track requires an independent foundation to be constructed in a manner that presents the least disturbance to the existing foundation and soil conditions. To accomplish this task a piertype foundation is suggested to conform to the existing turntable foundation.

Preliminary calculations indicate 32 piers of 80 ton capacity will be required to support new circumferential track. The piers will function as a unit and achieve lateral stability through the existing radial beam incorporated in the new construction. The concrete section would provide the dual functions of acting as a pier cap and supporting the 42 ft carriage track.

The existing concrete beams radiating from the center spindle to the ringwall footing will be broken out at the location of the proposed track. The reinforcing steel of the radial beams will be left intact. When the concrete of the 42 ft beam is poured, the exposed rebar will be covered and the entire turntable will again function as a unit. The carriage rail will be seated on the new circumferential concrete beam in the same manner as the existing rail.

An alternate possible solution was considered which would eliminate the need to reinforce the turntable. The idea was to split the reactor package on two or more dollies similar to the one presently in use. The dollies would be moved across the turntable separately with the loads kept below the present capacities of the turntable.

However, the split dolly method of utilizing the turntable has the following program disadvantages: (1) increased number of experiment flange connections which would be operated remotely under conditions of high temperatures, high pressure, and high radiation levels; (2) programs require the reactor be centered in the test structure (this may not always be possible with the multi-dolly arrangement); (3) the cost of obtaining the additional dollies; and (4) the additional rolling stock maneuvering problems presented at the hot shops, turntable, and test structure. Because of the many problems of transporting highly radioactive dolly mounted experiments, this alternative was discarded in favor of reinforcing the turntable as suggested above.

It should be noted that the intent of the preceding recommendations is not intended to restrict ideas and recommendations of the architect engineer but is intended to convey the type and general layout desired to reinforce the existing turntable.

c. Rolling Stock

Movement of test equipment in and around the test site is accomplished by means of a special shielded locomotive over a 4-track standard gauge railroad system. This shielded locomotive, located at the TAN complex, functions as a personnel carrier and prime mover of equipment about the test site (see Fig. VI-31). The locomotive specifications are as follows:

Weight:

Total locomotive	430,000 lbs
On drivers	224,800 lbs
Per driving axle, front truck	57,400 lbs
Per driving axle, rear truck	55,000 lbs
Per idle axle, front truck	114,700 lbs
Per idle axle, rear truck	90,500 lbs

Ratings:

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

Shielding:

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 couplings)	44 ft 7 in.
Height	19 ft 4 in.
Width	11 ft 10 in.

One 4-rail (3-standard gauges) railroad dolly, 47 ft long by 20 ft wide, is required to support the reactor vessel with the primary systems. Equipment to be located on the dolly includes: (1) the reactor vessel with the reactor supporting structure, (2) one primary heat exchanger, (3) two primary circulating pumps, (4) one pressurizer tank, (5) an ion exchange system booster pump, (6) two regenerative heat exchangers, (7) three ion exchange columns, and (8) an emergency primary coolant shut down pump. See Fig. VI-32 for the reactor equipment layout.

It is proposed to locate the reactor vessel in the center of the 47 ft dolly with the auxiliary coolant systems in front and behind. The reactor vessel will rest on vertical supports between the bottom head of the reactor vessel and the dolly bed. These supports are designed to restrain any lateral movement of the reactor vessel. The reactor support structure allows for vertical growth of the pressure vessel due to thermal expansion.

The two primary pumps and the primary heat exchanger are located on one end of the railroad dolly. The pressurizer tank and primary purification system are mounted on the other end of the dolly.

Connections between the facility equipment and the equipment mounted on the railroad dolly include: (1) a demineralized water flush line, (2) two reactor drain lines, (3) one demineralized water make-up line, (4) one ion exchange column drain lines, (5) a demineralized water supply line for the ion exchange columns, (6) two 3-inch lines for the supply and return water for one of the secondary ion heat exchangers, and (7) supply and return lines from the Dowtherm secondary pump to the primary heat exchanger.

All electrical and utility lines required to operate the equipment mounted on the railroad dolly, as outlined in the previous paragraphs, will be brought to a utility coupling plug on the south side of the dolly. This coupling plug is designed to insert into the shielded coupling station for connection of the utility and electrical power supplies.

On the north side of the railroad dolly is a second coupling plug. This coupling plug is essentially identical to the south coupling plug, except it is used for routing signal and instrument leads from the reactor to the control area. By separating the utility and power supplies from the instrument signals, magnetic fields and stray currents are minimized in the experimental oscillographs and tape recording systems.

Total weight of the shielded locomotive and reactor package is 795 tons. The unit breakdown is as follows:

Shielded locomotive (with water)	215 tons
47-foot dolly	100 tons (approximate)
Reactor and equipment (with water)	480 tons (approximate)

b) HALF SECTION THRU PIT FOUNDATION

c) CARRIAGE RAIL CONNECTION

a) FOUNDATION PLAN SHOWING NEW CONSTRUCTION

Fig. VI-30 - TAN Turntable Modification

Fig. VI-31 - View of Shielded Locomotive

Fig. VI-32 - Reactor Equipment Layout

6. LOFT Area

a. Grading and Drainage

The grade area immediately surrounding the LOFT buildings has a minimum slope of 2% away from the buildings. Exact grading layouts will be completed by the architect engineer subject to approval by AEC representatives. The final swales should not be in conflict with the ditch drainage of the railway as shown in Fig. VI-1.

b. Roads and Pavements

An access road 32 feet wide (two 12 ft lanes with 4 ft shoulders) is provided between the road running south from the TSF area and LOFT facility. (See Fig. VI-1)

The shielded roadway (tunnel) will provide access to the control building when radiation dose rates are too high to use the surface roadway which connects the access road at the tunnel entrance with the test building area. This surface roadway is the same width as the access road to the LOFT area.

The shielded roadway (tunnel) extends 1000 ft from the test building and has a slight grade toward the control building. Within 30 ft of the building, the tunnel slopes away from the building, entering the building at the basement level in the turnaround area. Concrete curbs and footings are provided for the 10 gauge corrugated metal pipe arch that forms the tunnel. A 10 ft clear ceiling height is maintained with 11 ft clear inside width. Traffic lights will be located at each end of the tunnel to provide traffic control to the tunnel. The earth fill over the structure provides a maximum dose rate within the tunnel of 7.5 mrem/hr following a destructive test with 15% of the fission products released to the containment dome. The length of the tunnel (estimated at 1000 ft for cost purposes) provides emergency protection of personnel exiting the tunnel at normal walking speed following a reactor shutdown due to a loss of coolant. Personnel exiting the tunnel 15 minutes after shutdown will not receive a total dose greater than 5 rem due to direct radiation. The dose rate at the end of the tunnel during normal operations should not exceed 7.5 mrem/hr.

c. Parking Area

A 12-car bituminous parking area for vehicles is provided (see Fig. VI-15). The parking lot is provided with a headbolt heater rack for cold weather starting of vehicles.

d. Security

A security fence is provided around the entire LOFT installation. This fence is located as shown in Fig. VI-16 and conforms to ID Standards for location and construction. Ditch barriers are installed where the fence crosses ditches. The fence is a 2 in. diamond mesh chain link, No. 9 gage, galvanized wire 8 ft high with three strands of barb wire on 45 degree outriggers.

A 16 ft by 24 ft concrete block guard house adjacent to the sliding motor operated vehicle gate across the area access road controls traffic into the LOFT area. Besides the local gate control, provisions are made for remote control of this gate and the railroad gates from the control and utility building during the destructive tests.

e. Area Monitoring

The possibility of a hazardous release of airborne fission products from the containment shell of the LOFT facility necessitates a site monitoring grid to be operating downwind of the facility. From the grid instrumentation knowledge will be gained as to the quantitative and qualitative nature of the fission products released, for comparison with existing theoretical calculations of consequences resulting from a loss of coolant accident involving a power reactor. The compilation of this information is an intimate part of the loss of coolant tests to determine the consequences of a serious release of fission products within a containment vessel. The monitoring grid will also be used to provide radiological information for Health Physics personnel following a destructive test.

The monitoring grid consists of two 60° arcs of which one is located northeast and one southwest of the facility. These arcs are in the prevailing wind direction for both day and nighttime weather

conditions. Portable instrumentation and mobile units will monitor the remaining 180° sectors following the radioactive cloud as the prevailing winds are reversed. The monitoring instruments will provide the following information:

- (1) gamma dose rates from the airborne release,
- (2) total gamma and beta exposure dose from the release,
- (3) fallout distribution, concentration, and complete isotopic identification,
- (4) particulate activity in the effluent and complete isotopic identification,
- (5) noble fission gas and iodine concentrations in the release and complete isotopic identification, and
- (6) total exposure dose from fallout.

Phillips Petroleum Company will be responsible for location and design of the area monitoring system. However, the Architect-Engineer will provide the necessary electrical power cables to the individual stations.

f. Stack

Provisions are made to exhaust the containment vessel through a filtering system to a 150 ft high stack. Installed in the stack are stack monitoring probes connected to monitoring equipment located in a pumice block equipment vault at the base of the stack.

C. Utilities

1. CUA Utilities

a. Water (see Fig. VI-33)

(1) Raw Water. Water from two deep wells is pumped from 200 to 250 ft below grade by two deep well pumps (1000 gpm, 450 ft-hd, 150 hp). The water is chlorinated by means of an automatic chlorinator before it enters the 500,000 gallon ground level steel storage tank.

The storage tank is equipped with a level indicator, high and low level alarms, and high and low level pump start and stop switches. A bypass around the storage tank permits operation of the water systems when the storage tank is being cleaned or repaired.

(2) Potable and Service Water. Two service pumps (650 gpm, 200 ft-hd, 75 hp), one of which serves as a standby, are provided to furnish water for all service and domestic requirements at the LOFT facility. The pumps are also sized to furnish the evaporation, windage and blow down water requirements of the LOFT cooling tower.

(3) Fire Water. Fire water to LOFT, CUA, and future STEP facilities is furnished by either of two 4000 gpm, 300 ft-hd fire water pumps. A crosstie from the service water system to the fire water system maintains system pressure until a fire hydrant is opened. At this time the electric driven pump is started automatically on low fire water pressure. The standby pump is a gasoline engine driven unit for emergency fire water demands and is started automatically at a pressure lower than the starting pressure of the electric pump.

b. Heating and Ventilating

The fire and service pump building and the deep well pump houses which do not require continuous occupancy are heated to 60°F by means of thermostatically controlled electric unit heaters. Ventilation is provided by gravity roof ventilators.

c. Electrical Power (see Fig. VI-34)

Electrical power for the LOFT and the CUA is provided by commercial NRTS power from the TAN substation and also by diesel generator and motor-generator-battery units located at LOFT.

(1) Commercial Power. The two (2) existing TAN transformers are rated at 7500 kva or, with fans, at 9300 kva and 13.8 kv. It is intended to use these transformers to supply power to the LOFT-CUA complex. During the active GE-ANP program the average electrical demand was 3000 kva. Power allocations in the order of 2650 kva have already been committed to the Experimental Beryllium Oxide Reactor (EBOR)

(1450 kva) and the Lithium Cooled Reactor Experiment (LCRE) (1200 kva). Part of these loads would be included in the old ANP demand. Thus the capacity exists for the LOFT-CUA complex on either of the two transformers without compromising the integrity of the "dual feeder philosophy" for the existing plant areas. However, dual feeders are not considered necessary for the LOFT-CUA complex.

Necessary switchgear and transformers for stepping the voltage down to 4160 volts will be installed at CUA to serve the deep wells, fire water pumps, and service pumps installed at CUA. 4160 volt power from CUA will be routed to the LOFT facility in underground conduit. Additional switchgear and transformers will be installed at LOFT to facilitate distribution of the power at 4160 volts to all motors larger than 75 hp and at 480 volts for all smaller loads.

(2) Battery Power. LOFT will be equipped with a 15 hp dc motor driving a 10 kw ac generator to supply the necessary regulated power for reactor instrumentation. Power is supplied to the motor from a battery bank which, in turn, is floating on a continuous trickle charge from a 30 hp, 20 kw motor generator running on commercial power. In the event of a commercial power failure, the charging motor generator will automatically clear itself from the battery bank.

(3) Diesel Power. LOFT will be equipped with a 200 kw diesel driven generator which will power a 480 volt bus completely independent of commercial power. There will, however, be provided a tie breaker between commercial and diesel power to maintain power on the diesel bus even though the diesel is down for routine maintenance. Diesel power will be supplied to emergency coolant pumps, motors, rod drive motors, emergency lighting, etc.

2. LOFT Utilities

a. Water

(1) Supply. Chlorinated raw water is continually supplied to the LOFT facility through the 10 in. fire water piping originating at CUA. The maximum water consumption, including the cooling tower makeup requirements, is 650 gpm at an operating pressure of 85 psi.

(2) Potable and Service Water. Potable and service water is supplied to the control and utility and process buildings from the 10 in. fire main originating at CUA. Potable water, chlorinated at the well, is piped to various points in the control and utility building for domestic use. Service water is used for air compressor cooling, air conditioner cooling, demineralizer feed water and cooling tower makeup. Service water is also piped from the utility area to the coupling station in the test building.

(3) Fire Water. There are two standard fire hydrants strategically located in the yard area. Inside the control and utility building are two hose stations, one near the control room and one in the utility area. Approximately 3000 gpm is supplied from the 10 in. diameter fire water main originating at CUA to provide the maximum cooling tower fire water demand ($\approx$ 2200 gpm) one fire water hydrant (600 gpm) and miscellaneous service water requirements. An automatic pneumatic actuated sprinkler system and three hose stations are located on the deck of the cooling tower.

Water at 400 to 500 gpm is supplied to two high velocity water guns located within the test building. The remote controlled water guns are used for fire fighting and decontamination after a test has been completed. A hose station is also located within the test building. An automatic dry chemical fire protection system is provided in the process building and serves to protect the Dowtherm loop. The system is designed in accordance with NFPA Bulletin No. 17.

(4) Demineralized Water (see Fig. VI-35). Demineralized water is provided for boiler makeup and experimental use by deionizing raw water in two 100 gpm mixed bed demineralizers. The demineralized water produced has a specific resistance of 500,000 ohms and a total solid content of less than 2 ppm. Concentrated sulphuric acid for regeneration is supplied from a 500 gallon portable carbon steel tank which is filled from the technical services area acid supply. The acid is transferred to the mix tank by gravity. Flake caustic is mixed with demineralized water to provide the liquid caustic for regeneration.

Acid and caustic used for regeneration is mixed in separate open top 55 gallon plastic lined drums and diluted through separate eductor valves during regeneration of the beds.

The demineralized water, stored in a 20,000 gallon underground stainless steel storage tank, is distributed by a supply pump (150 gpm, 200 ft-hd, 10 hp) back to the demineralizer units for regeneration, to the boiler and to the coupling station in the test building.

b. Liquid Waste (see Fig. VI-36)

(1) Cold and Warm Liquid Waste. The cooling water from the air conditioning equipment and air compressor drains into a 5000 gallon warm sump tank. The warm liquid waste from the control and utility building also drains by gravity into the sump tank. The warm sump pump (500 gpm, 70 ft-hd, 15 hp) transfers the process effluent to the 15,000 sq ft leaching pond in the LOFT fenced area.

(2) Hot Liquid Waste. Radioactive liquid waste, consisting mainly of decontamination flush water from the test building, drains from two trenches in the test building into two hot filtering sumps. The sumps are equipped with fine mesh screens for collection of debris following a destructive test. From the hot filtering sumps the effluent gravitates into a 50,000 gallon hot waste tank; or in the case of mildly contaminated water, valves can be switched to drain the water into the warm sump tank for transfer to the leaching pond. The hot liquid can be pumped from the hot waste tank by one of two pumps (100 gpm, 50 ft-hd, 2 hp) to a truck loading connection or to the leaching pond. Sampling of the tank contents will dictate whether the effluent is transferred to the leaching pond or trucked to the liquid waste evaporator in the technical services area for ultimate volume reduction and storage.

(3) Sanitary Waste. The sanitary waste disposal system is sized for 20 people and consists of a conventional septic tank and leaching pit.

(4) Corrosive Waste. The acid and caustic regeneration wastes from the demineralizers flow directly to the leaching pond in a separate corrosion resistant line.

c. Solid Waste

The existing NRTS burial ground is adequate for the amount of solid waste expected from LOFT. Handling equipment for this waste is also adequate.

d. Plant and Instrument Air (see Fig. VI-37)

One 100 scfm single stage, oilless reciprocating air compressor furnishes air at 150 psig to a 300 ft³ receiver. The 1 in. plant air header is supplied directly from the receiver through a motor valve which automatically closes if the receiver pressure drops to 100 psig. Thus an adequate instrument air supply is reserved in the receiver to operate instruments and motor valves. Plant air on leaving the receiver is reduced to 100 psig by means of a pressure controlled motor valve. The instrument air, after leaving the driers, is reduced to 50 psig by means of a pressure controlled motor valve and routed via a 1 in. header throughout the control and utility building and to the test and process buildings. Instrument air is further reduced to 20 psig by small locally mounted reducing valves.

e. Heating and Ventilating and Exhaust (see Fig. VI-38)

The heating and ventilating equipment for the control and utility building and the test building is located in the utility area. All fresh air to the two buildings is routed through a common intake plenum which houses non-freeze preheat coils and coarse filters.

High efficiency filters (minimum DOP Eff. 99%), activated charcoal filters, and tempering coils are located on the suction side of a 10,000 cfm, 15 hp fan which supplies air at the rate of 5 air changes per hour to the control and utility building. Sixty percent of the supply air is returned for recirculation. The remaining 40% is exhausted by two 2000 cfm, 1/4 hp exhaust fans located in the utility area. A 7-1/2 ton

self contained air conditioner is used to maintain 70° in the control and data reduction rooms.

Air is supplied to the test building by a separate 35,000 cfm, 25 hp fan equipped with tempering coils at the rate of 4 air changes per hour. Fifty percent of the air is returned to the suction side of the fan when permissible with the remaining 50% being discharged directly to the waste gas stack by the 35,000 cfm, 40 hp exhaust blower through a bypass around the exhaust filtering equipment. When contamination exists in the test building, flow through the supply and exhaust fans is reduced to 10,000 cfm, and the exhaust air is routed through the high efficiency filters (minimum DOP Eff. 99%) and the radioactive iodine removal equipment and is then discharged to the atmosphere through the 150 ft high waste gas stack.

Butterfly valves rated at the test pressure of the test building are installed in all ducts entering or leaving the test building and serve to provide containment integrity during a test. The valves are remotely operated from the control area to permit operation of the exhaust system after a test.

The process building houses the secondary and tertiary heat dissipation equipment. Since it is generally a machinery area, it is heated to 60°F by means of industrial type steam unit heaters equipped with fans, dry type washable filters, non-freeze heating coils and dampers. Fresh air intake dissipates heat generated from equipment. During winter months this amounts to approximately 2 air changes/hr and is increased to approximately 4 air changes/hr during the summer conditions. Exhaust is through powered roof ventilators.

The fuel oil pump house which does not require continuous occupancy is heated to 60°F by means of an industrial steam unit heater. Ventilation is provided by gravity roof ventilators.

f. Steam Plant (see Fig. VI-39)

Two 5000-pound-per-hour water tube boilers are located in the utilities area. These boilers supply saturated steam to the plant facilities and reactor decay heat removal system via a 2.5-inch

steam header. The boilers are packaged type, equipped with the conventional safeties and controls, and burn steam-atomized No. 6 fuel oil. Demineralized water is supplied for makeup.

Auxiliary equipment includes a 12,500-pound-per-hour deaerating feed water heater, two motor-driven feed water pumps (25 gpm, 700 ft tdh, 7.5 hp) and a 500-gal blowdown tank. Chemical feed systems are also provided to treat the boiler water with disodium phosphates for pH control and to treat the feed water with sodium sulfite for residual oxygen removal.

A 500-gal condensate surge tank serves as a collection and surge tank for the condensate return from the various facilities. Two motor-driven condensate pumps (25 gpm, 125 ft tdh, 1 hp) return the condensate from the tank to the feed water heater.

One motor-driven fuel oil unloading pump (50 gpm, 5 hp), two motor-driven fuel oil transfer pumps (2.5 gpm, 200 ft tdh, 0.5 hp), two duplex fuel oil strainers, and two fuel oil heaters are located in the oil pump house adjacent to the fuel oil storage tank.

The fuel oil storage tank has a storage capacity of 25,000 gallons. This storage volume is sufficient to fire one boiler at a maximum rate for a period of 20 to 30 days. The storage tank is insulated and equipped with a suction heater. The oil pumped to the boilers in excess of that burned is recirculated to the storage tank. Two displacement type meters, installed in fuel oil supply and return lines, measure the quantity of oil burned.

g. Communications

A commercially installed and serviced telephone system connected to the existing NRTS commercial system provides for primary operational communications. In addition, an intercom system with two-way units located in all normal working areas and a master station in the reactor control room is provided. The master station is capable of speaking to all locations collectively or selectively.

Manual/automatic coding fire alarm systems are strategically located throughout the LOFT area and consist of alarm boxes connected to centralized relay and terminal strip units. The coded fire alarm annunciates at the TAN fire station, at the main TAN security station, and at the NRTS central fire station as well as at the reactor process control room.

An evacuation siren is located in the reactor area and is actuated from the reactor control room.

Fig. VI-33 - LOFT Raw Water Distribution

Fig. VI-34 - LOFT/CUA Electrical Power

Fig. VI-35 - LOFT Demineralized Water Flow Diagram

Fig. VI-36 - LOFT Liquid Waste System

Fig. VI-37 - LOFT Plant and Instrument Air Flow Diagram

Fig. VI-38 - LOFT Heating and Ventilating Flow Diagram

Fig. VI-39 - LOFT Steam System Flow Diagram

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