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PHILLIPS
PETROLEUM
COMPANY

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ATOMIC ENERGY DIVISION

NATIONAL REACTOR TESTING STATION
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REACTOR SAFETY

STEP PROJECT

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Atomic Energy Division
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PREFACE

The purpose of this report is to present an engineering test program for investigating a loss of coolant accident for a pressurized water-cooled and -moderated reactor system. In addition to the program description and fundamental considerations which lead to the proposed program concepts, an experimental system and containment facility are described to implement the test program.

This report supersedes report IDO-16833 entitled Feasibility and Conceptual Design for the STEP Loss of Coolant Facility dated December 2, 1962, which has served its intended purpose of establishing the basic loss of coolant program concepts and providing criteria for initiating the design of the experimental system and facility. The experimental and facility descriptions contained herein reflect the design as it existed May 1, 1964, at the completion of Title I as prepared by the LOFT Architect Engineer (Kaiser Engineers) and the nuclear designers (Phillips Petroleum Co.). The radiological hazards resulting from the loss of coolant test program as previously postulated in the earlier report have been further analyzed and reported in IDO-16981 Preliminary Safety Analysis Report (PSAR) - LOFT Facility dated April 1964.

ABSTRACT

An engineering test program is presented to investigate a loss of coolant accident for a water-cooled and -moderated reactor using a complete nuclear plant. The preliminary design of the reactor and a facility, designated as LOFT, to be used for carrying out the test program is also described.

The experimental program is part of the U. S. Atomic Energy Commission's nuclear safety engineering and test programs. The purpose of the program is to demonstrate the consequences of a loss of coolant accident, to provide quantitative information on the behavior of nuclear plant components and fission products, and to determine the phenomena of principal importance in determining the accident consequences, thereby providing future guidance to the related research and development programs. The experimental program will be closely coordinated with related research programs sponsored by the AEC.

The test program will consist of five phases: Phase I -- Containment Pressure and Leak Tests, Phase II -- Coolant Blowdown Tests With a Dummy Core, Phase III -- Low Power Physics Tests and Power Operation, Phase IV -- Loss of Coolant Test, and Phase V -- Facility Cleanup and Post-Test Examination.

The reactor is a typical pressurized water reactor capable of operation at 50 MW(t). The reactor and primary coolant system is mounted on a railroad flatcar and will be installed in a conventional dry containment vessel. The containment contains special provisions for obtaining fission product samples and decontamination. The reactor core design is being carried out by Phillips Petroleum Company while the balance of the reactor system and facility is being designed by Kaiser Engineers.

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I. INTRODUCTION

As part of the Atomic Energy Commission's nuclear safety programs sponsored by the Division of Reactor Development, an engineering test program was initiated in 1962 for the purpose of demonstrating the consequences of a loss of coolant accident for a water-cooled and -moderated reactor and investigating the effectiveness of inherent and engineered safeguards. This experimental program is to be carried out by the Safety Test Engineering Program (STEP) of Phillips Petroleum Company at the National Reactor Testing Station. The purpose of this report is to present the experimental program and to describe the preliminary design of the reactor and facility as it existed as of May 1, 1964. The facility for carrying out the test program has been designated as the Loss of Fluid Test (LOFT) facility.

Earlier activities of the STEP project consisted of a study [1] to determine the nature and scope of the test program required to provide the safety information needed by industry and the AEC and to determine the feasibility of conducting the test program and general concept of the reactor and facilities required. This study was reviewed by various AEC agencies and industrial groups. Suggestions received from these groups were incorporated into the experimental program and facility concept to the extent possible within budget limitations and technical feasibility and are included in the experimental program presented herein.

During the past year, additional analytical studies also have been initiated to obtain a better understanding of the influence of various parameters on the phenomena which occur during and following a loss of coolant accident and to assess the instrumentation requirements. These studies have contributed to the overall nuclear safety effort by clarifying various aspects of the loss of coolant accident and by identifying areas where additional analytical and research efforts must be focused. In addition, preliminary design of the reactor and LOFT facility was started. Responsibility for design of the reactor core, core support structures, and control rod drive mechanisms has been assigned to Phillips Petroleum Company while responsibility for other reactor system components and the LOFT facility has been awarded to Kaiser Engineers. The Babcock and Wilcox Company, under subcontract, will review the overall design of the reactor system and conduct independent analytical studies.

The design philosophy, detailed experimental program, and description of the LOFT facility are presented in the following sections of this report.

II. REVIEW OF NUCLEAR SAFETY

1. REACTOR SAFETY AND SITING

The siting of nuclear reactors is one of the most critical problems confronting the continued development and use of nuclear power for generating electricity. The location of nuclear plants close to population centers is becoming increasingly necessary if nuclear power is to compete economically with power generated by conventional means. Before reactors can be located in large population centers, reactor designers and operators must satisfy themselves, AEC regulatory authorities, and the general public that the health and welfare of those people in the vicinity of nuclear reactors will not be endangered as a result of a reactor accident. The siting of nuclear reactors and the future development of nuclear power is therefore intimately related to reactor safety.

2. MAXIMUM CREDIBLE ACCIDENT CONCEPT

In order to assess the safety of a reactor system and to determine the suitability of a site for a given reactor, consideration should be given the identification of conceivable accidents and evaluation of these accidents as credible or incredible. The Atomic Energy Commission's site criteria [2], which are intended to be used as guides in site selection, use the concept of the "maximum credible accident" as a point of departure in such analysis and evaluation. The "maximum credible accident" is defined as the upper limit hazard, that is, fission product release, which must be used to analyze the safety of the reactor and select a site. Although this concept has important limitations with respect to the assignment of the probability of an accident and with respect to analyzing the consequences of the complex sequence of events which make up any nuclear accident, the concept appears to be both necessary and useful in any safety evaluation.

Historically, two types of nuclear accidents have dominated nuclear safety thinking since the origin of the nuclear industry. That is, they have at some time been considered to be maximum credible accidents. These are: the reactivity accident or reactor excursion and the loss of coolant accident.

During the early stage of development of nuclear power, the reactivity accident was generally considered to be the maximum credible accident and received the most attention as indicated by the 1955 Geneva conference paper by McCullough, Mills, and Teller. Due largely to the lack of experimental information it was generally believed that reactivity insertions which would bring the reactor above prompt critical would be extremely dangerous and should be avoided. Therefore, early nuclear safety investigations were aimed at the development of equipment and procedures for preventing the initiation of a reactivity disturbance and directed at determining the transient response characteristics of the reactor. Beginning with the Borax experiments by the Argonne National Laboratory and continued in Phillips Petroleum Company's Spert experiments, Atomics International's KEWB experiments, General Atomic's TRIGA experiments, Argonne National Laboratory's TREAT experiments, and

Los Alamos Scientific Laboratory's Godiva experiments, the accumulation of safety information from these experiments now allows a much more realistic and less restrictive attitude to the reactivity accident. It is now generally recognized and accepted that the transient behavior of many reactor systems can be confidently described analytically and can be designed to safely withstand super-prompt critical excursions.

Coincident with the change in attitude toward the reactivity accident in recent years the loss of coolant accident has come to be generally used as the maximum credible accident for water-cooled and -moderated reactors. Therefore, in any consideration of reactor safety and siting and the formulation of nuclear safety experimental programs attention must be focused on the loss of coolant accident, that is, on the nature of the accident and on provisions for preventing the accident or reducing or eliminating the consequences.

3. WAYS OF ACHIEVING SAFETY

Having identified the type of reactor accident of principal concern to reactor safety at this time, it is useful to distinguish the different ways of achieving safety. These may be categorized as follows: inherent characteristics of the system which may assist in preventing the accident or reducing the consequences, external controls and administrative procedures, and isolation or confinement of the consequences. In practice, a combination of all three are used in varying degrees to provide adequate safety margins.

In evaluating each of the means available for achieving safety, in the case of the loss of coolant accident it is apparent that the inherent characteristics of a water-cooled power reactor will not absolutely prevent the initiation of an accident. However, the inherent behavior of the fission products, such as settling, adsorption, absorption, and condensation, which are largely unknown, may be of importance in limiting the consequences of this accident. External controls are at present confined to rigid inspection during fabrication and throughout the operating life of the reactor, since instrumentation which is capable of preventing a failure of primary coolant system piping or other component is nonexistent. Since the first two of the ways for achieving safety do not provide absolute assurance that a loss of coolant accident will not occur, the third way, that is, isolation and containment, must be relied upon and, in fact, has been emphasized.

AEC site criteria, 10 CFR 100 [2], and the example calculation presented in TID-14844 [5], appear to stress reactor isolation. The possibility exists, however, that engineered safeguards, such as, containment, spray-washdown systems, cleanup filter systems, vapor suppression systems, and pressure relief systems, can be substituted for isolation. The substitution of engineered safeguards for isolation can only be achieved by experimentation and demonstration tests which will verify the reliability and effectiveness of the various engineered and inherent safeguards and which will provide the information necessary to understand all phases of potential accidents and their consequences.

4. SAFETY INFORMATION NEEDED REGARDING THE LOSS OF COOLANT ACCIDENT

The loss of coolant accident involves a sequence of complex interrelated events. In brief, these events are: rupture of the primary coolant circuit, release of the coolant from the primary coolant loop to the containment, violation of the integrity of the fuel cladding, release of fission products to the primary coolant system, deposition of fission products within the primary coolant system, and transport of those not deposited to the containment, dispersal of fission products within the containment, removal of fission products by inherent processes, and escape of fission products from the containment. Each of these events is intimately related and each is affected by a large number of both dependent and independent variables. For example, the size and location of any rupture which may occur in the primary coolant piping will significantly influence the transient thermal behavior of the fuel and subsequently the quantity of fission products released and the dispersal of these fission products. As a second example, the quantity and type of fission products released will be dependent upon such things as the fuel and cladding material and the degree of fuel burnup.

Because of the complex nature of the problem and the general lack of experimental data, rigorous analysis has not been possible. In order to perform a safety analysis for the purpose of selecting an acceptable site, reactor designers have relied upon showing that regardless of the quantity of fission products released, confinement of the fission products could be achieved. Although this approach provides adequate safety margins in such a situation, both the designer and safety evaluator must necessarily be conservative at every point at which judgment enters. Therefore, before advances can be made in the direction of reducing safety requirements, the development of experimental data will be required.

The nature and complexity of the safety questions and the need to develop on a timely basis information pertinent to safety evaluation and site selection suggests experimental programs of two basic types. These are: fundamental or basic research programs and large scale engineering or demonstration tests.

The basic research programs should be aimed at providing quantitative experimental data in the following general areas: (a) piping materials failure modes, (b) coolant blowdown phenomena, (c) clad and fuel melting, (d) fission product release, (e) fission product dispersal and inherent removal mechanisms, (f) containment leak measurement techniques, and (g) fission product fractionation and escape through small openings. Experiments which investigate the influence of each of the many variable parameters are needed to ultimately provide an understanding of each of these phenomena and to assist in interpreting and applying data obtained in integral system tests. Research and development programs have been initiated by AEC in many of these areas to provide the basic experimental data needed. Many of the programs, however, are in the early stages of development. For example, at present little is known regarding such things as fission product release in air-steam atmospheres and fission product dispersal and transport mechanisms.

Because of the complex interrelation of phenomena which occur during a reactor accident, two categories of engineering tests will be required. These two categories are: (a) scale model or full scale testing of reactor subsystems and engineered safeguard systems, and (b) large scale testing of a complete reactor plant. In the first category, such things as the following will require investigation: (a) coolant blowdown, ie, the influence of reactor and coolant system design on the mechanical and thermal behavior of the reactor and primary system components, (b) fission product transport during coolant blowdown, (c) fission product washdown or spray systems, (d) fission product filter systems, (e) vapor suppression systems, (f) pressure relief systems, (g) the relationship of containment air leak tests to fission product leakage under high humidity conditions, and (h) the efficiency of double penetrations. The second category of engineering test will consist of testing a complete reactor plant for the purpose of tying together the data obtained from basic research and engineering tests and obtaining guidance on important aspects of the meltdown phenomena. Only if the basic research and engineering tests are closely planned and coordinated can the results be interpreted and applied in a meaningful way to the critical problems of reactor safety and site selection. Integral engineering tests conducted solely for the sake of demonstrating that the consequences of a loss of coolant accident will not endanger the general public are not adequate to permit application of experimental results to the safety problems of water-cooled and -moderated reactors.

The test program discussed in this report is an engineering test program to investigate the behavior of a large scale reactor plant during a loss of coolant. The program objectives and general concept of the program are presented in the following section.

III. TEST PROGRAM CONCEPT

1. SCOPE AND OBJECTIVES

The scope of the program to investigate the loss of coolant accident consists of three parts. These parts are: analytical studies of all phases of a loss of coolant accident, a series of tests performed on a large-scale reactor plant, and analysis and interpretation of test results and correlation of the results with those of related research programs. The broad objectives of the program are: (a) to demonstrate the consequences of a loss of coolant accident for a water-cooled and -moderated reactor and to determine the capability of a containment building for retaining fission products, (b) to provide quantitative information on the behavior of reactor and plant components and the behavior of fission products, and (c) to determine which phenomena associated with the loss of coolant accident are of principal importance in determining the consequences. Figure 1 depicts an experimental program schematic outlining the interrelated analytical and experimental requirements to meet the program objectives.

2. ANALYTICAL PROGRAM

In order to achieve the above objectives, the initial efforts of the program will be to conduct analytical studies of the various phenomena which occur during a loss of coolant. At the present time, there appears to be no general agreement among technical people as to the combination of conditions which constitute a maximum credible accident; ie, what type and location of component failure, what coolant operating conditions, and what reactor design factors will result in the greatest fission product release. Furthermore, calculational techniques have not been developed which can be used to predict with assurance the transient thermal behavior of a reactor core and the fission product release and dispersal to a containment building. In view of this situation, studies will be made in the following areas: parametric studies of coolant blowdown phenomena to determine the influence of such things as rupture size, rupture location, coolant pressure, and coolant temperature on the mechanical and thermal response of reactor components, primary coolant system components, and the containment building; studies to determine the influence of these same parameters on fission product transport and dispersal; studies to determine the expected range of pressures, temperatures, and humidity conditions in a primary coolant system and containment building, and parametric studies of the transient thermal behavior of reactor fuel following a loss of coolant. The studies will utilize and correlate the results of related research and development programs completed or in progress.

The analytical program has already been initiated and a brief summary of the results to date are presented in Section V. The purpose of these studies is as follows: to assist in formulating the experimental program and in selecting the number of tests and test conditions, to provide a point of departure for the development of analytical models and calculational techniques, and to assist in

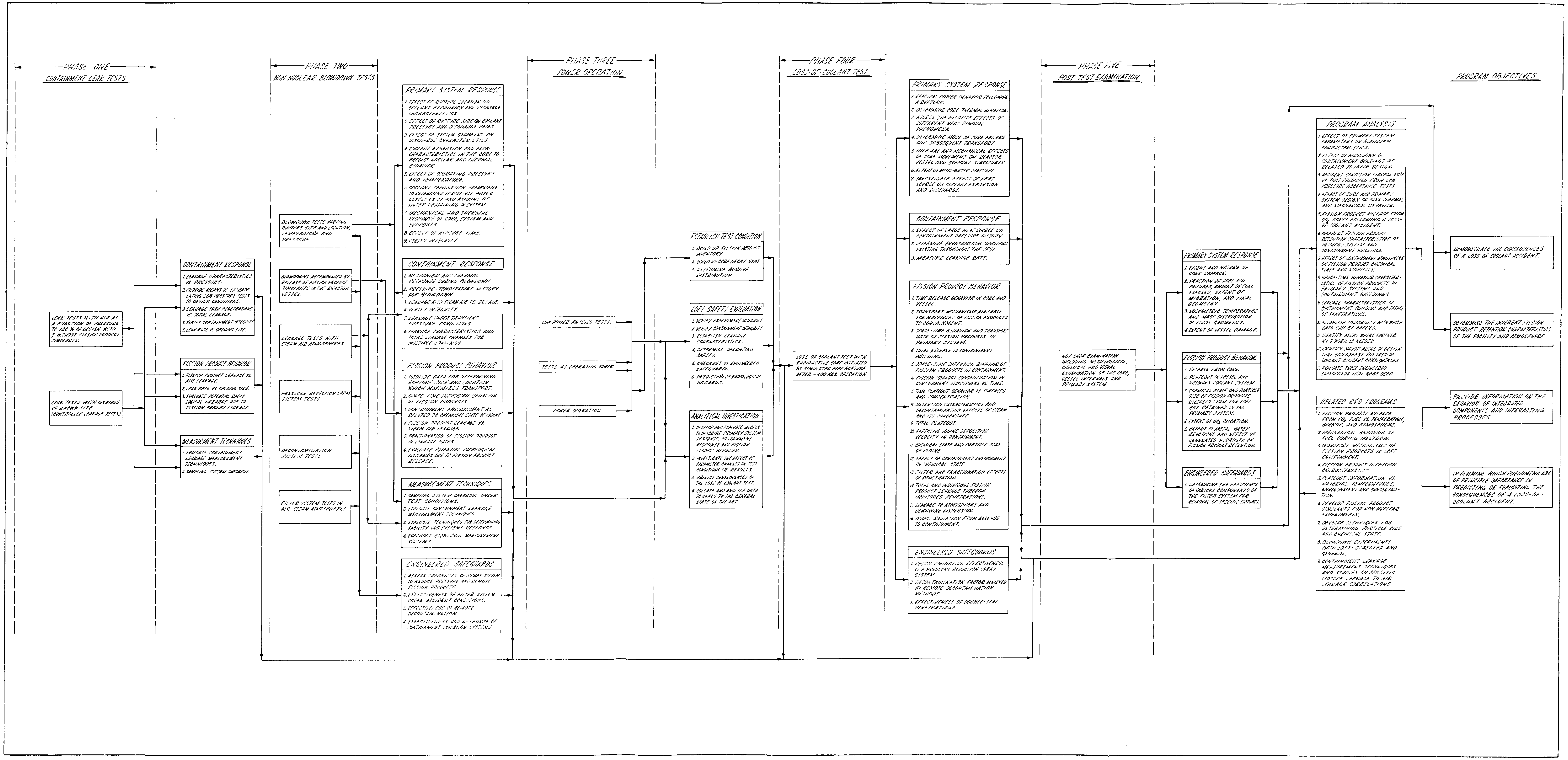


FIG. 1 LOSS OF COOLANT PROGRAM SCHEMATIC.

determining the experimental measurements which should be made and the selection of instrumentation.

3. ENGINEERING TEST PROGRAM

The second part of the program required to attain the program objectives consists of a series of tests to be carried out on a large reactor plant. The reactor plant is not a prototype of any existing or planned reactor but is designed to be representative of the general class of "advanced" pressurized water reactors.

In formulating the test program, the basic philosophy has been as follows: (a) to conduct tests which simulate actual loss of coolant accident conditions as nearly as practical within the limits of economic and technical feasibility; (b) to conduct tests which, first of all, will provide information on the inherent mechanisms available for reducing or limiting the accident consequences and, secondly, where practical, will provide information on engineered safeguards with emphasis on the containment; and (c) to conduct tests which are closely coordinated with related nuclear safety research and development programs and which are needed to interpret the results of the final loss of coolant test and to apply the results to the general class of water-cooled reactors.

The test program formulated has been divided into five phases. These phases are: Phase I -- Containment Pressure and Leak Tests, Phase II -- Coolant Blowdown Tests with a Dummy Core, Phase III -- Low Power Physics Tests and Power Operation, Phase IV -- Loss of Coolant Test, and Phase V -- Facility Cleanup and Post-Test Examination.

The containment pressure and leak tests, Phase I, are included for the purpose of providing data with which to correlate conventional containment building air leak tests to the fraction of fission products which can be expected to escape. These data are needed both to interpret and to apply LOFT test results to other containment systems.

The preliminary blowdown tests, Phase II, using the complete plant and a dummy core will simulate an actual rupture in the primary coolant piping near the reactor vessel within the limitations of space and engineering feasibility. Both the rupture size and location will be varied. Orifices will be used to control the rupture size, thus enhancing the ability to analyze the phenomena which occur. The blowdown tests will be conducted with core support grids of "conventional" design as well as with support grids which are specifically designed for the forces expected. A recognized limitation of the test results is the lack of decay heat in the dummy core. However, the lack of decay heat has the advantage of permitting closer correlation of results with laboratory experiments.

These preliminary blowdown tests are intended to provide the following information: determine the conditions which will result in dispersal of the largest quantity of fission products and to obtain quantitative data concerning coolant blowdown needed to improve the analytical models used for predicting the LOFT test results, extrapolation of LOFT results to other reactor systems, and to correlate the LOFT data with data obtained in other laboratories. Tests also are included to provide preliminary data on fission product leakage under realistic pressure, temperature, and humidity conditions and to provide preliminary information on the effectiveness of a containment spray system for the removal of fission products.

Phase III of the program is not an experimental phase, but rather a reactor operational period required to build in fission products. The philosophy behind the length of the operating period, ie, the percent burnup, is discussed further in Section IV.

The final loss of coolant test, Phase IV, will model an accident approaching the maximum credible, except that a low burnup core will be used. The present test program concept includes only a single test with a radioactive core. Consideration has been given to testing engineered safeguards, such as a core spray, which prevent core meltdown and to conducting a series of tests which approach core damage without fuel damage and fission product release actually occurring. Rather than to test a core spray at the outset, the philosophy has been to first conduct tests which will provide information on the inherent fission product removal mechanisms and which will assist in predicting the maximum credible accident. Tests to investigate preventive measures could then be undertaken. A series of tests which approach core damage does not appear economically feasible, since it appears that each test would require a new core because of fuel element warping and bowing and because it does not appear possible to provide instrumentation which could withstand more than one test.

Experimental conditions for the test will be selected to maximize the release of fission products from the fuel and the dispersal of the fission products to the containment. However, should the fuel reach the bottom head and melt through, provisions have been made for retaining the fission products in order to provide information on the diffusion of fission products from the simulated rupture rather than complicate the interpretation of results by permitting escape to the containment through the bottom head.

The information and data to be obtained from the loss of coolant test is presented in detail in Section V. In general, emphasis will be placed on obtaining information on the following: the physical behavior of the core and the fraction of the fission product inventory released from the fuel, the nature and magnitude of the fission products reaching the containment building and the efficacy of the natural mechanisms for removal, and the fraction of the fission product inventory which escapes the containment.

The final phase of the program, ie, post-test examination, Phase V, will consist of detailed metallurgical and chemical examination of the remains of the reactor core in the available hot shop facilities. Consideration has been given to obtaining photographs of the core behavior and sampling for fission products in the primary coolant system as a function of time. It appears economically and technically impossible to obtain either documentary photography or adequate sampling. The purpose of the post-test examination is, therefore, to evaluate the final configuration of the core, determine the damage to the vessel and internal components, and assess the magnitude of the fission product plateout in the coolant system by post-test examination.

4. DATA INTERPRETATION AND ANALYSIS

Upon completion of the LOFT test program, the efforts of the project will be directed as follows: reporting, analysis, interpretation, and evaluation of the results of LOFT tests; synthesis of LOFT test results with those of related research programs and reporting of results in a manner that the results can be utilized and applied by AEC and industry; and studies to develop analytical models and calculational techniques which accurately describe the observed experimental results. Only if the experimental results are carefully and completely analyzed can they be interpreted in a meaningful way and applied to nuclear safety problems. The purpose of this portion of the program is to fulfill this need.

5. RELATION TO RESEARCH AND DEVELOPMENT PROGRAMS

Recognizing the complexity of the loss of coolant accident, it is evident that the LOFT program cannot be expected to answer all of the safety questions which can be raised and provide the understanding ultimately needed concerning this accident. Only by a careful balance between the nuclear safety engineering test programs and the research programs can the needs of industry and AEC be met. By closely coordinating the LOFT program with other related research programs, the information obtained in the Commission's overall safety programs can be interpreted and presented in a meaningful way and applied with some assurance to siting and safety problems. An objective of the LOFT program is, therefore, to coordinate both the LOFT analytical and experimental effort with similar efforts of other laboratories.

Coordination of the LOFT program with the fission product behavior studies at ORNL and BNL, and with the blowdown and containment studies at Hanford Laboratories has already been initiated. Experiments are planned to investigate the following broad areas: coolant blowdown; instrumentation; fission product sampling and analysis; decontamination; fission product release, plateout, and transport; fuel meltdown; fission product removal; and containment leak test equipment and procedures. It is the intent that the nature of the experiments to be conducted and the instrumentation and sampling techniques will be coordinated to the extent that results obtainable at one laboratory can be compared directly to the results obtained at another. By coordinating the safety programs in this manner, maximum benefit should accrue to the AEC and industry.

IV. EXPERIMENT AND FACILITY DESIGN CONCEPTS AND PHILOSOPHY

1. GENERAL

The preliminary design of a reactor system and a facility, designated as LOFT, has been prepared for carrying out experimental investigation of the loss of coolant accident. In choosing the reactor and facility design to achieve the program objectives, consideration has been given to a number of factors, notably, economics, the site selected for LOFT, use of existing facilities, safety, cleanup, and the desirability of providing a nuclear plant which is representative of the general class of advanced pressurized water reactors. Compromises have necessarily been required by controlling certain variables in order to conduct an understandable experiment. The general philosophy behind the experiment and the facility design chosen is discussed in this section.

1.1 General Concept

The reactor to be used for carrying out the experimental program will be a conventional pressurized water reactor having a power capability of about 50 MW(t). The reactor will be fueled with low enrichment UO_2 clad in stainless steel in the form of rods. The core has an active height of 36 inches and is 42 inches in diameter. The coolant system is a single loop, single pass system with provisions for simulating a pipe rupture at two locations (coolant inlet and coolant outlet) near the reactor vessel. The entire primary system, including a shield tank surrounding the reactor vessel, is mounted on a four-rail railroad flatcar.

The reactor will be installed within a conventional dry containment building 70 feet in diameter by 127 feet high having a free volume of 302,000 cubic feet. The containment building is provided with the following special features: concrete missile shield, pressure reduction sprays, remote decontamination system, remote fission product sampling stations, remote decoupling station, and monitored penetrations. A secondary heat removal system of 50 MW capacity and a filter system also are provided. The reactor may be removed from the containment building to a hot shop through a large door equipped with a double pressurized seal. A shielded locomotive will be used to move the reactor.

1.2 Siting and Use of Existing Facilities

The LOFT facility has been located adjacent to the Flight Engine Test (FET) facility at the Test Area North (TAN) of the National Reactor Testing Station. This site was chosen for economic and safety reasons. The underground control rooms, auxiliary and equipment, and utility systems existing in the FET facility were specifically designed for remote operations as part of the program (ANP) and, therefore, have been used to the fullest extent to achieve economic savings. The four-rail railway existing between the FET and the TAN hot shop facilities and the shielded locomotive also were used to advantage in the design. Since these facilities permit transport of heavy radioactive equipment, they have been used to reduce the cost of cleanup in the test area and to transport the reactor components to the existing hot shop for examination upon completion of the experimental program.

With respect to safety, the site selected is well isolated from the nearest population center. The test area is designed for controlled access by operating personnel during performance of the experimental program. The climatology, geology, and seismology of the area are well known and data are available to assist in safety assessment.

2. EXPERIMENTAL DESIGN

2.1 Nuclear Characteristics

2.11 Core Size and Power Level. In the design of the reactor core to be used for the loss of coolant experiments consideration has been given to achieving the stated program objectives at a minimum cost. The criteria established for the reactor core were as follows: (a) to provide a core with physical and nuclear characteristics typical of those found in pressurized water power reactors, (b) to provide a core of a size and fuel power density such that both the temperature and physical behavior following a loss of coolant would approach that of most power reactors, and (c) to provide a reactor of minimum power capability consistent with the other criteria.

Analytical studies were conducted of various core sizes and fuel power densities. These studies indicate that unless either the fuel power density is increased considerably beyond that of existing power reactors or non-conventional means are provided for preventing heat loss by radiation, such as insulation or reflectors, the minimum size core capable of meeting the criteria is about 40 inches in diameter. A fuel power density of 256 kW/l was chosen since it approaches the upper limit of the fuel power density for most existing and proposed large power reactors. Although in theory, it appears possible to devise a means whereby the thermal behavior of a small core could be made to approach that of a large core, in practice, at the present time, calculational techniques do not exist which will permit the design of such a core and the associated heat loss or reflector device with any assurance. The general philosophy, therefore, was adopted for LOFT to design a core with a temperature behavior approaching that of a large power reactor core without the use of special devices for forcing a given temperature response.

2.12 High Burnup Versus Low Burnup. In designing a reactor core for the loss of coolant experiment, several alternate methods were considered with respect to fuel burnup, namely (a) new core with low burnup fuel, (b) new core with high burnup fuel, (c) previously operated high burnup core, and (d) seeding of new fuel with stable iodine to represent a high burnup core.

Use of virgin fuel and operating for a minumum length of time to establish near equilibrium values of the short half life isotopes, in particular the iodines, permits savings of both time and money. No special fuel handling equipment is required and the duration of full power operation can be reduced to approximately 400 hours. Use of low burnup fuel does, however, introduce recognized problems. Firstly, the ratio of stable iodine to radioactive iodine will vary from low burnup to high burnup fuel, ie, large quantities of stable iodine are not present in the low burnup fuels. Therefore, the fission product release and behavior from a low burnup core may not be identical to that which would be obtained from a core operated for an extended period. Secondly, high burnup cores inherently build up larger internal fission gas pressures within the fuel pins which could potentially change the fission product release as the fuel cladding temperature approaches melting. Nevertheless, extrapolation of fission product release to high burnup cores appears to be possible using experimental data from research programs now in progress.

High burnup can be obtained in a new core by operating for an extended period of time. As in the case of a low burnup core, no costly fuel handling equipment is required and experimental instrumentation can be easily attached to the fuel pins prior to installation in the reactor. The primary disadvantage is the long period of operation required to attain fuel burnup. Use of a new core operated to high burnup would thus delay the experimental program for at least a year. Further, it may not be possible to insure the integrity of in-core instrumentation.

Reduction in the extended operating time to attain a high burnup fuel can be accomplished by use of an existing core which has already been operated in a power reactor. This approach has the disadvantage that adequate in-core instrumentation can not be provided and, further, this approach requires expansive shielded handling equipment to permit shipping and loading of the reactor.

Seeding of new uranium dioxide fuel with stable iodine has been considered to reproduce the stable iodine concentration which exists in high burnup fuels. Fabrication of a core to which stable iodine has been added would require a development program and has, therefore, not been considered.

Economic considerations, the need to install temperature and strain instrumentation on the fuel rods and core supporting structure, and the need to provide experimental data at an early date contributed to the selection of a new core with low burnup. Further reasearch and development programs are required in support of the LOFT experiment to verify the correlation and ability to extrapolate fission product release to high burnup cores.

2.13 Zirconium Versus Stainless Steel Clad. Fuel technology on existing and proposed pressurized and boiling water power reactors was investigated with reactor designers and fuel fabricators to assist in selecting the fuel cladding material for the LOFT experiment. The studies were conducted late in the year of 1962 to establish the basic concept for the LOFT reactor design. At that time the predominate fuel cladding material in existing power reactors was stainless steel. Experimental investigations were in progess to determine the merits of thin wall, free standing, stainless steel clad versus zirconium clad fuel rods for future generation power reactor cores. It was generally concluded that the stainless steel clad would be more economical and would continue to be favored as a cladding material for water cooled power reactor cores for several years. As a result of the study and the preponderance of stainless steel clad cores to date, the decision was made to design the LOFT I core with stainless steel clad fuel rods. At the writing of this report the outlook has changed and many of the planned pressurized and boiling water power reactors are being proposed with zirconium clad fuels to benefit from the inherent neutron economies. It is recognized that the use of zirconium cladding increases the possibility and the severity of metal-water reactions, and further that the hydrogen generated may significantly influence fission product behavior by combining with iodine to form soluble hydrogen iodide. Recent studies on the conversion of the LOFT experiment to zirconium clad fuel indicate that time and money penalties would result in the loss of coolant program if such a change were incorporated into the reactor design at this time. For these reasons no change has been made in the LOFT core.

2.2 Dolly Mounted Concept

Assignment of facilities at the Test Area North (TAN) for use in conducting the loss of coolant experimental program has directly influenced the experiment concept. In particular, the FET facility was already connected by heavy duty four-rail trackage for transport of large nuclear test assemblies to the central services area. The central area, presently designated as the Technical Services Facilities (TSF) was designed to service a large variety of relatively unshielded radioactive nuclear systems. Implementation of the mobile concept for the LOFT reactor permits maximum utilization of the existing facilities, reduces the cost of the LOFT facility, enhances cleanup and decontamination of the containment building, and allows a post-test examination of the reactor and associated systems. Economies in time and money are realized during the fabrication of the experimental assembly and construction of the test facility using the dolly mounted concept, as the two efforts can proceed simultaneously in two separated areas. It is, therefore, feasible to fabricate the experimental equipment in the cold assembly building in the TSF area utilizing the existing machine shops, handling equipment, etc, without interference from the facility construction.

The effect on the test results caused by implementation of the dolly mounted concept is not believed to be significant. A permanently installed reactor system with the attendant biological shield could potentially provide a more restricted path for fission product transport, depending upon the location of the postulated rupture; however, because of the difficulty in simulating a transport path representative of any power reactor installation, the approach has been to design an experiment that would provide the least resistance to the transport of fission products to the containment building. This approach lends itself to a more exact analytical interpretation and counters the suggestion that a more hazardous condition could have prevailed. In addition, by measuring fission product plateout on material surfaces during the LOFT experiment and by a knowledge of the influence of fission product to concentration surface ratios, extrapolation appears possible. Research efforts have been started to provide the information necessary to make such an extrapolation.

2.3 Coolant System Piping

2.31 General. A series of nonnuclear blowdown experiments are included in the experimental program. Measurements will be made to ascertain the state of the coolant, expulsion rate, spatial and component temperatures throughout the system. In order to accomplish these objectives, the system will be intentionally designed to withstand twenty blowdowns.

2.32 Blowdown Device. The location of the rupturing device in the experimental system to simulate the often postulated offset shear of a primary coolant pipe will significantly affect the magnitude of fission products released to the containment building. Investigation of fission product transport mechanisms, which are dependent on such things as the amount of coolant remaining in the system and the natural convection phenomena, suggests that the system be designed for locating the rupturing devices on both the inlet and outlet coolant pipes. Although it is desirable to investigate simulation of piping failures at other locations in the system, additional rupturing devices would be economically prohibitive. The blowdown devices have been located as close to the reactor vessel as physically practical to enhance rather than retard the release of fission products.

Preliminary numerical analyses have indicated that it is possible with the proposed blowdown arrangement to discharge the coolant at a rate sufficient to void the core region in approximately 5 to 15 seconds after initiating the rupture from a bottom coolant pipe. This blowdown rate compares favorably with the blowdown time postulated for most pressurized water reactors (Table I). It is recognized that such things as pipe length, coolant flow path design factors, and rupture time could influence the blowdown rate and fission product transport. Because of the large difference in reactor design in this respect, laboratory scale experimental investigations are needed to determine these effects. An experimental program to provide the information needed is under consideration.

TABLE I

SUMMARY OF REACTOR CHARACTERISTICS

REACTOR PARAMETERS	LOFT	YANKEE	SAVANNAH	RAVENSWOOD	SAXTON	INDIAN POINT	SHIPPING PORT
Thermal Power - MW	50	392	69	2030	20	585	231
Primary System Volume	777	1470 [c]	680 [c]	11,930	290	2350 [c]	1235 [c]
Large Break Size-ft^2	1.43	1.43[d]	0.87[d]	---	0.0375	2.05	1.23
Blowdown Time-sec	10	16	30	---	80	60	26
Time for Cladding to Reach Melting Temp-sec [f]	200 [a]	155 [b]	500 [b]	---	180 [b]	180 [b]	Seed 205 [b]
System Pressure-psig	2330	2000	1750	2035	2000	1500	2000
Core Power Density-kW/ℓcore	62	58	21	75	54	76	23
Fuel Power Density-kW/ℓUO_2	256	167	86	198	200	288	79
Average Coolant Temp. °F	540	516	508	569	530	503	523
Core ΔT - °F	31	33	23	50	--	--	--
Reactor Vessel ΔT	--	--	--	---	22	32	30
Average Coolant Temp.-°F	1500	940	960	1720	1230	881	749
Average Cladding Surface Temp. °F	614	530	534	592	565	570	580
Average Heat Flux Btu/hr-ft^2 x 10^{-5}	1.67	0.86	0.62	1.46	1.37	1.28	Seed 0.98
Max. Hot Channel Heat Flux Btu/hr-ft^2 x 10^{-5} [e]	4.0	2.49	1.87	4.74	4.44	5.37	Seed 4.2
Hot Channel Factor ($F_A F_R$) [e]	3.05	2.9	3.0	3.25	3.24	4.2	Seed 4.3
Break Location with Respect to core	Above/ Below	Above/ Below	Below	---	Above	Below	Below

[a] 100 efph
[b] ∞ Operation
[c] One-half of total volume
[d] One-half of total flow area-offset shear break
[e] Does not include hot spot factors
[f] At design hot spot

2.4 Reactor Vessel Design

In selecting a reactor vessel for the loss of coolant experiment, consideration must be given to the vessel hydrodynamic and thermodynamic design to facilitate analytical predications and interpretation of experimental results. Therefore, the vessel has been designed to provide the following: (a) numerous instrument and inspection access ports, (b) remote disassembly of reactor vessel internals, (c) retention of molten fuel and core components within the vessel, and (d) the capability to withstand 20 blowdowns. The reactor vessel is designed for a single-pass cooling system. The coolant flows from the bottom of the vessel, upward through the core, and exits the vessel near the top flange forming a simple single-pass up-flow system.

Transient heat transfer analytical investigations have been conducted to predict the dynamic behavior and physical changes occurring within the reactor during core disassembly and meltdown. The analysis indicates that with the present LOFT core design, a loss of coolant from the reactor primary system would result in significant and wide spread clad melting with the fuel pellets and parts of the core structure subsequently migrating to the vessel bottom head. As a result, the head may be severely damaged to the extent that fuel material could penetrate the vessel. Experimentally, it is desirable to control this variable in order to achieve a better understanding of the phenomena associated with fission product transport. Consequently, a sealed high temperature container has been incorporated into the vessel design to receive fuel and molten structural material in the event melting of the bottom head should occur. The catch basin is designed so that it will not interfere with the heat transfer characteristics of a typical insulated reactor vessel head. Thus, if a vessel melt-through should be experienced, valuable information will be obtained.

2.5 Fission Product Behavior

It is planned to operate the LOFT experiment for a period of 400 hours at 50 MW(t) to obtain near equilibrium concentrations of the radioactive halogens and sufficient fission product decay heat generation rates to ensure heat-up of the core to melting temperatures following a loss of coolant. The I-131 activity after 400 hours of operation is approximately 80 percent of that which would exist at the end of core life.

Concern has been expressed regarding the concentration of fission products, particularly iodine, in LOFT because of the potential influence of concentration on fission product behavior. The state-of-the-art at the present time is not adequate to conclusively determine the effect of iodine concentration on the overall consequences of a loss of coolant accident. A brief discussion of the potential effect of fission product concentration on the program is presented below. A comparison of parameters affecting fission product behavior of several nuclear power plants to LOFT is shown in Table II.

As noted in Table II, if only the radioiodine is considered, assuming 100 percent release to the containment building, the concentration in the LOFT containment vessel is only a factor of approximately 3.2 less than that at the end of core life for the Yankee reactor (assuming no plateout in either case). Therefore, it can be concluded that the direct dose rate as a function of distance from the Yankee reactor would only be a factor of three greater than LOFT provided the retention characteristics of the containment buildings are somewhat the same.

A limited number of tests have been performed at ORNL to determine the chemical state of iodine as a function of total iodine concentration. The tests were performed with a 1 μg release and a 100 μg release of iodine. The concentration per unit volume is not known; however, it was found that when these concentrations were released into air, the fraction of organic iodide formed for the 1 μg release was approximately a factor of 10 greater than for the 100 μg release. This indicates that LOFT could have a greater fraction of organic iodides than Yankee, and that the iodine behavior in LOFT, such as plateout,

TABLE II

RATIO OF PARAMETERS AFFECTING FISSION PRODUCT BEHAVIOR-COMPARISON OF VARIOUS NUCLEAR PLANTS TO LOFT

	Ratios				
	Yankee	Indian Point	Elk River	Big Rock Point	Dresden
Power	7.8	11.7	1.46	4.8	14
Operating time	25	28	30	22	22
Containment volume	2.9	6.1	1.02	3.2	10
Containment surface area	1.8	2.9	1	2	4
I-131 inventory	9.8	13.7	1.9	4	17
Stable I inventory	660	1100	142	220	1,040
I-131 per ft^3	3.2	2.3	1.8	1.16	1.7
I-131 per ft^2	5.5	4.8	1.9	2.0	4.3
Stable I per ft^3	220	187	140	68	106
Stable I per ft^2	364	378	340	110	248
Total I per ft^3	89	71	55.6	26.6	4
Total I per ft^2	149	151	57	25	10

may not necessarily be representative of another power plant. It is concluded that the effects of concentration on the iodine behavior can only be investigated by continuing and expanding the small scale experiments and correlating these results with LOFT data in order to predict the behavior in a particular power plant.

There is also some concern that plateout as a function of concentration is not a linear phenomenon if a material surface becomes saturated with iodine. If this occurs then it is conceivable that a larger fraction of fission products might plateout in LOFT than in Yankee.

Sufficient experimental data is not yet available to predict the plateout behavior for a wide range of fission product concentrations. However, fission product behavior studies are planned to provide the needed experimental data for accurate extrapolation of the LOFT results to other reactors.

As shown in Table II, the total iodine concentration/ft^3 in LOFT containment atmosphere (assuming no plateout) is predicted to be approximately a factor of 89 less than for the Yankee reactor. It also should be noted that within the LOFT facility the concentration will vary from 4×10^{-2} g/ft^3 in the reactor vessel to a minimum of less than 7×10^{-7} g/ft^3 in the containment building. Assuming no plateout, the concentration will vary by at least a factor of 10 inside the containment building over the course of the test. If further stratification in the building is considered, it is conceivable that the concentration could vary by a factor of 10^6 during the course of the experiment.

It is recognized that these arguments do not answer all the questions concerning the effects of iodine concentration; however, they do point out that other factors must be considered. Difficulty in evaluating the gross effects of iodine concentration of fission product behavior suggests that additional laboratory studies should be conducted concurrently with the integrated loss of coolant program. It has been recommended, therefore, that the effects of individual parameters such as iodine concentration, organic concentration, and plateout on the overall fission product behavior be investigated.

3. FACILITY DESIGN CONCEPT

3.1 Containment Building Size and Pressure

Most pressurized water nuclear power plants have three basic provisions for containment of fission products: the cladding material encasing the fuel, the reactor vessel and coolant system, and the reactor containment housing the nuclear system. The third provision is required in the event an accident occurs which results in a rupture of the primary coolant system and subsequent melting of the fuel clad. Since the STEP loss of coolant program purposely destroys the first two provisions for containment of fission products, the containment vessel will be designed, throughly instrumented, and tested to aid in the investigation of pressure, temperature response, and fission product distribution and plateout as a function of time. To ascertain building integrity before each test, to establish leak rates as a function of pressure and temperature, and to allow remote decontamination of the building interior following a planned fission product release, several special provisions not normally found in conventional dry containment structures will be incoporated into the test building.

The inherent fission product retention characteristics of a containment vessel are dependent on such things as the surface areas, penetration design, materials, temperature, pressure, and the fission product concentration in the containment atmosphere. To attempt to mock-up these parameters in a manner representative of all nuclear power plants is not feasible. Therefore, the approach taken has been to size the containment vessel for the initial nuclear system. In order to obtain a pressure in the containment building that is representative of the pressures postulated for the Maximum Credible Accident (MCA) in typical nuclear power plants, the containment building was sized to contain the released coolant from the initial nuclear system with a resultant peak pressure of approximately 20 psi. When considering the MCA for LOFT, which hypothesized the complete blowdown of the primary and secondary systems plus decay heat contributions from the reactor, the building design pressure was established at 40 psig. The design pressure is adequate to contain the additional energy released from a potential metal water reaction using either stainless steel or zirconium clad fuel.

3.2 Penetrations

Containment systems are normally designed to provide a structure with overall leakages not to exceed approximately 0.1 percent of the building free volume per day at design pressures. The principal sources of containment vessel leakage have been reported [3] to occur at (a) penetrations which breach the containment barrier, (b) large equipment doors which depend on seals provided by soft gasket materials, (c) personnel and equipment air locks with multiple seals, (d) openings in the containment shell which communicate directly with the outside atmosphere such as ventilation ducts, and (e) isolation valves which serve as barriers to the escape of the containment atmosphere through piping systems or ducts. Based on the premise that major sources of leakage are from containment penetrations, the design criteria for LOFT has included the requirement to monitor all instrument, electrical, and piping penetrations. The general philosophy on design of the large personnel and equipment access doors has been to preclude leakage by employing double seals with the capability of annulus pressurization to twice the containment building design pressure.

Provision will be made on all electrical cable and piping penetrations whereby the fission product leakage through each penetration can be collected, the penetrations can be pressure and leak tested independently of the containment building, and the gross gas leakage measured when the containment vessel is pressured with either dry air or an air-steam atmosphere. With this arrangement, data can be obtained during the loss of coolant test on the total fission product leakage through a particular type penetration having a known gas leak rate. Thus, the radiological hazards resulting from this leakage can be determined. This arrangement also is expected to reduce the potential radiological exposures during the LOFT test as well as reducing the number of radiological measurements external to the containment building.

Consideration also is being given to the construction of a controlled leak in the containment vessel with which to determine the fission product leak rate as a function of the gas leak rate. This provision is intended to provide information which may be used to evaluate the accuracy of gas leakage measurements.

3.3 Sampling Systems

3.31 General. One of the objectives of the loss of coolant program is to obtain quantitative and qualitative information on fission product behavior within the containment building. Identification of the various gaseous and particulate isotopes and the chemical state and particle size of each are of particular interest in determining the ability of power reactor containment structures to limit fission product dispersion to the environs. In order to obtain this information in the LOFT facility, removable and nonremovable sampling systems are incorporated to enable selective sampling as a function of time during the entire duration of the loss of coolant test. Twelve stations, consisting of removable gaseous, plateout, and particulate sampling devices are strategically located in the building. Samples obtained from each location and removed from the building will be radiochemically analyzed and examined using gamma spectroscopy techniques to establish fission product characteristics at any given time. Representative surface and spatial temperatures also will be obtained concurrently with the programmed sampling. With the development of these new sampling techniques, it must be recognized that research and development efforts are required to determine the adequacy of the design and to establish sampling accuracies. In addition, research and development work is required to develop mechanisms for measuring the chemical state and particle size of the fission products.

3.32 Removable Samples. The removable sample stations are located on the floor, at the 38-foot level adjacent to the vertical walls, and midway between the building centerline and the vertical wall in the dome area. The stations are oriented in three sectors within the building with two additional stations located in the coupling station sector to more adequately sample the area of greatest anticipated leakage. After retrieval of the samples from the building, the samples are loaded into shielding casks and transported by truck to the chemical laboratories. Since the samples will be removed every 30 minutes during the early phases of the test, vehicular access to the underground area is required at all times during the conduct of the test.

3.33 Nonremovable Samples. The nonremovable samples consist of mostly plateout and a few gaseous samples located at relatively close intervals throughout the building and on major machinery surfaces. The building mounted nonremovable samplers also are oriented in the same three sectors specified for the removable sample stations. The sampler devices are designed to obtain integrated samples from the time the test is initiated until the test is terminated. Removal of the sealed samplers is accomplished manually after removal of the experimental assembly and personnel access to the building is attained.

3.4 Missile Shield

To preclude the penetration of the containment vessel and subsequent uncontrolled release of fission products to the atmosphere following an MCA of the LOFT reactor, a missile shield is provided on the interior of the building. This shield consists of a one-foot thick concrete wall attached to the inside of the containment shell from ground level up to the tangent point with the hemispherical dome lined with concrete six inches thick. In addition to the missile protection afforded by the concrete lining for the LOFT reactor, the concrete missile shield design was chosen to represent existing and proposed power reactor dry containment technology. The pressure temperature decay characteristics following the loss of coolant test is expected to closely follow those calculated for most power reactor systems and will provide a representative internal driving force for the release of fission products to the atmosphere.

In addition to the containment vessel proper, missile protection is afforded to other sensitive items of equipment in the test cell area. The shielded viewing window is covered during a destructive test by a heavy hinged steel plate which can be remotely opened and closed. A protective shield is provided for the crane trolley which, when drawn up by the crane hook to the crane bridge rails, will protect the trolley mounted hoist mechanisms from possible missile impingement by the reactor below. The electrical connections to the dolly are made up in large, cylindrical shaped, explosion proof, pressurized junction boxes to ensure the integrity of the experimental instrumentation cabling systems.

All of these protective systems are based on the highly improbable assumption that the explosive discharge of steam and water from the primary piping break, coupled with the attendant reactor temperature excursion, will cause the violent ejection of mechanically attached system components or fragments throughout the containment building.

V. TEST PROGRAM

1. INTRODUCTION

The test program for investigating the consequences of a loss of coolant accident involving a pressurized water-cooled and -moderated power reactor is described in this section. The program will be carried out using a complete nuclear power plant consisting of a 50 MW(t) pressurized water reactor, containment building, and associated auxiliary equipment. The purpose of this program is to conduct tests which will demonstrate the consequences of a loss of coolant accident and provide quantitative information on the behavior of the nuclear plant components, the behavior of fission products under accident conditions, and to determine which phenomena associated with this accident are of principal importance in determining the consequences. Emphasis will be placed on providing information concerning: (a) the fraction of the fission product inventory released from the fuel, (b) the nature and magnitude of the fission products reaching the containment building and the efficiency of inherent removal mechanisms, and (c) the fraction of the fission product inventory which escapes from the containment vessel.

The test program will be preceded by an extensive analytical effort and coordinated with other experimental research programs on various aspects of the loss of coolant experiment. The analytical effort is needed to assist in selecting the conditions for the tests, to predict test results, and to assist in the selection of the type and number of measurements to be made and the instrumentation to be used.

The program consists of five phases: (a) Phase I -- Containment Vessel Pressure and Leak Tests, (b) Phase II -- Coolant Blowdown Tests With a Dummy Core, (c) Phase III -- Low Power Physics Tests and Power Operation, (d) Phase IV -- Loss of Coolant Test With a Radioactive Core, and (e) Phase V -- Facility Cleanup and Post-Test Examination.

The first two phases of the program (Phases I and II) encompass those tests which are required to evaluate the various experimental equipment and instrumentation to be used during the loss of coolant test (Phase IV), to evaluate the safety of performing the loss of coolant test, to provide preliminary data needed to interpret and apply the test results, and to provide data which will assist in improving the analytical models used to predict the consequences of a loss of coolant accident. The first series of tests, identified as Phase I, will consist of pressure and leak tests on the containment building. During these initial tests the total gas leak rate from the containment vessel and through the penetrations will be determined. These data will then be used to reevaluate the fraction of fission products which reach the containment that may escape to the atmosphere.

After the completion of Phase I, the primary coolant system containing a dummy core will be installed in the containment vessel and a series of approximately 20 primary coolant blowdowns will be performed. These blowdown tests are incorporated to investigate the effects of rupture size, rupture location, coolant temperature, and coolant pressure on the reactor and plant components and fission product behavior. During these tests, information will be obtained on gas leak rates through monitored penetrations, the effect of rupture size and location on iodine transport from the reactor vessel and the ability of the containment spray system to terminate the gas or fission product leakage to the atmosphere in the event of an accident. All data obtained during this phase of the experiment (Phase II) also will be used to evaluate the experimental instrumentation to be used in measuring these consequences.

Following the first two phases of the program, the core will be installed and low power physics and engineering measurements made to determine the reactor characteristics. The reactor will then be operated for at least 400 hours at 50 MW(t) to build up the fission product inventory and after-shutdown heat flux. This part of the program is identified as Phase III.

Phase IV of the program will be initiated by simulating a rupture of the primary coolant pipe, allowing the coolant to be expelled from the system and the core temperature to rise. This phase of the program will permit measurements to be made of the fission product behavior during and following core meltdown and of the inherent mechanisms that tend to reduce the fission product leakage to the containment and to the atmosphere. Fission product measurements will be made at selected points throughout the containment building as function of time to provide information on their time-space behavior over a period of approximately five days. In addition, samples will be obtained with which to determine the chemical state and particle size of the fission products.

Phase IV will be terminated by actuating the containment building spray system. The fission products will then be removed from the containment atmosphere by filtration and exhausting through the stack and the internal surfaces of the building and the external surface of the nuclear system will be remotely decontaminated. When the fission products have been sufficiently reduced, the containment building will be opened and the entire nuclear package will be removed to the Hot Shop for post-test examination. This phase of the program is identified as Phase V.

2. SUMMARY OF ANALYTICAL EFFORT TO DATE

Concurrently with the development of the experimental program and with the preliminary design of the reactor and facility, analytical studies have been initiated. The studies have been directed at evaluating existing models and calculational techniques and at performing parametric analysis. The general areas under study have been coolant blowdown, core thermal and physical behavior, and fission product release and dispersal. In view of the state-of-the-art, these studies have necessarily been of a simplified nature. A discussion of the analysis is presented in the LOFT Preliminary Safety Analysis Report (IDO-16981). A brief summary of the work is presented below.

2.1 Primary Coolant Blowdown

A loss of coolant accident originates as a result of failure of some component in the primary coolant system followed by coolant expulsion or blowdown. As a point of departure, estimates of the events occurring during the blowdown, their magnitude and time-scale, are needed to accomplish the following objectives:

(1) Verify the system design,

(2) predict the temperature-time configuration behavior of the LOFT core,

(3) indicate the types and locations of instrumentation desirable for this portion of the LOFT experiment, and

(4) predict effects of varying the system geometry and operating conditions.

It is realized that the more detailed an analysis is the more information it will yield. However, the possibility of achieving a very detailed and accurate analysis of the coolant blowdown phenomena is made somewhat remote by the incompleteness and lack of consistency of the presently available information regarding the proper choice of assumptions and techniques to use for various aspects of the calculations. For this reason, the problem was first attacked by determining some of the gross features of the blowdown, and then proceeding to more refined and detailed analyses.

The first step toward determining the blowdown characteristics was the estimation of the shortest possible blowdown time. Since only a rough estimate was warranted, a rather simplied approach was tried. The variables of major influence on blowdown time are pipe rupture area, system volume, mean vessel-to-containment pressure difference, and mean coolant density. A dimensional analysis of these variables by the Rayleigh method yielded a functional relationship between them. By using the values of the calculated blowdown times and the other parameters listed above for several other water reactors, the constants in the functional relationship were evaluated. The final simplified relationship was found to be:

Blowdown time, $T = \frac{V}{K\,A}$ for $400 \leq \frac{V}{A} \leq 2500$

where V = Coolant volume, ft^3

K = Constant ft/sec

A = Rupture area, ft^2

T = Time in sec

This relation predicts the calculated blowdown time for eight different water reactors to within 13 percent. For LOFT, based on a system volume of 717 ft^3 and the rupture of a 20-inch, schedule 160 inlet coolant pipe near the vessel, the predicted value is nine seconds.

It is evident that the above relation amounts to a correlation of calculated times. Since most of the calculated times were probably based on the same or similar analytical techniques, the use of the correlation for LOFT is roughly equivalent to calculation of the blowdown time by the "standard" techniques used by other reactor designers.

The next step in gaining an insight into the blowdown phenomena consisted of determining the gross state of the coolant and the corresponding mass flow rate from the system, both as a function of time. One of the major factors which contributes to the magnitude of the mass flow rate and, hence, to the time rate of change of in-vessel conditions is the manner in which the coolant in the vessel expands. There is a range of possibilities here, the extremes of which are: (a) homogeneous expansion -- a uniform mixture of steam and water which continuously increases in quality as the blowdown progresses, and (b) separated expansion -- a distinct water level exists in the vessel, so that the fluid entering the outlet pipe is either all water or all steam.

It has been found by experimenters that during relatively slow blowdown transients in systems with small outlets a separated expansion occurs. In a system with a comparatively large break, however, it is generally accepted that homogeneous expansion will be approached.

Since it is evident that the blowdown can be very rapid, calculations to the present time have assumed a homogeneous expansion. With further assumptions that the mixture was in thermal equilibrium and that no heat was added to the fluid, the method of Harris [4] was used to calculate the mass of coolant in the system versus pressure.

Having a mass-pressure relationship, the next problem in arriving at the mass or pressure as a function of time is to determine the mass flow rate. The initial depressurization (subcooled blowdown) of the system from operating pressure down to saturation conditions was estimated to take approximately 30 to 50 msec, during which time the temperature remained constant. This time was calculated from a dynamic force balance on an idealized model of the primary system. Water hammer effects and possible two phase critical flow during this portion of the blowdown were not included. Since the hydraulic forces acting on the core and other components are probably most severe during the subcooled blowdown, a more detailed analysis of this aspect will be made in the future. An attempt was then made to estimate the peak core pressure drop. It appeared from the above analysis that the maximum flow rate through the core during the subcooled blowdown may be approximately four to eight times greater than the flow rate during operation. By assuming that the pressure drop is proportional to the square of the flow rate, it was determined that a Δp in the range of 200 to 400 psi might be expected.

Further analysis of the subcooled blowdown is being made in support of the core design by Phillips Petroleum Company. Babcock and Wilcox are extending this work in a parametric study of reactor blowdown and are including effects of both subcooled and saturated conditions. This work covers the parametric range of break size and location, coolant pressure and temperature, flow resistance in the vessel and piping, and piping length from the vessel to the break. The architect-engineer, Kaiser Engineers, also is carrying out an analysis of the nonnuclear blowdowns to determine the transient thermal stresses existing in the primary system components.

The response of the containment to both nuclear and nonnuclear blowdowns has been investigated by the three companies to support the facility design and to estimate leakage rates for both the test and postulated accident conditions.

2.2 Core Meltdown

The second event of major concern in a loss of coolant accident to which analytical studies have been directed is that of the reactor power response and the transient thermal behavior of the core.

The decrease in reactor power following a rupture of the inlet coolant pipe to the reactor vessel depends on the rate of moderator loss and the increase in the neutron leakage probability. Both of these factors affect the fission rate depending on the rate of coolant expulsion and pressure decay within the core. After a relatively short interval of time, the production rate of delayed neutrons becomes insignificant and the heat production within the fuel pins is then determined by the shutdown decay beta and gamma production. The rate of change of core neutron multiplication and, consequently, fission power has not been investigated to date, but it is concluded from a survey of existing literature that the drop in power is prompt and that the integrated fission energy after the break occurs has little effect on the subsequent heat-up of the fuel.

In the event of a break in the exit line, ie, in an elevated portion of the system, the fission power production could be significant for some time following the initiation of rupture. With certain types of control rod drive mechanisms such as magnetic jacks, the entire rod and extension assembly is free to move after a scram. Thus, if a break occurred and the reactor were scrammed due to loss of pressure, the rod could conceivably "float" during the time that high coolant velocities exist in the core. Depending on the excess reactivity available in the control rods, the reactor could experience a significantly positive reactivity insertion and subsequent power transient. This positive reactivity addition is, of course, negated to some degree by the loss of moderator and the increase in neutron escape probability. This effect has not been thoroughly investigated to date.

Since high coolant velocities and, hence, improved heat transfer rates exist during the blowdown period, the temperature of the fuel and cladding will drop sharply to a new reference or initial temperature before core heat-up begins. As the conductance between the UO_2 pellets and the cladding is considerably smaller than the conductance between the cladding and the coolant, the controlling factor establishing the value of the new temperature reference level is the rate of heat loss from the pellet. The effect of the radial gap between the pellet and the cladding on thermal conductance has been investigated experimentally, and for the gap size used in LOFT the effective conductivity is expected to be 0.5 Btu/hr-ft-°F. This conductivity also is strongly dependent on the degree of thermal expansion and thermal cracking in the UO_2 which tends to reduce the thermal barrier associated with the gap. Due to the uncertainties in predicting a reasonable conductance, the fuel and cladding temperatures were assumed to drop to the value of the ambient operating coolant temperature (≈540°F). This assumption is in error in that the time to reach clad or fuel melting is thus over-estimated.

The rate of temperature rise of the fuel is dependent on the magnitude of the heat removal mechanisms. Calculations indicate that radiation heat loss is the major heat removal mechanism, and the rate of the temperature increase is essentially that for an insulated condition until the temperature is several hundred degrees above the clad melting point. This is true even for cores of significantly reduced radial size (20.5- to 12-inch radius). The time to reach clad melting under these conditions is estimated to be approximately 730 seconds. Although the effect of several simultaneous heat removal mechanisms could not be described explicitly with the code used, it has been estimated that natural convection heat transfer to flowing steam may remove from 3 to 20 percent of the total decay heat generated. The rate of temperature rise would, therefore, decrease slightly causing the time to subsequently reach clad or fuel melting to be increased from 730 seconds to a maximum of 900 seconds.

Axial conduction does not appear to be a major heat removal mechanism during the core meltdown. Although heat loss by this mechanism is probably negligible, axial conduction may tend to flatten the temperature profile both in the UO_2 and in the cladding. The cladding in a particular region of the core would therefore tend to be at a higher average axial temperature at the onset of melting. This phenomenon could reduce solidification of the stainless steel as it flows downward along the pin. The effective contact conductance between internal surfaces of the stacked column of pellets appears to be the controlling variable in predicting axial temperature profiles in the fuel.

To evaluate the effects of radiation heat transfer from a fuel pin to the surrounding atmosphere, it was assumed that saturated steam was available to the core. The calculations indicated that the radiation heat losses in the lower section of the core to the steam were of the same magnitude as those from convective losses. However, the overall heat loss from the core by this mechanism is small, since the steam, in rising through the core, radiates heat back to the colder surfaces near the top of the core transferring heat back to the fuel. Thus, this mechanism primarily affects the temperature distribution in the core and the overall heat loss is negligible.

On the basis of the present analysis, it can reasonably be predicted that the onset of clad melting will occur within 1000 seconds after blowdown. It also can be reasonably predicted that in excess of 75 percent the cladding will melt. The fuel pellets will no longer remain free standing due to the ceramic nature of the material and to the thermal cracking. The core is expected to lose its integrity and to move to lower regions of the vessel.

If, contrary to expectations, the fuel pellets remain in a free-standing configuration, some fuel could be expected to reach melting temperature at a time greater than 3000 seconds.

As the fuel cladding melts, it is assumed to flow down along the remaining pin surfaces to the fuel element restraining grids. As these grids are at a much lower temperature than the clad, solidification of the cladding will occur. These grids will then act as a barrier to the movement of fuel from its normal position to some position near the lower environs of the vessel. The temperature-time history of the UO_2 stainless steel mixture as it rests upon these grids is difficult to estimate. The effective conductivity of this mixture depends on the physical state of the UO_2, the amount of mixing or stratification of the two constituents, and the internal surface conductance from the UO_2 to the stainless steel. The radiation heat loss from the mixture must also be a gross estimate since neither a true emissivity for the mixture nor the effective temperature of the surfaces to which the mixture radiates can be rigorously determined.

A wide range of values for the above parameters have been studied. It can be concluded that the fuel will eventually melt through the restraining grids and fall either to the core support plate or to the vessel bottom head.

If the fuel does come to rest on the support plate, this plate is of sufficient thickness to act as a very effective insulator. The UO_2-stainless steel mixture will reach temperatures in excess of 3000°F and a new heat transfer phenomenon will then become available. As the temperature rises, the various constituents of the stainless steel will begin to "boil off" from the mixture and, thereby, provide a large heat sink in the form of the heat of vaporization and from improved boiling heat transfer in the mixture. Investigators to date have not been successful in estimating the effect of this phenomenon. Sufficient energy exists in the fuel to melt through the support plate and it appears that the possibility of severe stratification of the mixture will allow considerable heat to be transferred to the support plate, thereby eventually causing it to fail. There is also a distinct possibility that the temperatures within the fuel could reach the UO_2 melting point during this time.

As the design of the vessel and retention chamber is not yet firm, the transient thermal analysis has been of necessity rather general in nature and based upon features which may not exist in the final design. The analysis has shown, however, that the reactor vessel bottom head may be severely damaged due to local melting and may, in fact, be breached. Local melting of UO_2 also appears to be a distinct possibility even before a vessel breach would occur. This conclusion again depends on the heat sink provided by the boiling of stainless steel and the increased heat transfer capability of boiling steel. The core size may have an effect on the severity of damage to the head but does not appear to be a controlling factor as far as the maximum attainable temperatures in the fuel are concerned.

In general, it may be concluded that with the present LOFT core design, a loss of coolant from the reactor primary system would result in significant and widespread clad melting. Fuel pellets and parts of the core structure can be expected to subsequently migrate to the vessel bottom head. As a result, the head may be severely damaged and temperatures within the fuel could conceivably attain the melting point of UO_2.

2.21 Projected Effort. In the foregoing discussions of the initial phase of the LOFT transient thermal analysis, several areas have been noted where additional analytical effort must be initiated or the analysis extended, and where basic knowledge of material properties and heat transfer phenomena must be acquired. These particular areas are listed below.

(1) Further investigate the heat loss from fuel and clad during blowdown to better establish a temperature profile within the pins and the reference temperature for core heat-up calculations.

(2) Conduct further analyses to determine the core power behavior during blowdown for breaks above and below the core in order to establish the energy stored in the fuel during the blowdown period.

(3) Develop a two-dimensional (R-Z) model to handle transient heat transfer within the core and support structures and to account for all possible heat loss mechanisms.

(4) Determine the actual decay heat distribution within the core by taking into account the gamma-ray leakage from regions internal to the core and from the core as a whole.

(5) Attempt to derive a transient heat transfer model which will describe the fuel pins explicitly in order to effectively describe the radiation heat transfer from pin-to-pin.

(6) Determine an effective conductivity for the UO_2 pellet and gap configuration both radially and axially.

(7) Complete an analysis of the temperatures and stresses of the vessel and structure for the core meltdown situation.

(8) Develop an analytical technique for describing transient heat transfer in the UO_2-stainless steel mixture at temperatures from operating condition to the stainless steel boiling point.

(9) Determine the high temperature properties of the core materials which are of importance in describing the transient thermal behavior of the core. These properties include emissivities, specific heats, and thermal conductivities from approximately 1000°F to the melting point of the UO_2.

2.3 Fission Product Behavior Model

Although a considerable quantity of experimental data are available regarding the fission product release from UO_2 as a function of temperature, burnup, and atmosphere, essentially no information is available regarding transport mechanisms and plateout behavior inside the reactor vessel and containment building. The development of calculational models for describing these mechanisms with any assurance is not possible. It should be pointed out, however, that experiments are planned as part of the AEC's nuclear safety program to investigate these mechanisms and as data become available from these research programs, a more realistic model will be developed.

Calculations have been made using the assumptions contained in guide TID-14844 [5]. In addition, diffusion calculations are planned to evaluate the influence of the principal variables in fission product transport.

3. PHASE I -- CONTAINMENT VESSEL PRESSURE AND LEAK TESTS

3.1 General

This phase of the experimental program is directed at investigating the containment pressure and leak characteristics. A series of tests will be performed to ensure that the air leak-rate from the vessel meets the design specifications and to determine the leak rate as a function of pressure. These tests are expected to provide a high degree of assurance that the containment vessel will withstand the pressure associated with primary coolant blowdown as well as providing information on the leak rate as a function of pressure. This information is needed for a correlation of the air leak rate with the leak rate of fission products in air-steam atmospheres to be accomplished in Phase II, for a final assessment of the radiological hazard that may prevail during the loss of coolant test (Phase IV) and for interpretation of the final results of the loss of coolant test.

The first series of leak tests, which are normally considered as acceptance tests, will be performed to investigate the pressure capabilities of and the leak rate from the containment building. Although the tests will be performed by the construction contractor, the tentative procedure and the data to be obtained are presented herein for information. After the tests have successfully demonstrated that the pressure capabilities and leak rates are within the design specifications, tests will be performed as part of the experimental program to determine the leakage characteristics of the containment. These tests will include:

(1) Determining the total leak rate from the containment building as a function of pressure.

(2) Determining the leak rate through building penetrations as a function of pressure.

(3) Calibrating and adjusting controlled leak.

(4) Determining the ability of the operational and test instrumentation to function properly under pressure conditions.

(5) Evaluating the existing techniques as well as the techniques that may be developed in the future for measuring containment leakage.

Several of the leak tests will be accompanied by a release of radioactive tracers to gain early information on the fission product leakage from the containment and the filtration effect of the penetrations. All leak tests in this series will employ dry air at ambient temperature. A more detailed discussion of the leak tests to be performed and the data to be obtained are discussed below.

3.2 Tests Performed by the Construction Contractor

3.21 Locate and Repair Gross Leaks. With the containment vessel pressure at approximately 10 psig, a local leak test will be performed. The test will consist of applying a soap solution over or around all shielded penetrations and accessible welds and detecting the leaks by the formation of bubbles. It is possible by this technique to detect leaks of the order of 10^{-2} cm^3/hr (0.003 percent of the free volume per day). After the leak test has been completed, the pressure will be reduced to atmospheric pressure and the leaks located will be repaired.

3.22 Pneumatic Pressure Test. The pneumatic pressure test will consist of pressurizing the containment building to 48 psig (120 percent of design pressure) and maintaining this pressure for approximately one hour. Strain gauge measurements on the containment vessel will be made during the test.

3.23 Design Pressure Tests. Upon completion of the facility but prior to acceptance, the containment vessel will be tested at the design pressure of 40 psig. The total leak rate including the leakage through monitored penetrations will be determined. Should the total leakage exceed 0.2 percent of the free volume per day, the pressure will be reduced and the major leaks located and repaired. The monitored (double) penetrations are designed whereby they can be pressured and leak tested independent of the containment building. Thus, an accurate knowledge of the location of all leaks as well as the leak rate is expected.

The total leak rate during this test and the ensuing tests will be measured by the reference system method. This method generally involves measuring the change in pressure of the containment air relative to that of a hermetically sealed and pressurized reference system located within the containment vessel. The difference in the two pressures, provided the temperatures are equal, is proportional to the leak rate. A description of this method and the special experimental requirements and limitations are discussed in Section VII-3. Consideration also is being given to the feasibility of adding a known quantity of inert gas to the containment building during some of the leak tests to assist in determining the leakage through the penetrations. An investigation is currently being made to determine the possible types of gas and the detection techniques which might be employed.

3.3 Tests Performed as Part of the Test Program

3.31 Leak Tests at Several Pressures. This portion of the Phase I leak tests consists of determining the total gas leakage from the containment vessel at the pressure expected during the loss of coolant in Phase IV, ie, at 24 psig and at several other pressures less than 24 psig. This information is needed to predict with some assurance the fission product leakage to the surrounding environment and the subsequent radiological hazards that may prevail during conduct of Phase IV of the program. It also will provide some information on the reliability with which the leak rate at high pressure can be predicted from measurements made at low pressure. The monitored penetrations will contain fission product filters (May Pack), and possibly a cold trap to remove the fission products from the gas stream. Not only will this approach greatly reduce the fission product leakage to the environment, it also will provide a relatively accurate measure of the fission product leakage from the containment building during the five-day test period. With this data a prediction can be made of the hazards that would have prevailed during any desired meteorological condition.

The containment building also will contain a controlled leak of known size located near the coupling station. The gas leak rate through this controlled leak will be determined as a function of pressure up to 24 psig. The fission product leakage through this controlled leak will be captured by filters and analyses made to provide information on the fission product leakage through an orifice of know geometry, size, and air leak rates. Since this orifice will offer a minimum of resistance to fission product leakage, information can be obtained to scope the minimum fission product filtration or fractionation that could be expected in passing through a leakage path.

During the pressure tests, all reactor control, reactor process, and test instrumentation that has been installed and checked out prior to the initiation the test will be reexamined to determine their operational reliability under pressure conditions. In addition, other techniques for measuring the containment leakage, such as the absolute pressure drop and the makeup volume methods, will be evaluated. A description of these methods is discussed in Section VII-3.

3.32 Pressure and Leak Tests Involving Radioactive Tracers. Trace quantities of I-130 and Kr-85 will be released to the containment vessel during a pressure test in an attempt to determine the filtering effect of the controlled and monitored leakage paths for radioiodine and to determine the general location of the uncontrolled leakage paths. In addition, these tests are expected to provide some early data on the iodine retention qualities of the containment vessel wall and equipment surfaces.

The filtering effect of the leakage paths will be determined by measuring the radioiodine concentration inside the building and in the penetration filters. The internal air will be continuously circulated throughout the test to provide a homogeneous distribution of the radioactive tracers.

3.33 Leak Tests During Other Phases of the Experimental Program. A leak test at 24 psig will be performed prior to each coolant blowdown test not involving a release of radioactive materials. Prior to the test involving a release of radioactive materials, in particular the Phase IV loss of coolant test, the controlled leak will be calibrated and adjusted to give a total leakage rate, through controlled plus uncontrolled leaks, of 0.2 percent of the free volume per day.

4. PHASE II -- LOSS OF COOLANT TEST WITH A DUMMY CORE

4.1 Scope and Objectives

Phase II of the experimental program consists of a series of preliminary coolant blowdown tests to: (a) investigate the effects of the rupture size, rupture location, coolant temperature, and coolant pressure on the response of a nuclear plant, (b) determine the effects of rupture size and location on the transport of fission products to the containment building, and (c) evaluate the reliability and effectiveness of the containment building to retain fission products and the spray system to reduce the building pressure and, thus, to terminate the fission product leakage from the containment vessel. The tests will provide an opportunity to evaluate the operational reliability of the test instrumentation and techniques for initiating and simulating the rupture under test conditions. In addition, these preliminary blowdown tests are needed to assure that Phase IV of the program can be performed safely and to provide some of the data needed to evaluate and apply the results of the Phase IV loss of coolant test.

Coolant blowdown through 4-, 10-, and 18-inch ID openings will be investigated as a function of primary coolant temperature (450 to 600°F) and pressure (1200 to 2500 psig). It should be recognized that further analytical studies may indicate some minor changes in these parameters are needed. Ruptures will be simulated in both inlet and outlet pipes as near as practical to the reactor vessel. Following these tests, additional blowdown tests will be carried out in which trace quantities of I-130 and Kr-85 will be released inside the reactor vessel. This experiment is intended to provide information regarding the rupture location that will result in the maximum concentration of I-130 to be transported to the containment building. This data will then be used in selecting the rupture location for the Phase IV loss of coolant test. The fission product sampling system, pressure reduction spray system, and decontamination system also will be evaluated during these tests.

This phase of the program 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 restrictions that will prevail during the actual core meltdown test. All but the last nonnuclear blowdown test will utilize a core support structure which is designed to withstand the hydrodynamic forces and rarefaction waves associated with rapid depressurization of the system. The last nonnuclear blowdown test will employ a core support structure of conventional design in accordance with engineering practices established in the nuclear industry. The core and core support structures will be sufficiently instrumented to determine the stresses on fuel pins, control rods, and grid plates, and the pressure drop across the core during blowdown.

The control rods at the time of each rupture will be located at the critical position expected during the actual core meltdown test. The movement of rods resulting from the primary coolant blowdown will then be measured. The rods will be manually scrammed after the blowdown. This part of the test will supply information on the probability of a reactivity accident occuring during primary coolant blowdown.

For all nonnuclear blowdowns it is tentatively planned to turn off the primary coolant pumps at the time the rupture is initiated to preclude pump damage.

4.2 Test Procedure

4.21 Rupture at 300°F and 1000 psig. The first blowdown test will involve a four-inch ID rupture on the outlet line while the coolant is at 300°F and 1000 psig. This test is expected to provide an assessment of the coolant blowdown characteristics and their effects on the nuclear system and containment vessel. Following this test and all subsequent tests the primary coolant piping, dummy core, control rods and drives, and the containment vessel will be visually inspected for damage.

The data to be obtained during this and all subsequent blowdown tests will include:

(1) Pressure and temperature of the containment environment and reactor vessel as a function of time,

(2) strain on the containment vessel and the temperature gradient through the vessel walls,

(3) strain on the primary coolant piping, reactor vessel walls, fuel pins, control rods, thermal shields, core barrel and core grid plates, coolant blowdown rate,

(4) state of the coolant at the rupture opening and inside the reactor vessel during blowdown,

(5) the amount of coolant remaining in the primary coolant system,

(6) control rod motion caused by blowdown,

(7) pressure drop across the core vs time, and

(8) containment vessel leakage rate as a function of time.

4.22 Blowdown Tests at 540°F and 2330 psig. This test series will consist of blowdown through 4-, 10-, and 18-inch ID rupture openings on both the inlet and outlet pipes while the primary coolant is at 540°F and 2330 psig (LOFT operating conditions). The tests to be carried out will be as follows:

(1) 4 inch rupture on outlet pipe,

(2) 10 inch rupture on outlet pipe,

(3) 16 inch rupture on outlet pipe,

(4) 4 inch rupture on inlet pipe,

(5) 10 inch rupture on inlet pipe, and

(6) 18 inch rupture on inlet pipe.

These tests are expected to provide quantitative data on the blowdown rates, pressure transients, etc, that is needed to apply the test results to other nuclear systems. In addition, the data will be used to evaluate the analytical models for predicting the blowdown phenomena as a function of rupture size and location as well as identifying some of the important parameters affecting coolant blowdown, core meltdown, and fission product behavior.

During some of these tests, the samples of the containment atmosphere will be collected and analyzed for organic impurities and other impurities that may affect the chemical state of iodine. This information will assist in evaluating the iodine behavior during the Phase IV loss of coolant test.

4.23 Blowdown Rates vs Temperature and Pressure. A limited number of coolant blowdowns will be performed to investigate the effects of coolant temperature and pressure on the blowdown rate and subsequent containment vessel pressure. These blowdowns will involve simulating an 18 inch rupture on the primary coolant outlet line. The primary coolant temperatures and pressures to be investigated are:

(1) Temperature 450°F, pressure 2330 psig,

(2) temperature 600°F, pressure 2330 psig,

(3) temperature 540°F, pressure 1700 psig,

(4) temperature 540°F, pressure 2500 psig, and

(5) temperature 600°F, pressure 2500 psig.

The data obtained from these tests will assist in determining the parameters affecting the coolant blowdown rate and evaluating the analytical models for predicting blowdown rates. The selected temperatures and pressures encompass the normal operating conditions for most pressurized water nuclear power plants, thus the data obtained will be applicable to other nuclear plants.

During the test at LOFT operating conditions, measurements will be made to determine the total leakage rate from the containment vessel as a function of pressure. This information is needed to determine the leak rate that can be expected during the Phase IV loss of coolant test. By comparing these data with those obtained during tests involving air, the effects of elevated temperature and an air-stream atmosphere on containment leakage can be established. Techniques for measuring the leakage during transient conditions are being investigated.

4.24 Containment Spray System Test. Immediately following the test involving an 18 inch ID rupture on the outlet line, the pressure reduction spray system will be turned on and the time required to reduce the containment building pressure to atmospheric pressure will be determined. The primary coolant at the time of the rupture will be at 540°F and 2330 psig.

The specific purpose of this experiment is to verify that the spray system meets the design requirements. However, the data obtained will assist in evaluating the effectiveness of engineered safeguards of this type for reducing the containment leakage.

The data to be obtained will include the spray water flow rate and temperature in addition to the data identified in Section V-4.

4.25 Fission Product Transport Studies. The nonnuclear blowdown test series will be terminated with three blowdown tests accompanied by the release of trace quantities of I-130 and Kr-85.

The first two blowdowns will involve 18 inch ruptures on the inlet and outlet coolant pipes for the purpose of:

(1) Determining the rupture location that provides the maximum transport of fission products to the containment vessel,

(2) Determining the space-time history for the transport of iodine and krypton to the containment vessel, and

(3) Determining the plateout behavior of iodine in the reactor vessel, primary coolant piping, and containment vessel.

These tests also will provide an opportunity to evaluate the effectiveness of the filtering system for the removal of iodine from the containment atmosphere as well as the effectiveness of the decontamination methods for removing iodine from the containment vessel walls. The fission product sampling techniques to be used during the Phase IV loss of coolant test also will be evaluated under actual test conditions.

During each test, measurements will be made to determine:

(1) The iodine plateout on the containment building walls as a function of time,

(2) The iodine and krypton concentration in the containment atmosphere as a function of time,

(3) The iodine concentration in the water collection sump,

(4) The iodine and krypton concentration as a function of distance from the containment vessel, and

(5) All measurements identified in Section V-5.21.

The capsules containing the I-130 and Kr-85 will be located near the geometric center of the dummy core to simulate an actual fuel element rupture. The isotopes will be released at a time after pipe rupture that is representative of the time predicted for the start of clad meltdown. The quantity of isotopes to be released, type of containers to be used, and the method of initiating the release are presently being investigated.

After the concentration of iodine and krypton in the containment building has reached a maximum and the desired data have been collected, the containment air will be exhausted to the atmosphere through the filtering system. Iodine samples will be obtained both upstream and downstream of the filters to determine their effectiveness for iodine removal.

The third blowdown test involving I-130 and Kr-85 release will be performed to evaluate the effectiveness of the pressure reduction spray system for iodine removal. The test will be performed in the same manner discussed above, except that the spray system will be actuated when the iodine in the containment vessel has reached a maximum. The sprays will operate for 30 minute intervals and between each successive interval, the iodine concentration in the containment air, on the surface of the building walls, and in the radioactive collection tank will be determined. This test will continue until the iodine in the atmosphere has either been removed or an equilibrium condition has been reached.

Following each blowdown test involving radioactive tracers, the containment vessel will be remotely decontaminated to determine the decontamination effectiveness that can be expected following core meltdown.

5. PHASE III -- CRITICAL TESTS AND POWER OPERATION

5.1 General

Phase III of the experimental program includes all low power and high power tests that are required to provide a thorough understanding of the reactor nuclear, hydraulic, and heat transfer parameters. An understanding of these parameters is essential for the interpretation and analysis of the experimental results and for the development of mathematical models for applying the data to other reactor systems.

Following the last coolant blowdown test in Phase II, the dummy core will be removed along with the core support structures and instrument and shock absorber support plate. The results obtained from the nonnuclear blowdown tests will determine the type of core support structure to be employed in the final loss of coolant test. The core support structure and the UO_2 pin core will then be installed in the reactor vessel.

The initial critical experiment will be aimed at verification of physics measurements obtained from the core during precursor critical testing. Additional physics measurements will be made with the nulcear system at various operating conditions to establish temperature, pressure, and density coefficients with and without coolant flow. These tests will consist of:

(1) Determining the initial critical rod positions for both hot and cold conditions,

(2) temperature coefficient measurements as a function of temperature,

(3) determining flux distribution in the core, and

(4) check-out of all in-core test and control instrumentation.

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

5.2 Power Operation

The reactor will be operated at 50 MW(t) for a period of 400 hours to produce a fission product inventory and a shutdown decay heat flux approximating that of infinite operation. At the end of the power run, all radioiodine isotopes will be at essentially an equilibrium condition except I-131, which will be at 76 percent of equilibrium. The fission product inventories as a function of operating time for several selected isotopes are presented in Table III.

The primary coolant during power operation will be 540°F (average) and 2330 psig.

5.3 Power Tests

The magnitude and distribution of the fission products within the core are required in order to accurately determine the fission products released during the loss of coolant test. The parameters will be determined by measuring the radial and axial thermal neutron flux distribution in the core at the beginning of the power run and every 100 hours throughout the run. A final measurement will be made just prior to the loss of coolant test.

In addition to the flux measurements, several measurements will be made throughout the power run to assist in evaluating the core meltdown model and to assure safe operation of the plant. These measurements will include:

(1) Fuel and clad temperatures,

(2) reactor power,

(3) critical rod positions,

(4) coolant temperatures and pressures,

(5) fission gas pressures, and

(6) other miscellaneous data of importance to the safe operation of the reactor.

6. PHASE IV -- LOSS OF COOLANT TEST WITH A RADIOACTIVE CORE

6.1 General

This phase of the experimental program consists of modeling a loss of coolant accident. The major objectives of this experiment are to evaluate the consequences and to obtain quantiative data pertaining to coolant blowdown, core meltdown, and fission product behavior under accident conditions. This information is of importance to the nuclear industry and the AEC in the establishment of siting requirements and in safety assessment.

TABLE III

FISSION PRODUCT INVENTORY (CURIES)
FOLLOWING 100, 400, 800, AND 1600 HOURS OF OPERATION

Isotope	Half Life	Inventory at Shutdown 100 Hrs	400 Hrs	800 Hrs	1600 Hrs
Bromine					
82-85		1.16×10^6	1.16×10^6	1.16×10^6	1.16×10^6
87	54.5 sec	1.09×10^6	1.09×10^6	1.09×10^6	1.09×10^6
88	16.3 sec	1.27×10^6	1.27×10^6	1.27×10^6	1.27×10^6
89	4.4 sec	2.02×10^6	2.02×10^6	2.02×10^6	2.02×10^6
90	1.6 sec	2.20×10^6	2.20×10^6	2.20×10^6	2.20×10^6
Total Br		7.74×10^6	7.74×10^6	7.74×10^6	7.74×10^6
Krypton					
83	114 sec	2.28×10^5	2.28×10^5	2.28×10^5	2.28×10^5
85 m	4.4 hrs	5.32×10^5	5.32×10^5	5.32×10^5	5.32×10^5
85	10.6 yrs	8.66×10^1	3.64×10^2	7.33×10^2	1.47×10^3
87-95		9.45×10^6	9.45×10^6	9.45×10^6	9.45×10^6
Total Kr		1.02×10^7	1.02×10^7	1.02×10^7	1.02×10^7
Strontium					
89	50.5 days	1.16×10^5	4.30×10^5	7.72×10^5	1.26×10^6
90	28.0 yrs	7.15×10^2	2.20×10^3	5.05×10^3	1.14×10^4
91-97		1.21×10^7	1.21×10^7	1.21×10^7	1.21×10^7
Total Sr		1.22×10^7	1.25×10^7	1.29×10^7	1.34×10^7
Zirconium					
95	65 days	1.18×10^5	4.43×10^5	8.14×10^5	1.38×10^6
Ruthenium					
106	1.0 yrs	1.30×10^3	5.14×10^3	1.01×10^4	1.96×10^4
Tellurium					
132	77.0 hrs	1.22×10^6	2.01×10^6	2.06×10^6	2.06×10^6

TABLE III (Cont.)

Isotope	Half Life	Inventory at Shutdown 100 Hrs	400 Hrs	800 Hrs	1600 Hrs
Iodine					
131	8.0 days	3.54×10^5	9.98×10^5	1.24×10^6	1.32×10^6
132	2.3 hrs	1.20×10^6	2.01×10^6	2.06×10^6	2.06×10^6
133	20.8 hrs	2.92×10^6	3.03×10^6	3.03×10^6	3.03×10^6
134	52.5 min	3.38×10^6	3.38×10^6	3.38×10^6	3.38×10^6
135	6.7 hrs	2.68×10^6	2.68×10^6	2.68×10^6	2.68×10^6
136-139		5.77×10^6	5.80×10^6	5.80×10^6	5.80×10^6
Total I		1.63×10^7	1.78×10^7	1.82×10^7	1.83×10^7
Xenon					
133	5.3 days	9.44×10^5	2.62×10^6	2.98×10^6	3.03×10^6
135 m	15.3 min	8.43×10^5	8.43×10^5	8.43×10^5	8.43×10^5
135	9.2 hrs	2.75×10^6	2.76×10^6	2.76×10^6	2.76×10^6
137-142		9.82×10^6	9.82×10^6	9.82×10^6	9.82×10^6
Total Xe		1.44×10^7	1.60×10^7	1.64×10^7	1.65×10^7
Barium					
140	12.8 days	5.63×10^5	1.66×10^6	2.33×10^6	2.71×10^6
Praseodymium					
143	13.7 days	2.97×10^5	1.37×10^6	2.09×10^6	2.53×10^6
Cerium					
144	290 days	2.70×10^4	1.06×10^5	2.09×10^5	4.01×10^5
Total Fission Product Activity Including Above Isotopes		1.91×10^8	2.04×10^8	2.10×10^8	2.16×10^8

In general, the areas to be investigated are:

(1) The containment leakage characteristics under the actual loss of coolant conditions,

(2) the coolant blowdown characteristics,

(3) the fraction of the core (fuel and clad) that can be expected to melt and the physical behavior of the core,

(4) the identity and fraction of fission products released from the fuel,

(5) the identity and fraction of fission products that reach the containment vessel,

(6) the identity and fraction of fission products that escape from the containment vessel,

(7) the natural or inherent factors that tend to remove fission products from the containment atmosphere and what reduction factors can be obtained, and

(8) the dispersion of fission products that escape from the building under known meteorological conditions.

This phase of the experiment is separated into four parts: (a) coolant blowdown, (b) core disassembly and meltdown, (c) fission product behavior inside the containment vessel, and (d) fission product dispersion external to the containment vessel. The purpose, measurements to be made, and the general plant conditions for each part are discussed below.

6.2 Coolant Blowdown

This part of the experiment will involve simulating a rupture of the primary coolant pipe and permitting the coolant to be expelled from the system. The test will be identical to that proposed in Section V-3, except that a single rupture will be involved and the plant will be at the normal operating conditions (2330 psig and 540°F).

The specific purpose of the experiment is to rapidly bare the core of coolant in a manner representative of a severe pipe rupture and to establish the containment environmental conditions that are representative of those following a loss of coolant accident. Sufficient data also will be obtained to evaluate the effects of core decay heat on the mass flow rate, the effects of decay heat on the containment pressure, fission product transport to the containment vessel, and fission product leakage from the containment.

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

(1) Reactor power;

(2) primary coolant flow, temperature, and pressure inside the coolant system and on each side of the rupture opening;

(3) coolant mass flow rate as a function of time and the total time to expel the coolant;

(4) steam velocity and temperature near the core and at the rupture opening as a function of time;

(5) temperature gradient across the containment vessel walls as a function of time;

(6) temperature of the containment air and equipment inside the containment vessel as a function of time;

(7) containment vessel pressure, temperature, strain, and leak rate as a function of time;

(8) strain on fuel rods, control rods, reactor vessel, and piping;

(9) pressure drop across the core;

(10) temperature of the fuel rod (clad and fuel), core structural materials, control rods, reactor vessel, and the remaining coolant as a function of time;

(11) fission gas pressure as a function of time; and

(12) relative humidity as a function of time.

The primary coolant blowdown will be initiated while the reactor is operating at 50 MW(t) and the primary coolant is at 2330 psig and 540°F. The size of the rupture will be 18 inches in diameter; however, the location will be determined from theoretical blowdown analyses and the results of the preliminary blowdown tests involving radioiodine releases.

Immediately prior to the blowdown, the automatic control rod scrams will be bypassed to avoid an automatic scram during blowdown. During the blowdown the rods will be held at the established position until the power level has been reduced approximately one decade by the void and temperature coefficients. The reactor will then be manually scrammed. This technique of shutting the reactor down is representative of the stuck rods case which is postulated to provide a minimum reduction in the fuel and clad temperature during blowdown.

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

6.3 Core Meltdown

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

(1) Fuel, clad, core support, and reactor vessel temperature as a function of time,

(2) fission gas pressure as a function of time,

(3) velocity and temperature of coolant through the core as a function of time, and

(4) extent of core damage and configuration existing after the test has been terminated (from post-test examination).

6.4 Fission Product Behavior

The principal uncertainty in safety analysis and site selection is the uncertainty involved in predicting the fission product behavior. The general practice in safety assessment has been to use the fission product model suggested in TID-14844 [5]. This model suggests that 100 percent of the noble gases, 50 percent of the halogens, and 1 percent of the solids are released from the fuel during core meltdown and that all those released except for 50 percent of the iodine remain available for leakage to the environment. Although the time required to transport fission product from the reactor vessel to the containment building is not specified, the hazards analyst generally assumes that it occurs instantly at the time of the system rupture.

It is the intent of this phase of the experimental program to obtain quantitative data related to the validity of these assumptions. These data will provide quantitative information under actual accident conditions pertaining to: (a) the fission product release from the fuel, (b) the inherent mechanisms that tend to absorb fission products such as the reactor vessel wall, containment building wall, machinery surfaces, and steam, (c) the quantity and type of fission products that are transported to the containment building, the transport mechanism, and the time involved, and (d) the quantity and type of fission products that escape through penetrations. Data also will be obtained to assist in explaining the behavior of fission products, in particular iodine. For example, measurements will be made to determine the chemical state and particle size of iodine, temperature of the various plateout surfaces, plateout as a function of time and concentration, and an analysis of the containment and reactor vessel atmosphere to determine the concentration of oxygen, hydrogen, and organic impurities which may affect the quantity of fission product release and the chemical state of iodine.

In order to obtain the above information a radiological sampling system has been included in the facility design. This system will provide samples of the plateout on wall surfaces, gas and iodine activity in the containment atmosphere, particulate activity, chemical state of iodine, and gross fission product particle size at 12 selected locations inside the containment building as a function of time. In addition, plateout samples will be taken at a number of strategic locations in the building just prior to the termination of the test to determine the total plateout on the machinery and wall surfaces.

A discussion of the samples to be taken, the frequency of sampling, and the data to be obtained are presented below. It should be noted that the techniques discussed are only concepts of the actual sampling devices. The devices to be used and the techniques for removing the samples from the building are being designed by the Architect Engineer.

6.41 Plateout Sampling. The quantity and type of fission products deposited inside the containment building will be determined by periodically collecting and analyzing plateout samples that are located throughout the containment building. To adequately define the plateout characteristics of the various plateout media, it is necessary to construct the samples of identical material and surface composition (surface coatings, roughness, etc) as the material being investigated.

To determine the time-space behavior of fission products, such as iodine, and the saturation characteristics of the exposed containment building, plateout samples will be collected at representative positions throughout the containment volume as a function of time. These samples are identified as "removable samples". Other samples remaining inside the building until the termination of the test are identified as "nonremovable samples". All samples will be placed parallel to the surface being monitored and exposed to the containment atmosphere until they are to be removed from the vessel or until the termination of the test. The general location of the sample stations and the frequency at which samples will be taken are discussed in Section VII-3.

The containment wall temperature will be obtained at each location whenever a sample is removed. In addition, the following measurements will be made throughout the sampling or test period to assist in the evaluation of the fission product behavior.

(1) Temperature of the atmosphere both inside and outside the containment vessel,

(2) temperature gradient across the containment vessel walls,

(3) pressure and relative humidity in the containment vessel,

(4) gas leak rate from the vessel, and

(5) temperatures and air-steam velocities inside the reactor vessel.

The tentative procedure for analyzing the removable plateout samples will be as follows:

(1) Remove the activity from the plateout sample by leaching;

(2) perform a total radioiodine analysis. This will be accomplished by performing a gross gamma determination on the final silver iodine product;

(3) store the silver iodide for two days, then obtain a gamma energy spectrum and determine the quantity of I-133 and I-135;

(4) store the silver iodide for two to three weeks and rerun the gamma energy scan. Determine the concentration of I-131 by spectrum stripping;

(5) take aliquots of the solutions from step (1) immediately after dissolution and perform an analysis for Te-132, Cs-137, Sr-89-90 by radiochemical procedures; and

(6) store the remaining solution for one month and then obtain a gamma energy spectrum. Determine the concentrations of Ba-La-140, Ce-141, Ru-103, Zr-Nb-95 by the spectrum stripping.

After the nonremovable sample holders have been removed from the building the samples will be sealed in containers in preparation for analysis. The analysis of each sample will include the following:

(1) obtain preliminary gamma energy spectrum on each sample;

(2) store samples for approximately two weeks to a month and rerun the gamma energy spectrum. Determine the concentration of Ba-La-140, Ce-141, I-131, Ru-103, and Zr-Nb-95 by the spectrum stripping technique; and

(3) analyze for Sr-89-90, Cs-137 and other desired long-lived nuclides by radiochemical procedures.

6.42 Gas Sampling. Gas sampling encompasses the collection of representative samples of the containment building atmosphere following core meltdown. These samples will be used to determine the volatile and noble gas concentration in the air as a function of time. By obtaining samples as a function of time, information can be obtained on the time-space behavior of the isotopes in the atmosphere and the plateout characteristics of the containment vessel wall as a function of concentration in the air. In addition, these data will assist in evaluating the total quantity of fission products released from the fuel and the concentration available for leakage from the containment vessel as a function of time. Other samples will be obtained during the cleanup of the building to determine the effectiveness of the spray system, filter systems, and decontamination solutions for removing fission products from the atmosphere. The type and location of the gaseous sample containers in the containment building are described in Section VII-3.2.

After the "gas bombs" have been removed the high efficiency filter and carbon cartridge will be removed. The filter, carbon cartridge, and gas bomb will then be analyzed separately by the following tentative procedures:

Filter

(1) Obtain preliminary gamma energy spectrum;

(2) store for approximately a month and rerun gamma energy spectrum. Identify the concentrations of Ba-La-140, Ce-141, I-131, Ru-103, and Zr-Nb-95 by the spectrum stripping technique; and

(3) dissolve the filter and solids in HNO_3 or HNO_3-$HClO_4$. Analyze aliquots for Sr-89-90 and other desired long-lived nuclides by radiochemical procedures.

Carbon Cartridge

(1) Obtain preliminary gamma energy spectrum to determine total iodine concentration;

(2) store samples for two days and then rerun gamma energy scan to determine concentration of I-133 and I-135 by spectrum stripping technique; and

(3) store samples for two to three weeks and rerun gamma energy spectrum to determine concentration of I-131 by the spectrum stripping technique.

Gas Bomb

It is assumed that the gas bomb will be of a physical size to permit direct placement in the counting shield. If this is not the case, the gas will be transferred to the desired container.

(1) run preliminary gamma energy spectrum to determine gross concentration of Xe and Kr; and

(2) store for two days and rerun gamma energy scan to determine the concentration of Kr-85 and Xe-133 by the spectrum stripping techniques.

6.43 Particulate Sample. The particulate sampling device is designed to collect samples of the particulate activity and iodine in the atmosphere as a function of time after core meltdown. These data are necessary to evaluate the time-space behavior of the particulate activity and to determine the concentration of various isotopes in particulate form that are released from the core. In addition, these data will determine the particulate activity that is available for leakage from the containment vessel. The sample device and system are described in Section VII-3.

Since the filter cartridge has not been selected for insertion in the sampling device, the general procedure for analyzing the samples cannot be identified at this time. However, the samples will be analyzed for Ba-La-140, Ce-141, I-131, Ru-103, Zr-Nb-95, Sr-89-90, and Te-132.

6.44 Condensate Sampling. Fission product concentrations in the steam condensate will be obtained as a function of time following the loss of coolant. These data will provide information on the ability of the condensed steam to remove fission products from the atmosphere and from the containment vessel walls. In addition, this information will assist in obtaining a fission product balance. Samples also will be collected throughout plant cleanup to determine the effectiveness of each procedure or solution such as water spray and decontamination solutions for fission product removal.

6.45 Chemical State. The chemical state of the iodine released to the containment atmosphere, as a result of a loss of coolant accident, is important in understanding various iodine retention mechanisms and for the design of devices for iodine removal from the containment atmosphere.

It is proposed that at least four iodine collection devices, capable of quantitatively distinguishing the two basic forms of iodine released, be used for atmospheric sampling of iodine during the test. The two basic forms of iodine are elemental iodine and organic iodine compounds. Each collection device is composed of several sampling media, thus allowing several samples to be taken at each location at various times during the duration of the tests.

The technique or sampling device to be used for determining the chemical state is currently the subject of a separate research program.

6.46 Particle Size. An attempt will be made to identify the particle size of the fission products as a function of position inside the containment vessel. These data will be of value in evaluating the biological effects of the released fission products and in evaluating the filter efficiency following a loss of coolant accident.

Because of the environmental conditions inside the containment vessel, it is recognized that the most common techniques for determining particle size such as the sticky paper and impactor techniques may produce questionable results. Thus, a program is proposed to develop techniques for obtaining this information.

6.5 Fission Product Dispersion External to the Containment Vessel

In order to obtain information on fission product dispersion and to provide adequate warning should hazardous conditions develop, the area surrounding the LOFT facility will be extensively monitored. The grid for locating the monitoring instruments will consist of a series of concentric circles with radii extending from 50 to 6400 meters. The general layout of the grid and instrument locations is presented in Section VII-3. Heavily instrumented portions of the grid are two 60° sectors oriented for the most probable lapse and inversion meteorological conditions. The various types of instrumentation to be used on the grid are high-volume air samplers (airborne particulate activity and iodine), particle sizing samplers (airborne activity as a function of particle size), fallout plates (deposition dose), film badges (total beta and gamma doses), ionization chambers (dose rates), telemetering area monitoring stations (airborne activity, dose rate, and integrated dose), fission gas detectors (noble gas activity), and chemical dosimeters (total dose). Some of the larger external arcs will be monitored by mobile surveillance vehicles capable of tracking and recording data from the released effluent. These vehicles also will be used to recover the instrumentation and samples from the grid.

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

A complete meteorological history also will be maintained throughout the time that fission products are released from the containment vessel. The data to be obtained will include:

(1) The air temperature as a function of elevation,

(2) the wind direction and velocity as a function of elevation and distance downwind,

(3) relative humidity, and

(4) barometric pressure.

These data are required in order to evaluate the analytical models for predicting the fission product dispersion.

7. PHASE V -- FACILITY CLEANUP AND POST-TEST EXAMINATION

This phase of the experimental program consists of termination of the experiment, cleanup of the facility, and the post-test examinations to be performed on the reactor system and core.

After it has been concluded that all useful data have been obtained from the loss of coolant test, the test will be terminated. The containment building sprays will be employed to reduce the internal pressure subsequent to removing the airborne activity with the halogen and particulate removal filter system. Following filtration of the building contents, all internal surfaces including the experimental assembly will be remotely decontaminated to reduce the residual surface activity. After each phase of the cleaning operation has been completed, a plateout sample and air sample from each of the 12 sampling stations and a water sample from the collection tank will be taken. The fission product concentration remaining with each sample will, therefore, indicate the effectiveness of the previous filtering and decontamination procedure.

The large railroad doors then will be opened and the railroad dolly containing the nuclear package will be removed to the hot shop for post-test examination using the shielded locomotive.

The general procedure for each part of the operation and the data to be obtained are discussed below.

7.1 Fission Product Removal from the Containment Atmosphere

Fission product removal from the atmosphere will be accomplished by the use of the pressure reduction sprays and filtration system. The pressure reduction sprays will be activated for the purpose of reducing the containment pressure to ≈ 1 psig, if it has not already reached this point by normal pressure decay, and to aid in the removal of airborne and deposited fission product activity. The sprays will be operated for a period of 1/2 hour at which time a plateout sample and air sample from the 12 sampling stations and a water sample from the collection tank will be taken. After the above samples have been collected, the spray system will be reactivated and allowed to operate for an additional hour or until the pressure is reduced to 1 psig. Following final deactivation of the pressure sprays and prior to activation of the filtration system, a plateout sample and air sample from the 12 sampling stations and a water sample from the collection tank will be taken.

Further reduction of the airborne activity will be accomplished by either a "once through" venting or a recirculation of the containment air through the fission product removal filter system. The technique to be used will be dependent upon the magnitude of the airborne activity as well as the meteorological conditions existing at the time filtration is initiated. The "once through" venting technique involves discharging the airborne activity through a series of filtering media (as described in Section VII-3) to the atmosphere via an exhaust stack. This method of fission product reduction will be used only if the airborne activity is at a level low enough to prevent exceeding the allowable on-site and general population exposure criteria and only if the meteorological conditions favor subsequent release.

In the event of excessive airborne activity or unfavorable meteorological conditions the containment vessel atmosphere will be recirculated through the filter media until such a time as exhaust to the atmosphere is considered feasible. Following the fission product filtration and prior to remote decontamination, plateout samples and air samples will be obtained from the 12 sampling stations.

7.2 Decontamination

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

Fission product plateout, air, and water samples will be obtained from the appropriate sample stations following decontamination. Analysis of the various samples taken during the pressure reduction, filtration, and remote decontamination process will indicate the efficiency of each decontamination step as well as the final airborne and deposited fission product activity remaining at the conclusion of the cleanup operation.

7.3 Transport of Test Package to Hot Shop

Removal of the test package to the hot shop will be accomplished whenever the air activity inside the containment vessel is at a level low enough to permit operation of the railroad access doors. After the doors have been opened, the Mobot will enter the test facility for television examination of the reactor test package. All pieces of loose material, if present, which can be lifted by the Mobot will be placed on the railroad dolly in such a manner as to preclude any dislodging of the material during transport of the test package to the examination area. After the appropriate connections have been disconnected, the shielded locomotive will enter the containment building and remove the test package to the hot shop.

7.4 Post-Test Examination

The reactor system will be transported to the hot shop for the post-test examination. The transport of the reactor system is expected to begin approximately ten days after completion of the destructive test. The reactor vessel is to be located in the center of the rail dolly and will be the focal point of all operations connected with the post-test examination. The reactor system is being designed to allow movement into the hot shop and to allow for access of the hot shop manipulators into the opened vessel. Because of the excessive weight of the reactor system, some modifications will, of necessity, be made to the turntable outside the hot shop.

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

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

After arrival in the hot shop, the first major operation will be a complete external investigation of the system. During this investigation, pipes will be checked for breaks, welds will be checked for cracks, and the vessel and all other components will be checked for any signs of mechanical damage. All irregularities will be thoroughly documented.

7.41 Core. Upon completion of the external investigation, two lower vessel, eight-inch inspection nozzles will be remotely opened to allow entry of borescopes and small TV cameras to examine the under side of the core. Hopefully, it will be possible to determine the geometry and condition of the core prior to starting remote disassembly operations. Depending on the extent of core damage, a secondary critical experiment may be warranted to obtain experimental information and develop techniques for deactivating a power reactor in the event a maximum credible accident should occur.

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

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

Through a study of the fuel and cladding inventories, it should be possible to determine the percentage of fuel and cladding which have melted. Knowing the percentages of meltdown and the final and initial configurations, an attempt will be made to arrive at some concept of the meltdown history. The meltdown information will be related to information which is to be derived from mechanical studies of the core components. These studies should provide information concerning thermal and mechanical stresses, temperature gradients, mechanical distortions, etc. By relating the information obtained from the meltdown and mechanical studies, it is hoped that an understanding of the core disassembly process can be obtained.

Simultaneous with and subsequent to the core removal operations, a thorough search will be made to determine if there exists any indications of chemical reactions. If this search indicates the presence of chemical reactions, all available techniques will be used to determine the relative percent and magnitude of these reactions.

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

In order to arrive at some estimate of the relative fission product plateout percentages, some knowledge of the available isotopes will be required. Because of the delay time before and during the post-test examination, the short half-life fission products will have decayed out. The following isotopes have sufficiently long half-lives and are produced in sufficient amounts to warrant investigation in this experiment: Kr-85, Sr-89, Sr-90, Zr-95, Ru-103, Ru-106, I-131, Te-132, Xe-133, Cs-137, Ba-140, Ce-141, Ce-144. To arrive at an estimate of the amounts of these isotopes which were available for plateout, an analysis will be made of the fission products still contained in the molten and unmolten fuel. This analysis will consist of sampling and comparing the relative amounts of the contained fission products. The differences in contained fission products will then be related to the relative amounts of molten and unmolten fuel in order to obtain an estimate of the fission products which were released and the amounts which plated-out; some conclusions concerning the plateout characteristics of the previously named isotopes can be reached.

Upon completion of the post-test examination, the reactor system, with the exception of the vessel, is expected to be in a serviceable condition. The only components which will have been removed are those which were located inside the reactor vessel itself. Therefore, reactivation of the LOFT system may require only the replacement of the internal vessel components, providing the system integrity has not been reduced by thermal stress.

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

VI. REACTOR EXPERIMENTAL ASSEMBLY

1. EXPERIMENTAL ASSEMBLY

1.1 General

The nuclear system proposed to carry out the LOFT program is a pressurized water system containing a low enrichment, rod type, stainless steel clad oxide core having a power capability of approximately 50 MW(t). The core proposed for these tests reflects, as far as possible, contemporary reactor physics and engineering technology. No attempt has been made to simulate or duplicate the features of any particular operating or planned nuclear system. Instead, the design incorporates those features common to most commercial power reactors of the pressurized water type.

The underlying philosophy of the dolly mounted reactor systems design is the remote assembly, disassembly, and maintenance of the experimental equipment in the TAN hot shop during and after the experiments. The dolly mounted nuclear system also will facilitate cleanup and movement from the test building to the examination area(hot shop) following the final loss of coolant test. During the performance of the test, the dolly containing the nuclear system will be housed inside the LOFT containment which is capable of withstanding the pressure associated with coolant expulsion and of limiting the fission product leakage to the atmosphere.

The railroad dolly is approximately 48 feet long by 20 feet wide, is of steel construction, and is equipped with commercially available running gear. The reactor pressure vessel is mounted approximately at the center of the car. The overall height of the equipment assembled on the dolly does not exceed the height limitation of the TAN hot shop door.

The reactor, primary coolant system, and associated equipment installed on the dolly are supported from an integral structure resting on four load cells designed to provide weight loss information as a function of time during the system blowdown. The dolly and experimental equipment are shown in Figure 2.

The dolly floor and the supporting framework are designed to provide smooth surfaces without pockets to facilitate decontamination. Instrumentation and control leads are armored and located for maximum protection from the blowdown jet, flying missiles, and moisture. The instrumentation and control leads are shielded from stray electrical fields and are connected through containment vessel penetrations to instrumentation and controls located in the control and equipment building. The containment vessel penetrations containing instrumentation and control leads are completely separate from the power wiring penetrations.

1.2 Reactor Pressure Vessel

1.21 General. The 57-3/4-inch ID by 24-foot-high reactor pressure vessel assembly, shown in Figures 3, 4, and 5, consists of a cylindrical shell with a removable top head joined to the vessel by a flanged and bolted joint. The

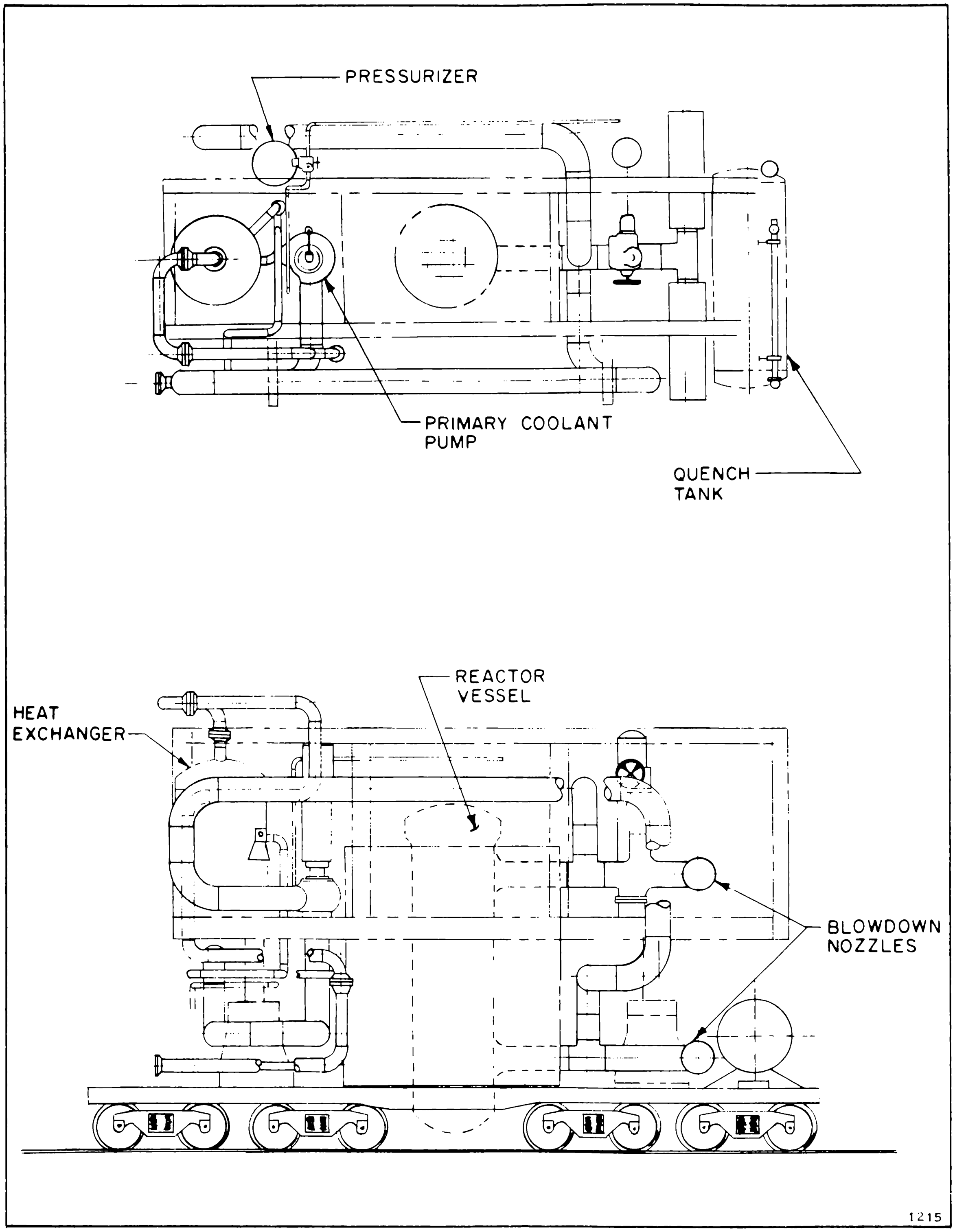


FIG. 2 LOFT DOLLY ELEVATION AND PLAN VIEW.

vessel interior contains an inlet flow baffle plus provisions for supporting the core structure and the thermal shields.

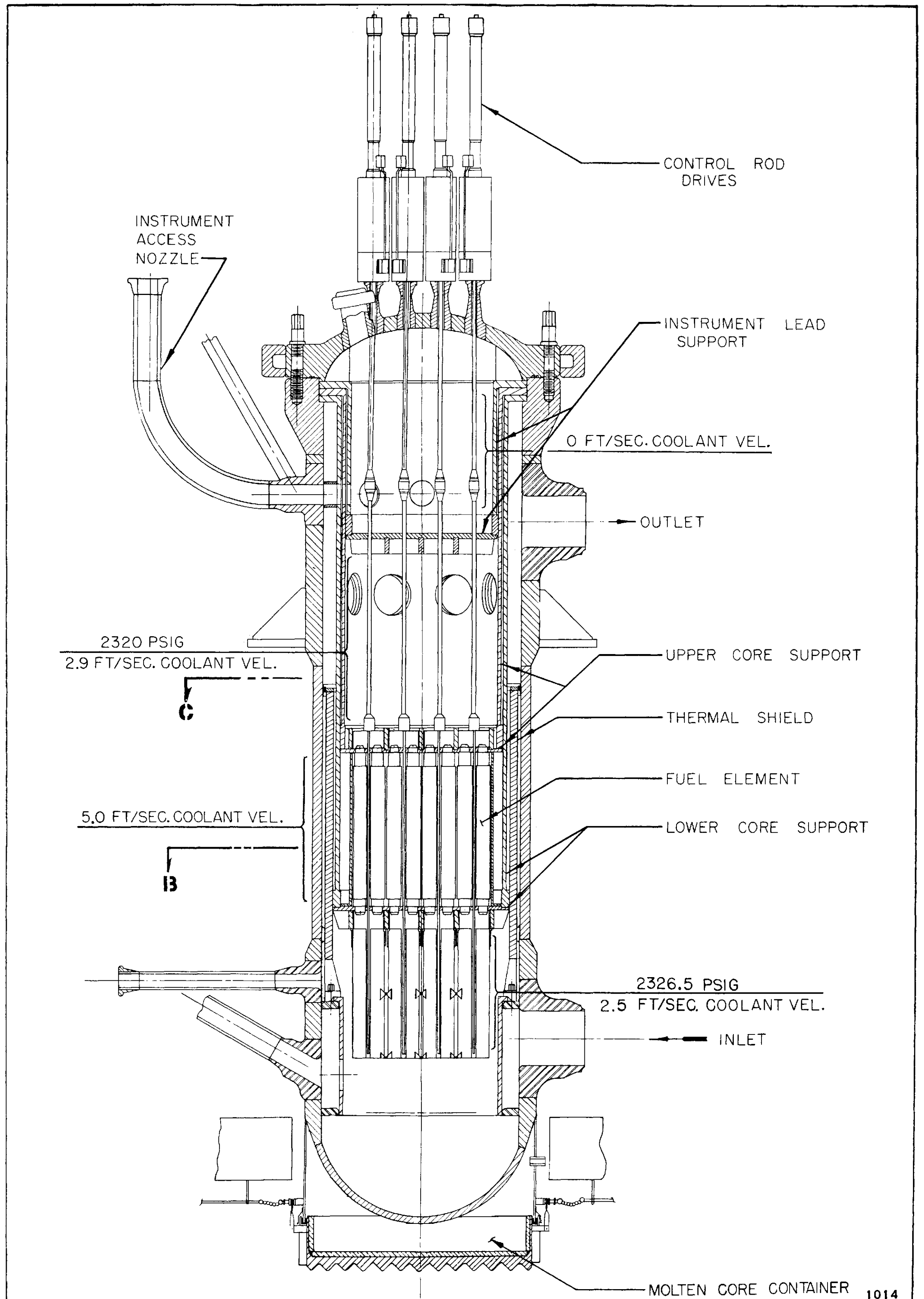
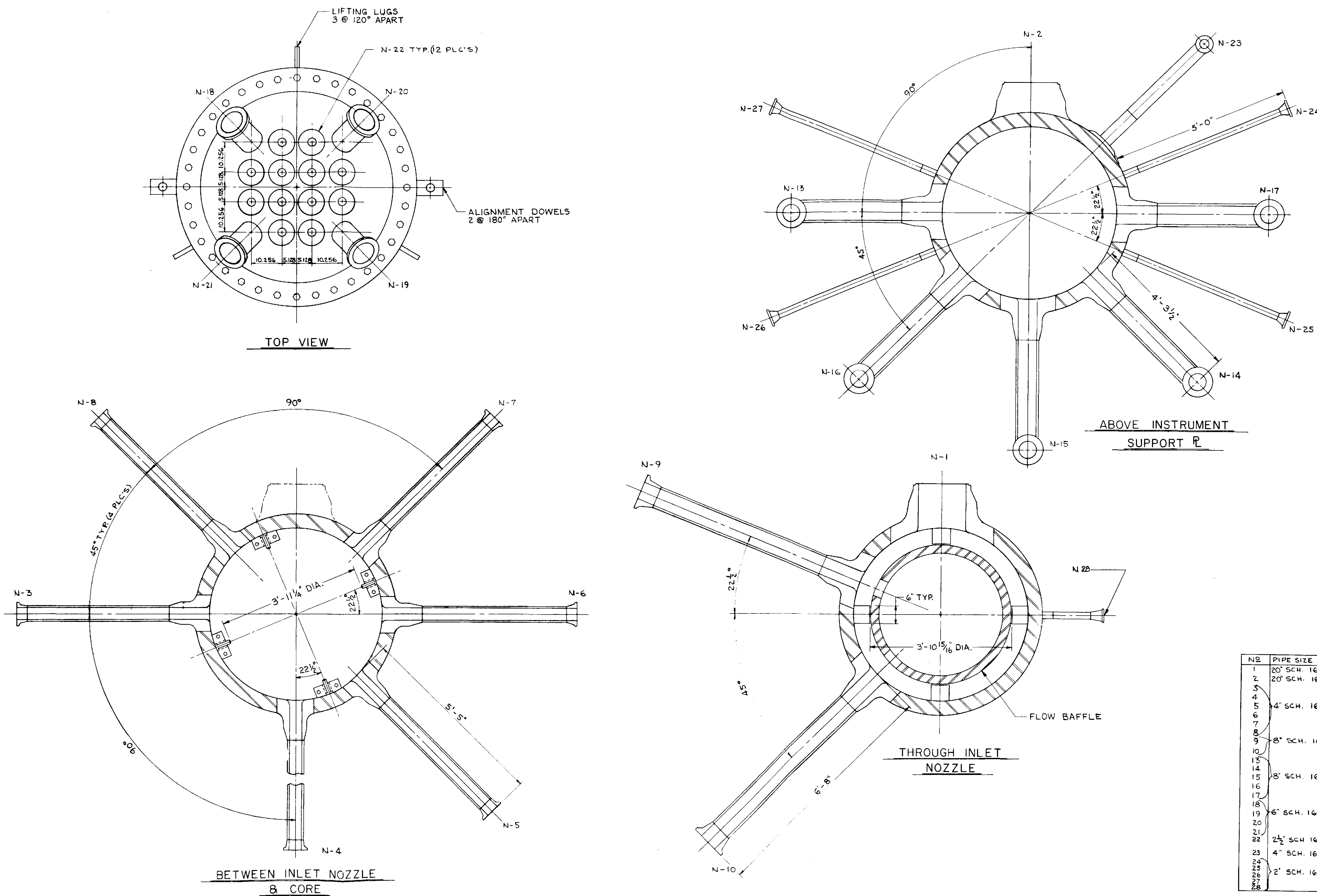


FIG. 3 LOFT VESSEL CROSS SECTION.



NOZZLE SCHEDULE

NO	PIPE SIZE AND I.D.	DESCRIPTION
1	20" SCH. 160 - 16.064"	PRIMARY COOLANT INLET
2	20" SCH. 160 - 16.064"	PRIMARY COOLANT OUTLET
3, 4, 5, 6, 7, 8	4" SCH. 160 - 3.438	INSTRUMENTATION ACCESS
9, 10	8" SCH. 160 - 6.813	INSPECTION ACCESS
13, 14, 15, 16, 17	8" SCH. 160 - 6.813	INSTRUMENTATION ACCESS
18, 19, 20, 21	6" SCH. 160 - 5.189"	INSTRUMENTATION ACCESS
22	2½" SCH 160 - 2.125"	ACTUATOR NOZZLE
23	4" SCH. 160 - 3.438"	PROBE NOZZLE
24, 25, 26, 27	2" SCH. 160 - 1.689"	TELEFLEX NOZZLE
28		CATCH CAN NOZZLE

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FIG. 4 LOFT NOZZLE SUMMARY

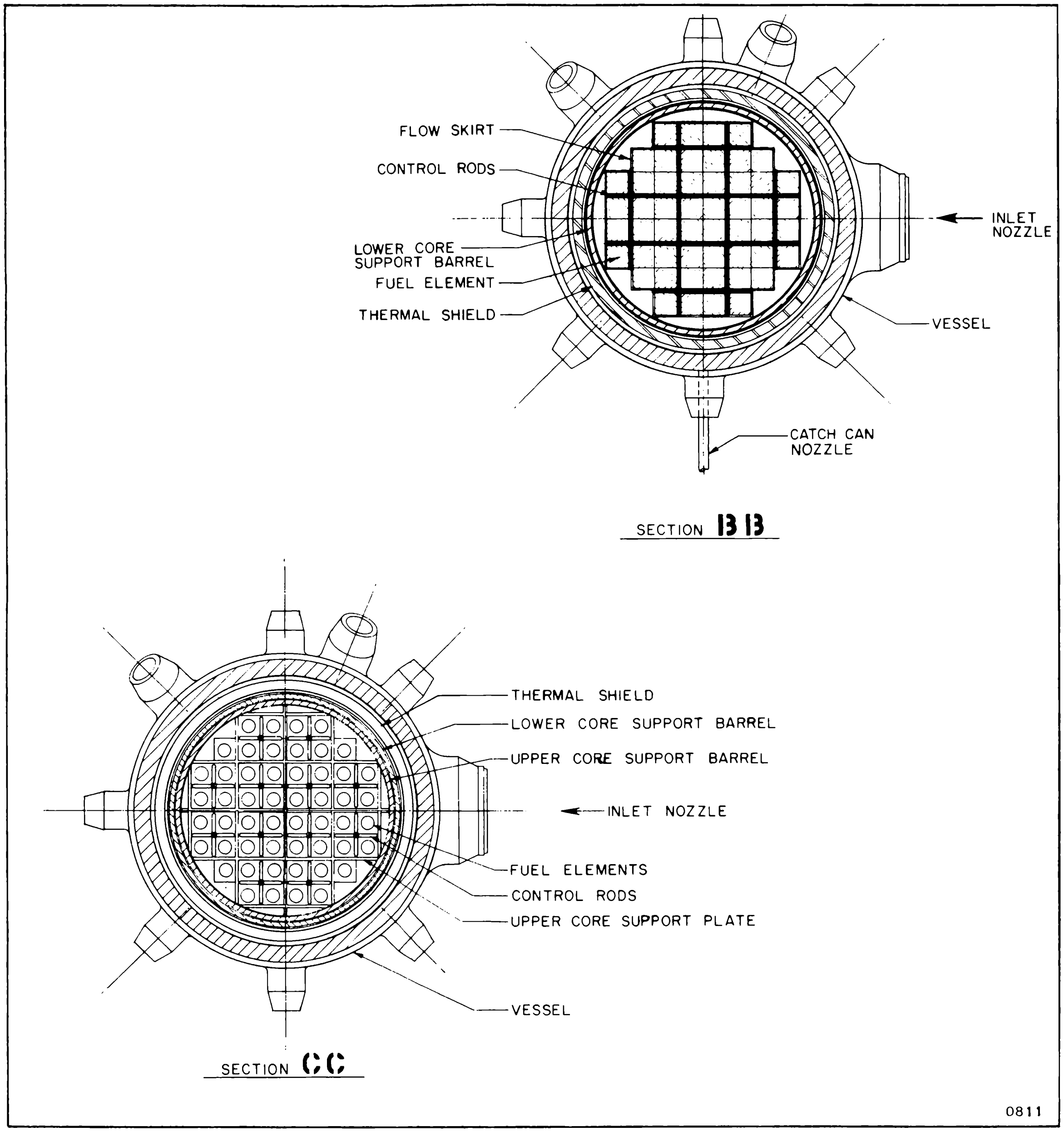


FIG. 5 LOFT VESSEL SECTIONS.

A catch pan is designed to attach below the bottom head of the reactor pressure vessel to confine any fuel that might melt through the vessel and to prevent the accompanying fission products from escaping to the containment vessel by any route other than the blowdown nozzle. The container is capable of retaining 75 percent of the core material (6.25 cubic feet) at 3500°F. The catch pan is designed so that the convective heat loss from the bottom head, prior to possible melt-through, is maintained at approximately the level expected if there were no catch pan.

An inlet flow diffuser baffle is provided in the bottom of the vessel to distribute the coolant flow uniformly and prevent direct coolant impingement on the control rod shrouds. The baffle is designed to withstand the full force of the blowdown.

Studs used for the main closure are designed for remote tightening and loosening and have anti-seize platings or films applied to all fastenings used both internally and externally on the vessel. The vessel is designed in accordance with ASME Code.

1.22 Materials of Construction -- Reactor Vessel. Materials selected for the reactor pressure vessel are as follows:

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

1.23 Physical Description -- Reactor Vessel

Vessel ID	- 57-3/4 inches
Vessel OD	- 67 inches
Height	- $\approx$ 24 feet
Weight	- 100,000 pounds (less internals)
Cylindrical shell thickness	- 4-1/2 inches

1.24 Nozzle Description

Two, coolant nozzles

Four, 6 inches for ID sample (top head)

Two, 8 inches for inspection (bottom head)

Five, 8 inches for instrumentation (shell above core)

Six, 4 inches for instrumentation (shell below core)

One, 3 inches for instrumentation (shell above core)

Twelve, actuator nozzles (top head) and nozzles as required for operational instrumentation

Four, 2 inches for instrumentation (shell above core)

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

1.25 Components to be Remotely Handled. The following parts of the vessel are designed for remote disassembly after the experiment:

(1) vessel head, including bolting and seals,

(2) internal core support barrels,

(3) flanges on two, 8-inch inspection nozzles in the bottom head and four, 4-inch sample nozzles in the top head,

(4) control rod actuators, and

(5) thermal shield.

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

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

1.27 Insulation. The vessel is insulated on the exterior surface to minimize thermal stress in the heavy wall during the loss of coolant tests. The insulating material is radiation resistant, fire-proof, and will not break down to produce toxic or corrosive products. It is anchored to the vessel exterior by studs applied in the field, and the entire exterior is lagged with sheet metal which is spot welded in place. Overlaps in the lagging are provided to reduce moisture penetration and contamination. Lagging material was selected on the basis of corrosion resistance and low gamma neutron shutdown activation.

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

1.3 Core Support Structure and Thermal Shield

The general layout of the support structure, the thermal shield, and the core are shown in Figure 3. The reactor core support structure consists of an upper and lower plate, an upper and lower support barrel, and 12 control rod shrouds. This structure supports the core weight, aligns the core assemblies and control rods, and provides for the proper channeling of the upward coolant flow. The entire core support structure rests as an integral unit on the support ledge of the reactor pressure vessel and is restrained by the compressive load that is applied by the top head. A compression flange and guide pin positioning holes are provided to restrain and orient the core support structure.

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

A 52-inch ID concentric thermal shield surrounds the reactor core in the annulus between the lower core support barrel and the vessel wall to reduce the gamma heating rate of the reactor vessel wall. The two-inch thick thermal shield is supported and oriented by brackets which are an integral part of the vessel. This shield extends 18 inches below the bottom of the active core. The support brackets are located in such a manner that they do not obstruct the horizontal access of the lower instrument nozzles.

1.4 Reactor Core (Figure 6)

The reactor is fueled with slightly enriched uranium dioxide in the form of sintered cylindrical pellets stacked in stainless steel tubes. The individual

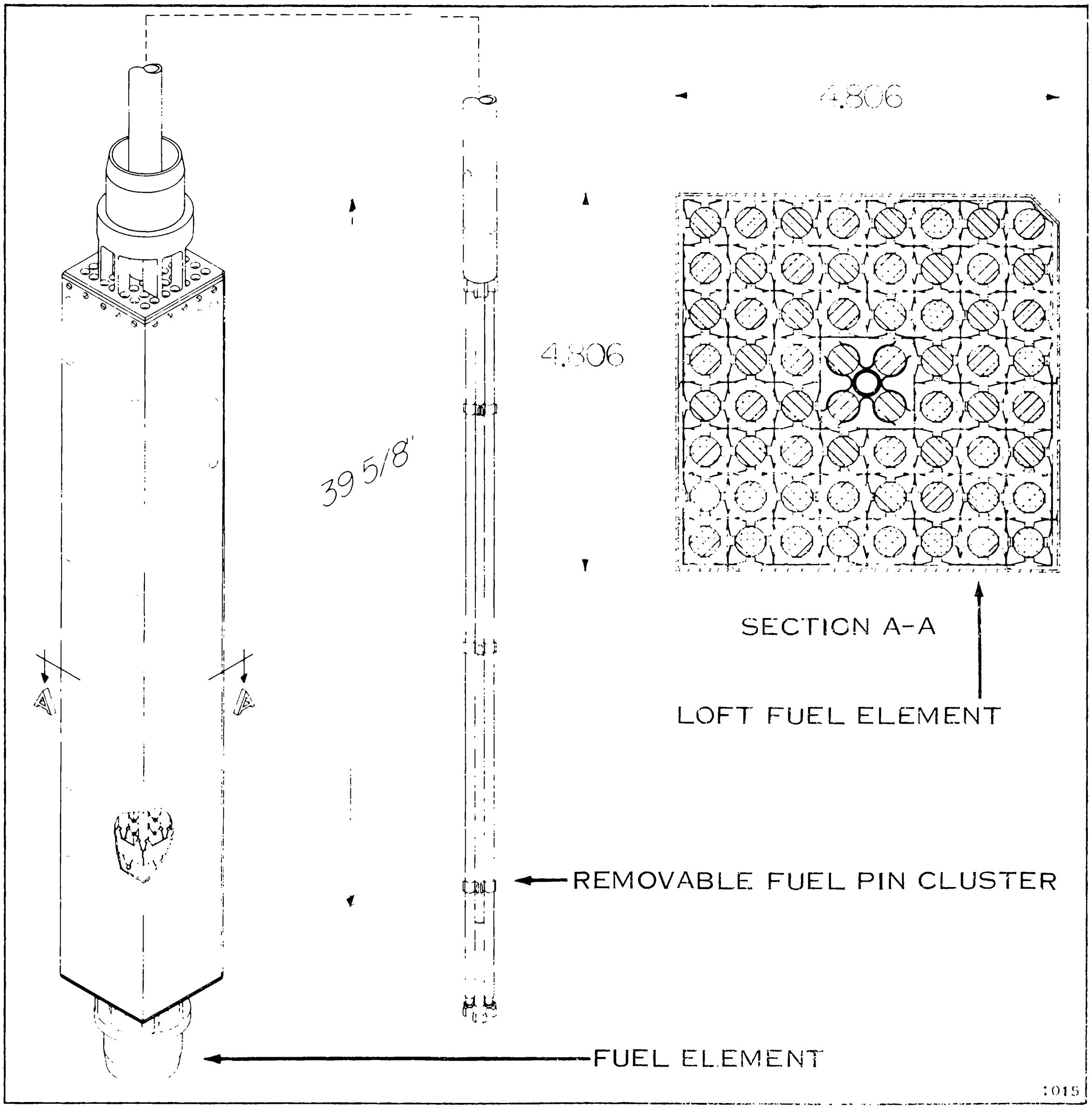
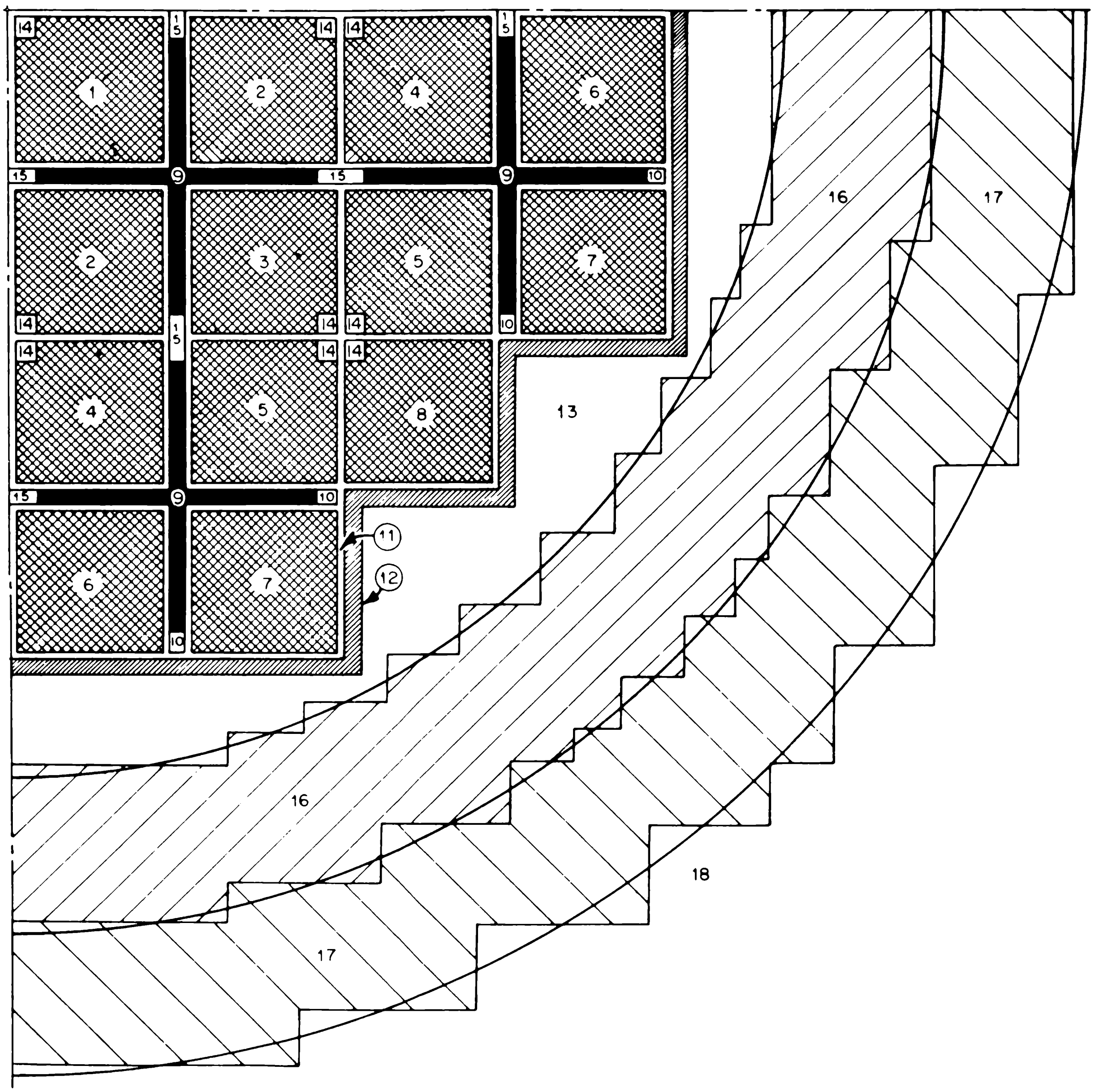


FIG. 6 LOFT FUEL ELEMENT ASSEMBLY.

tubes or pins are grouped into 52 fuel elements. Each fuel element has the capability of retaining 64 individual fuel pins with an active fuel length of 36 inches. The pins are maintained in proper position by spring clip spacer grids. These are welded to a perforated stainless steel enclosure in parallel planes perpendicular to the length of the element. For fuel element details see Figure 7.

The fuel element is equipped with restraining grids at both top and bottom. These grids allow the required flow through the assembly, the removal of the



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

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FIG. 7 LOFT CORE QUARTER SECTION.

corner pin for possible replacement with poison pins, and restrain the remainder of the fuel pins in their vertical position.

All 52 fuel elements contain a removable subassembly section to allow the removal of four enclosed fuel pins while the top core support plate is in position. The fuel elements are positioned in the core by top and bottom end boxes which mate with the upper and lower core support plates. The core is surrounded by a form-fitting enclosure which confines the upward coolant flow within the fuel bearing zone. This enclosure is safety-locked and -bolted and doweled to the flange of the lower core support barrel.

The total primary coolant flow is 5.6 x 10^6 lb/hr of which 4.5 x 10^6 lb/hr flows past the fuel pins, and 1.1 x 10^6 lb/hr is the leakage flow. The rate of leakage flow was determined from heat removal requirements of the flow skirt, the lower core support barrel, the thermal shield, and the vessel wall (Table IV). Additional allowance was made for the coolant flow in the control rod channels.

TABLE IV

LEAKAGE FLOW REQUIREMENTS FOR HEAT REMOVAL

Component	Heat Flux $X10^{-4}$ Btu/hr-ft^2	Flow (gpm)	Inside Temp (°F)	Outside Temp (°F)	Max Temp (°F)
Flow Skirt					
(inside)	0.856	597	543	552	553
(outside)	0.298				
Support Barrel					
(inside)	0.444	666	558	558	561
(outside)	1.13				
Thermal Shield					
(inside)	1.13	237	558	558	598
(outside)	0.456				
Pressure Vessel					
(inside)	0.456		558	600	600

1.5 Control Rods and Drives

Twelve top-entry control rod drive mechanisms are provided for movement of the cruciform control rods. The drive mechanisms, capable of sealing a 2750 psig, 600°F water environment, are mounted on access nozzles located on the vessel top head on 10-inch centers. The drive mechanisms are capable of individual or group operation and simultaneous scram of all rods.

The control rods are located in the fuel element interstices. Each rod is composed of an active stainless steel poison section containing 1.5 weight percent B^{10} and a zirconium follower section to minimize the flux peaking when the absorber section is withdrawn.

1.6 Reactor Shield Tank

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

(1) Reducing the thermal neutron flux emanating from the shield to less than 10^5 neutrons/(cm^2)-(sec) during normal power operations at full power

(2) Reducing the dose rate from fission products and activation gammas to less than 200 mrem/hr at the surface of the shield at 15 minutes after a normal shutdown. The reactor is assumed to have operated for 3333 MWd prior to shutdown.

The dolly is designed to permit transport of the experiment with the reactor shield filled with water.

2. NUCLEAR CHARACTERISTICS

2.1 Reactor Physics

Reactor physics calculations for the LOFT reactor were performed using two dimensional four-group diffusion theory. Experience with similar light water moderated cores employing low enrichment UO_2 fuel elements has shown that this approach is satisfactory for conceptual design purposes. The results of calculations previously carried out for the MARTY critical experiment and the Spert I UO_2 core [6] were found to be in agreement with experimental data.

Thermal constants for neutrons with energies from 0 to 0.625 eV were obtained by basing the calculations on the Wigner-Wilkins energy distributions. These were calculated by using the 650 SOFOCATE program [7]. An initial neutron energy spectrum was computed with the atom densities homogenized on a volume fraction basis. Constants for the pure materials within a fuel pin unit cell were obtained by averaging their energy dependent cross sections over the previously calculated energy spectrum. With these constants, a P-3 spherical harmonics program [8] was used to calculate a fine flux profile over the unit cell. These calculations were used in determining flux disadvantage factors for the UO_2 and stainless steel cladding. A second Wigner-Wilkins spectrum was then computed with atom densities, both flux and volume, weighted by the unit cell flux. The constants obtained from this procedure allowed the fuel core region to be treated as homogeneous when considering thermal neutrons in the diffusion calculations.

The fast group constants were determined from the GAM-1 Code [9]. In calculating effective resonance integrals for U-238, the code takes into account geometric effects of fuel lumping and Doppler broadening. Except for this heterogeneity correction, the fueled core was assumed to be homogeneous with respect to fast neutrons.

Except for the control rod poison sections, both fast and thermal group constants for all nonfuel interior regions were obtained by averaging the cross sections over the GAM-1 and the SOFOCATE fuel core spectrums, respectively. Thermal constants for the control rod poison sections were calculated by assuming that these sections were "black" to thermal neutrons. This conclusion was reached by assuming a logarithmic boundary condition. Fast group constants for the poison sections were obtained from the MUFTR-D Blackness Coefficients Edit [10]. All constants for regions exterior to the core were calculated by averaging the cross sections over a water spectrum.

PDQ-3 Computer Code [11] calculations were made in X-Y geometry with both quarter-core and full-core representations to establish the reactivity characteristics and neutron flux distribution for the LOFT core design. TURBO [12] Computer Code calculations were used to determine excess reactivity requirements for this size reactor having an average coolant temperature of 540°F. The quarter core representation used for the diffusion theory calculations is shown in Figure 7. The use of different compositions to describe each fuel element region permitted several nonuniform fuel loading studies to be undertaken with a single mesh-composition overlay to describe the core. Provisions also have been made for representation of dummy fuel pin regions and of filler plugs in the water annuli between fuel elements.

With the core geometry in the same configuration as previously described, the U-235 content of the UO_2 may be varied to help obtain the necessary reactivity in the cold core. In addition, the number of dummy poison pins may be varied to effect some changes in reactivity. The core must have sufficient reactivity to maintain criticality as the reactor temperature and power are raised to operating conditions. This is necessary to override equilibrium xenon and samarium poisoning, and to sustain fuel burnup for a maximum operating time of 1600 hours.

To arrive at the required reactivity and, hence, the fuel enrichment, TURBO lifetime calculations were made. The results of initial calculations indicate that the reactivity requirement for fission product poisoning and fuel burnup is approximately 0.055 Δk. Therefore, the neutron multiplication (λ) at the beginning of core life must be about 1.058. To determine the fuel enrichment necessary, several PDQ problems were run varying the enrichment in two distinct core regions (assemblies number 1 through 5 and assemblies number 6 through 8 as shown in Figure 7). The purpose of the two-zone fuel loading is to reduce the peak power densities and attain a flatter radial power distribution thereby achieving meltdown in a larger portion of the core. Therefore,

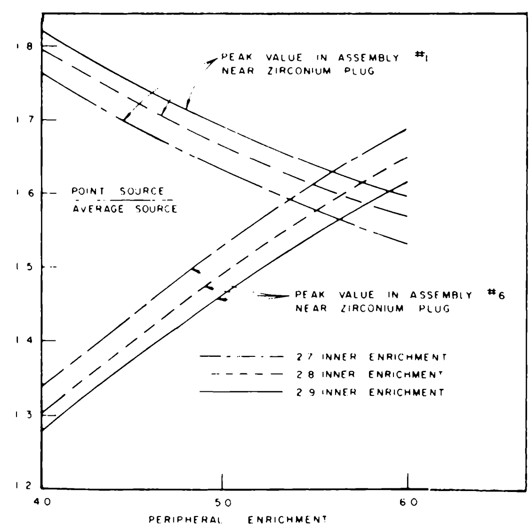


FIG. 8 LOFT SOURCE RATIO VERSUS PERIPHERAL ENRICHMENT.

optimum fuel loading has been considered as one which results in a minimum peak-to-average power ratio. Values from the enrichment studies on peak-to-average power behavior are plotted as a function of peripheral fuel enrichment in Figure 8. The crossover points of the upper and lower curves represent the minimum peak-to-average power attainable for a given set of enrichments. In all problems the peak power density in the inner zone occurred in the corner of assembly number 1 near the zirconium filler plug.

The eigenvalue results and descriptions of these problems are presented in Table V. In Figure 9 the normalized eigenvalues are plotted as a function of peripheral fuel enrichment. The intersections of the curves with a line $\lambda = 1.058$, the desired eigenvalue, then determine three pairs of enrichments which will give the necessary hot core reactivity.

TABLE V

EIGENVALUES AS A FUNCTION OF INNER AND PERIPHERAL FUEL ENRICHMENTS

(Moderator Temperature 544°F and Fuel Temperature 1500°F, Zirconium Followers)

Fuel Enrichments (%)			
Inner Zone	Outer Zone	λ	$\rho = \frac{\lambda - 1}{\lambda}$
2.7	6.0	1.0494	0.0471
2.7	4.0	1.0324	0.0314
2.8	6.0	1.0585	0.0553
2.8	5.0	1.0509	0.0484
2.8	4.0	1.0423	0.0406
2.9	4.0	1.0497	0.0473
2.9	6.0	1.0655	0.0615

The normalizing factor of 1/1.02 was used in similar calculations for the MARTY experiments resulting in calculated eigenvalues that were approximately two percent high. In Figure 10 the outer fuel enrichment is plotted as a function of inner enrichment for the crossover points of Figures 8 and 9. The intersection of these two curves then determined the optimum fuel loading for this particular reactor configuration. This loading is seen to be 2.82 and 5.71 percent for the inner and outer zones, respectively.

2.2 Reactor Control

As shown in Figure 5, the LOFT core employs 12 cruciform control rods. Boron stainless steel, enriched to 1.5 percent by weight in B^{10} has been tentatively selected as the control rod material. The follower sections were fabricated from zirconium. The design was based on the requirement that sufficient control rod material be available to render the cold clean reactor at least five percent subcritical under normal operating conditions and to shut the reactor down in the event of a stuck rod in the full-out position. The worth of all 12 control rods was found to be 23 percent in reactivity with poison dummy pins in the core. When the dummy pins are replaced by fuel, the rod worth is decreased to 21 percent. The worth of eleven rods (one rod stuck out) was found to be 15.6 percent if the dummy pins were present in the core. On the

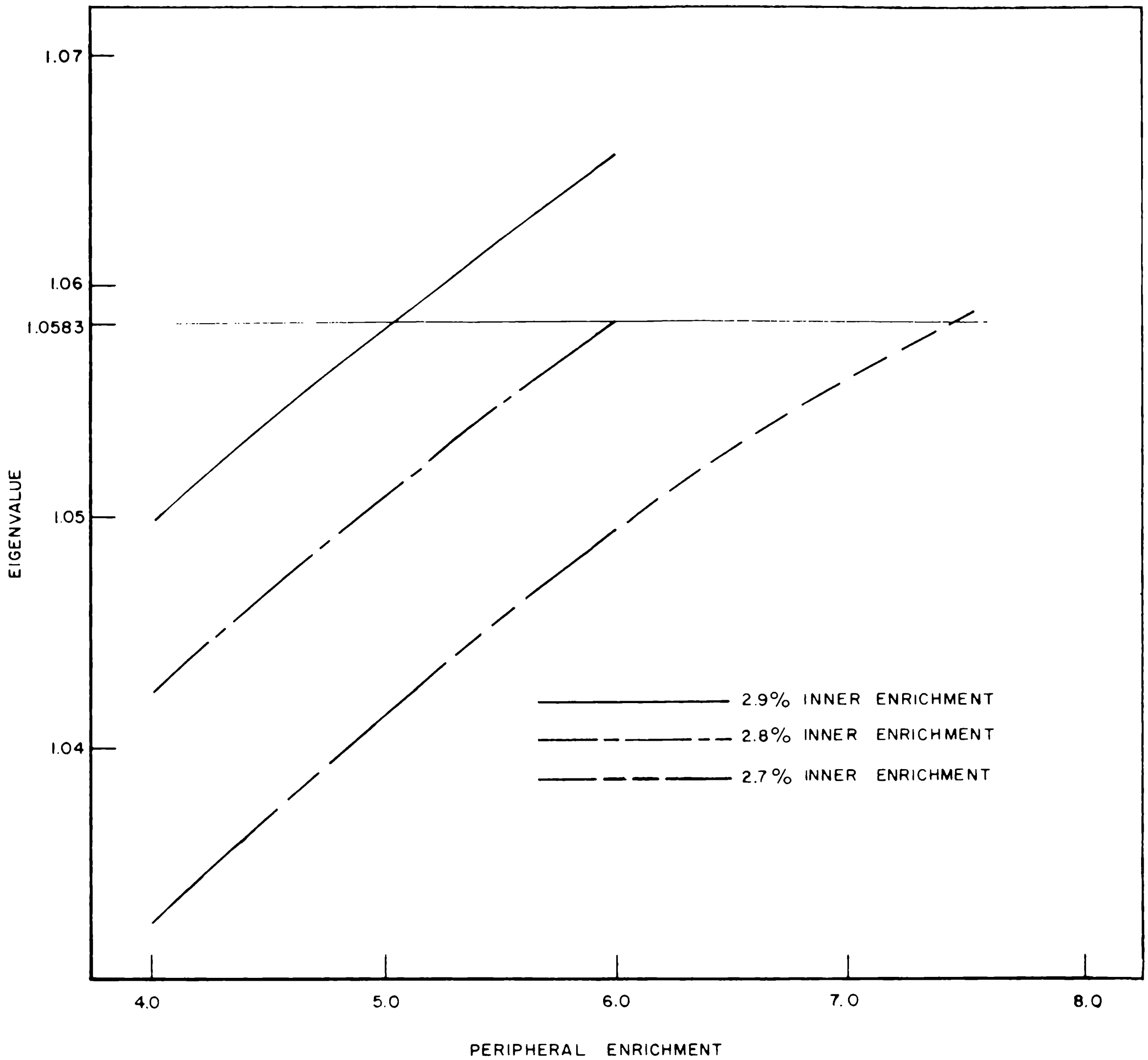


FIG. 9 LOFT EIGENVALUES VERSUS PERIPHERAL ENRICHMENT.

basis of these preliminary studies, it is estimated that the worth of eleven rods will be decreased by about one percent when the dummy pins are replaced with fuel pins.

In general, it was found that zirconium filler plugs have practically no effect on rod worth or core reactivity, but reduce local power densities approximately 12 percent below those occurring with water slots. Reactivity shimming by use of the poison dummy pins was found to increase cold core reactivity by 0.0189 $\Delta\rho$ when all dummy pins were replaced by fuel. Pertinent eigenvalues and the peak-to-average power density calculated for this core with dummy pins and optimum fuel loading are listed in Table VI.

2.3 Gamma Heating (External to Core)

In order to set the cooling flow in regions external to the core and to analyze thermal stresses in the support barrel and pressure vessel, a series of gamma heating calculations was made. The calculations of total gamma heating considered two sources: primary, which were computed using GRACE-I Computer

Code [13] and secondary, which were computed using GRACE-II Computer Code [14]. The primary gamma sources considered were those appearing in the core from thermal and epithermal neutron capture, prompt fission, and fission product decay. The secondary gammas considered were those resulting from thermal capture in regions external to the core. Input information to these codes concerning energy group structures, prompt fission and fission product decay, gamma spectra, buildup factors, and attenuation coefficient were obtained from the work of Avery et al [15]. One dimensional diffusion theory calculations were made to determine the primary gamma sources. The thermal flux exterior to the core also was determined with this calculational technique using as input the fast removal flux from previous GRACE-II results. GRACE-I was then used to calculate gamma heating in the external regions using the resultant thermal fluxes.

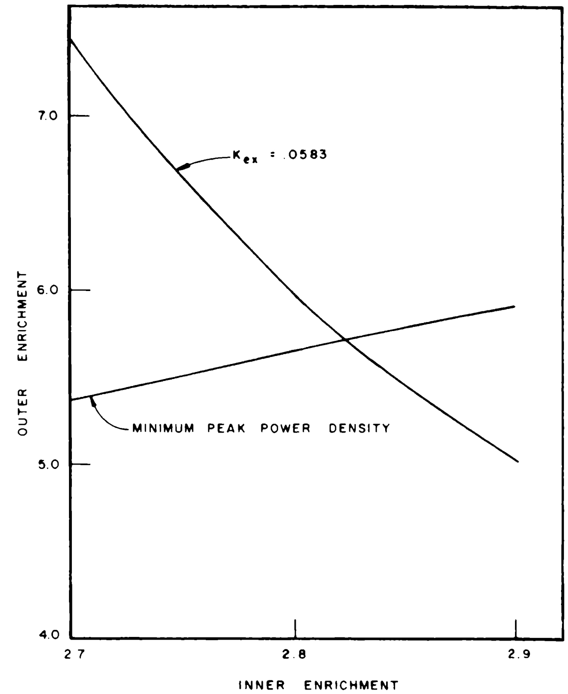


FIG. 10 LOFT OUTER ENRICHMENT VERSUS INNER ENRICHMENT.

TABLE VI

LOFT EIGENVALUES AND PEAK-TO-AVERAGE POWER DENSITY

K_{eff}(cold)	1.128
K_{eff} (hot, full power)	1.057
K_{eff} (cold, 12 rods in)	0.896
K_{eff} (cold, 11 rods in)	0.959
Radial Peak/Average Power	1.60

Figure 11 presents a radial plot of thermal flux from the core out into the reactor shield tank and Figure 12 gives the gamma heating plot from the core out into the reactor shield tank. The values of gamma heating are axial maximums as they are based on the core midpoint values assuming an axial cosine distribution.

2.4 Summary of Design Features

A summary of the important design features of the core is presented in Table VII. Even though some changes may be made in the reactor design as a result of additional information which will be obtained later, present calculations

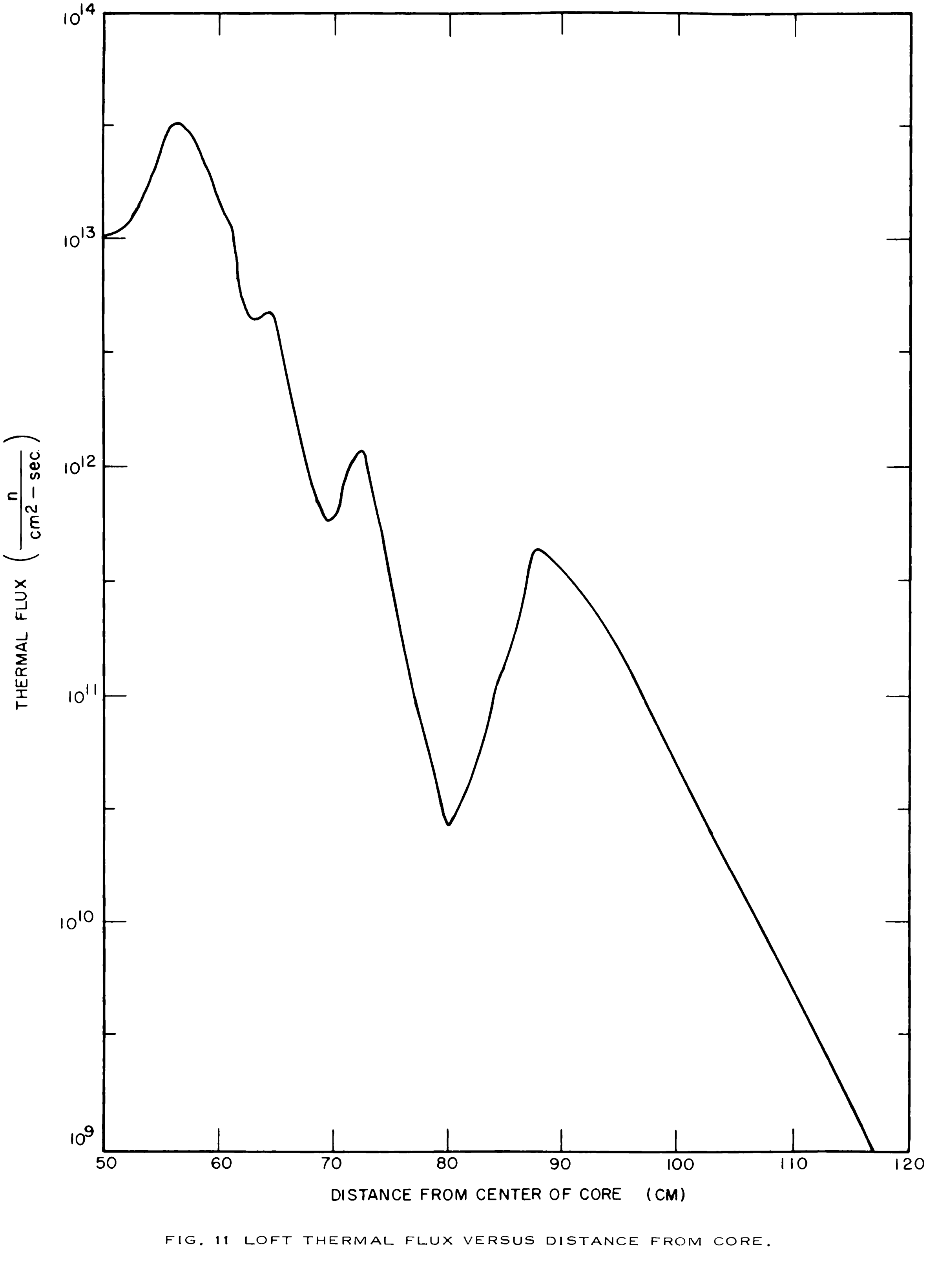


FIG. 11 LOFT THERMAL FLUX VERSUS DISTANCE FROM CORE.

TABLE VII

LOFT CORE DESIGN FEATURES

Active length (in.)	36
Number of fuel assemblies	52
Number of pins/assembly	64
Fuel element	
Fuel pellet diameter (in.)	0.357
Gap between clad and fuel (in.)	0.002
Clad thickness (in.)	0.015
Fuel pin diameter (in.)	0.391
Rod spacing (in.)	0.580 (center to center)
Clad material	Stainless steel
Average fuel temperature (°F)	1500
Average clad temperature (°F)	614
Fuel enrichment	
Outer zone (%)	5.71
Inner zone (%)	2.82
Control rods	
Number	12
Type	Cruciform
Material - poison section	Boron-stainless (1.5% B^{10})
Follower section	Zirconium
Normal Operating power MW(t)	50
Average heat flux, Btu/(hr)(sq ft)	1.67×10^5
Maximum heat flux, Btu/(hr)(sq ft)	4.00×10^5
Physics	
K excess (cold), (%)	12.8
K excess (hot), (%)	5.7
Shutdown reactivity (cold), (%)	11.6
End of life peak-to-average power distribution	
Axial	1.5 (approximately)
Radial	1.6 (approximately)
Core lifetime	1600 efph
Core area ft^2	9.96
Reflector	H_2O
Average coolant temperature (°F)	540
Average coolant rise in core (°F)	31
Primary coolant system pressure (psig)	2330

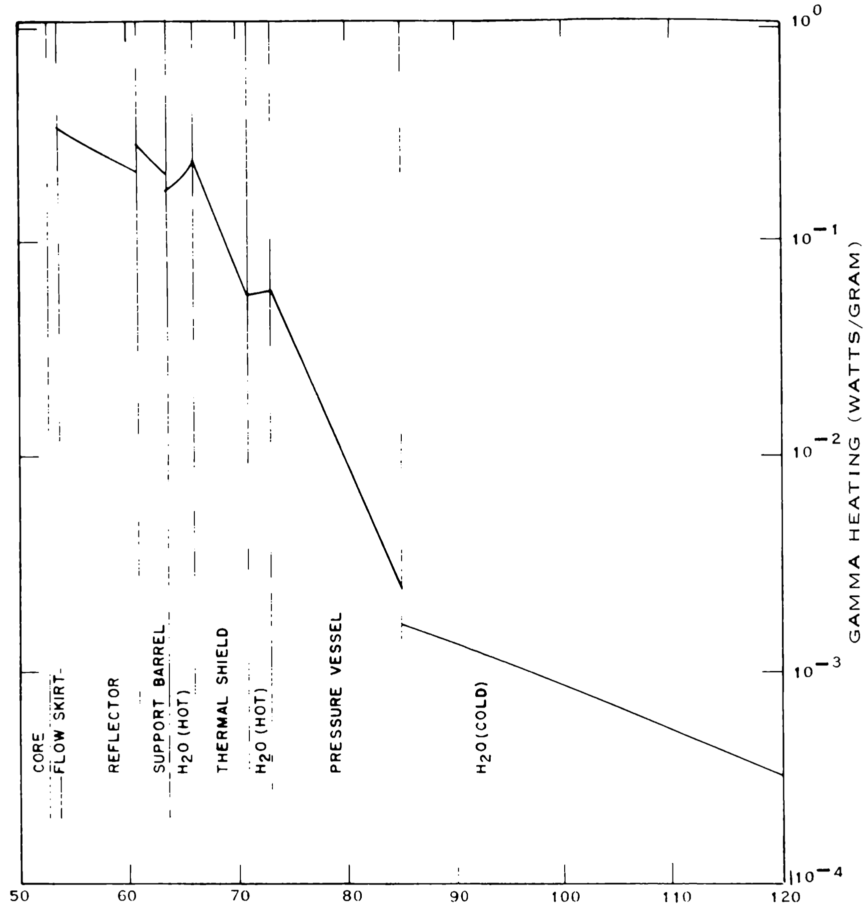


FIG. 12 LOFT GAMMA HEATING VERSUS DISTANCE FROM CORE.

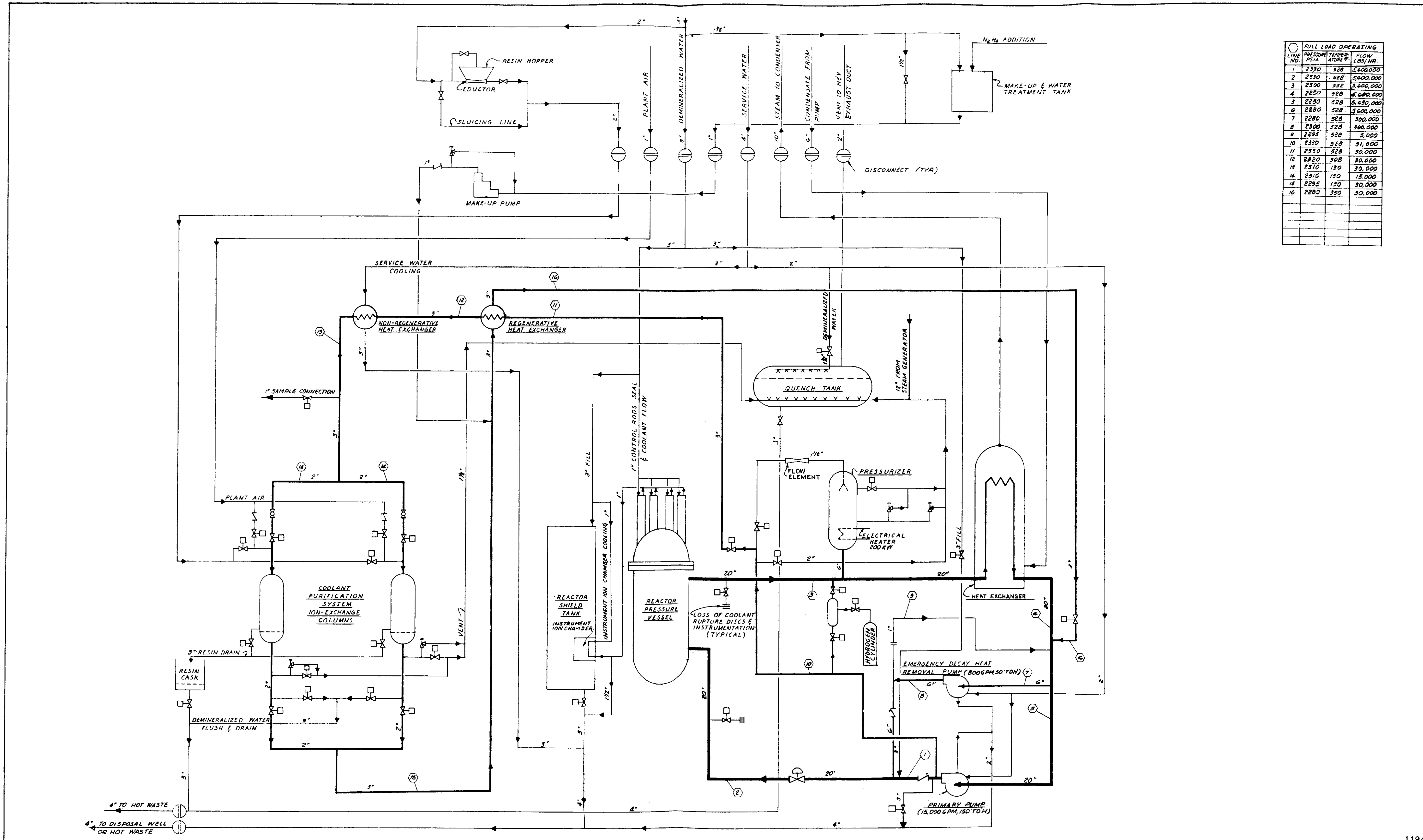
indicate that the basic design as presented here can meet the requirements of the LOFT program.

3. EXPERIMENTAL HEAT REMOVAL SYSTEMS

3.1 Primary Coolant System

3.11 System Description. The LOFT primary coolant system, shown in Figures 13 and 14, is a closed loop of circulating water under high pressure. Its function is to extract 50 MW of heat from the reactor core and to transfer this heat through a heat exchanger to the secondary coolant. High pressure is necessary to prevent water from boiling at the relatively high temperatures.

The primary loop is composed of the reactor pressure vessel, the tube side of the heat exchanger, the primary coolant pumps, and the connecting piping. Under normal operating conditions, demineralized water enters the reactor pressure vessel at approximately 528°F. Most of the primary water passes upward through the core, while a portion passes between the core and the pressure vessel wall. In passing through the reactor vessel, the water temperature is raised to approximately 552°F while the operating pressure at this temperature reaches 2310 psig.



LINE NO.	FULL LOAD OPERATING PRESSURE PSIA	TEMPERATURE °F	FLOW LBS/HR.
1	2330	528	5,600,000
2	2330	528	5,600,000
3	2300	552	5,600,000
4	2280	528	5,600,000
5	2280	528	5,630,000
6	2280	528	5,600,000
7	2280	528	300,000
8	2300	528	300,000
9	2295	528	5,000
10	2330	528	31,000
11	2330	528	30,000
12	2320	308	30,000
13	2310	130	30,000
14	2310	130	15,000
15	2295	130	30,000
16	2280	350	30,000

FIG. 13 LOFT PRIMARY SYSTEM FLOW DIAGRAM.

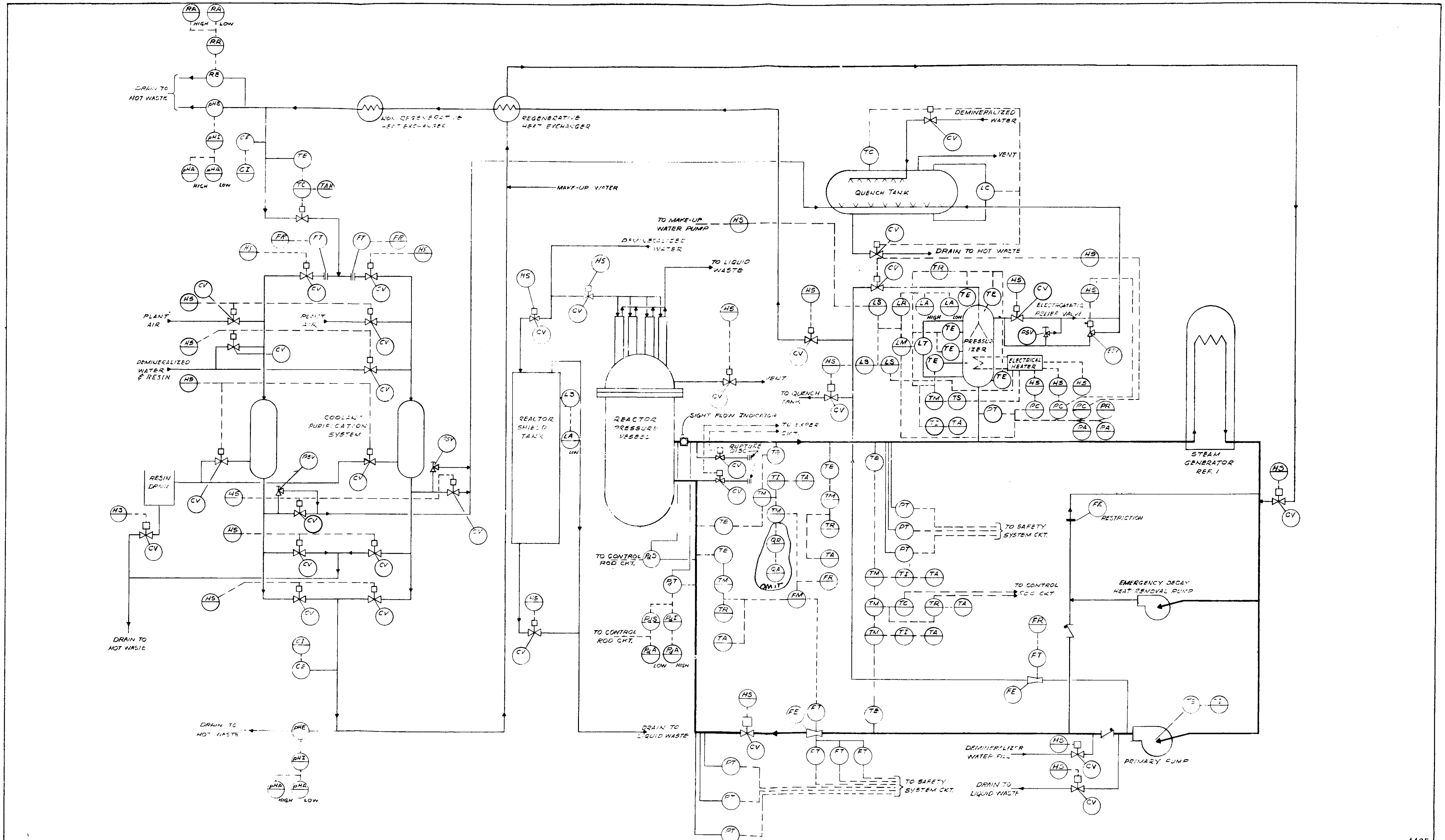


FIG. 14 LOFT PRIMARY SYSTEM INSTRUMENTATION.

The water then leaves the pressure vessel, passes through the tube side of the heat exchanger, and returns to the primary coolant pump. The primary coolant pump forces primary water through the loop at a rate of approximately 15,000 gpm (5.6 x 10^6 lb/hr) and delivers 50 MW of heat to the secondary coolant. In the heat exchanger, water gives up its heat to the secondary coolant on the shell side. This heat removal reduces the primary water temperature to approximately 528°F and delivers 50 MW of heat to the secondary coolant.

Auxiliaries included within the primary coolant system are: an emergency decay heat removal pump, a coolant purification system, a chemical additive and water make-up system, pressurizer system, and experimental blowdown nozzles.

The pressurizer system maintains the primary loop pressure at approximately 2330 psig for nuclear operation and 2500 psig for nonnuclear operation. The flow through the pressurizer is quite small; hence, the water in the pressurizer can be maintained at a different temperature from the bulk circulating coolant. Electrical immersion heaters keep the water in the pressurizer at the saturation temperature corresponding to the pressure; eg, approximately 669°F for 2500 psig. Rapid load fluctuations momentarily tend to change the bulk circulating water temperature resulting in a change in water volume. However, two phases exist in the pressurizer so that the steam volume will expand or contract to maintain a relatively constant pressure in the primary loop.

The emergency decay heat removal pump is used in the event the primary pumps fail. It operates continuously from the emergency power supply. It is connected in parallel with the primary pumps and circulates 800 gpm (3.0 x 10^5 lb/hr) when the primary pumps are shut down.

A portion of the coolant (approximately 70 gpm) from the primary coolant pump passes through a purification system. It passes through the tube sides of a regenerative and nonregenerative heat exchanger in series, ion exchange demineralizers (to remove both corrosion and fission products), a booster pump, the shell side of the regenerative heat exchanger, and then back to the suction side of the primary pump.

When the liquid level in the pressurizer reaches a predetermined level, a valve opens and permits some of the coolant to flow into a quench tank. System makeup water along with the additives required to maintain low oxygen levels or adjust pH are injected through the shell side of the regenerative heat exchanger by a high pressure makeup pump.

3.12 Primary Coolant System Components

(1) Primary Loop Piping. Primary loop piping is fabricated from firebox quality, carbon steel, seamless pipe. The piping is designed for a pressure of 2750 psig at 600°F. Pipe internal diameter is designed to limit water velocities from 25 to 30 ft/sec. Exterior surfaces are insulated to reduce the surface temperatures to 140°F. Thermal insulation is provided with a sealed, moisture-protecting jacket which resists the effects of the post-test decontamination environment in the containment building.

(2) Primary Coolant Pump. The primary coolant is circulated by a canned-motor centrifugal pump with a capacity of 5,600,000 lb/hr at approximately 2450 psia and 528°F. Total discharge head is equal to the sum of the

pressure drops through the reactor pressure vessel (20 psi), primary piping, primary heat exchanger, and control valve. The construction material is carbon steel. Design pressure of the pump casing is 2750 psig at 600°F.

(3) Primary Heat Exchanger. The primary heat exchanger is the shell and tube type and is designed to cool 5,600,000 lb/hr of demineralized water from 552 to 528°F, transferring the heat to the secondary coolant. The tube side containing the primary coolant has a design pressure of 2750 psig at 600°F. Construction material is carbon steel.

(4) Coolant Purification System. This system purifies the primary coolant by removing corrosion and fission product impurities. As can be seen in Figure 13, a portion of the primary coolant leaving the primary coolant pumps is cooled from approximately 528 to 130°F by passing through the tubes of a regenerative and nonregenerative heat exchanger arranged in series. The coolant then passes through one or both of two ion exchange demineralizer columns. The effluent from the ion exchange columns passes through a booster pump, the shell of the regenerative heat exchanger, and then to the suction side of the primary pumps.

The primary coolant purification flow rate is obtained from the following equation:

$$\frac{No}{N} = \exp\,(f/v)t$$

where

$\frac{No}{N}$ = activity reduction factor,

f = ion exchanger maximum flow (gpm),

v = primary coolant volume (gal), and

t = time (minimum).

The activity level in the primary coolant system following a fission break in the fuel cladding must be reduced by a factor of 100 in approximately six hours. This criterion is based on shutting the reactor down upon detection of the increased activity level and circulating the primary coolant through the bypass demineralizer until the system can be purged by demineralized water to the warm waste system.

The ion exchange resin in the demineralizer columns is of the mixed bed type with the anion resin in the hydroxide form and the cation resin in the lithium form. Lithium form cation resin is selected to provide high pH (9.0 to 10.5) effluent. Since no provisions are made for regeneration of the spent resin, it is removed from the columns by sluicing with demineralized water and replacing with fresh resin. Two columns are provided so spent resin from one column can be removed and replaced with fresh resin while the other column is in operation. Each of the ion exchange columns consists of a 14-inch ID with a 30-inch deep resin bed, and is rated at 35 gpm and 130°F (maximum).

The ion exchange booster pump is a single-stage centrifugal pump of the canned-motor type, with a capacity of approximately 70 gpm of demineralized water under suction conditions of 130°F and 2330 psig. The head developed by the pump at 70 gpm is equal to the sum of the pressure drops in the ion exchangers and the shell side of the regenerative heat exchanger.

The regenerative and nonregenerative heat exchangers are designed to jointly cool 70 gpm of primary coolant from 600 to 130°F and to reheat the demineralizer column effluent to at least 350°F before reinjection into the primary coolant loop. The nonregenerative heat exchanger is designed for demineralized water at the primary system design pressure on the tube side and service water on the shell side.

The regenerative heat exchanger is designed for demineralized water at full primary system design pressure in both the shell and tubes. Both the regenerative and nonregenerative exchangers are constructed of carbon steel.

(5) Chemical Addition and Water Makeup System. This system performs two functions:

(1) provides a means by which certain chemicals can be added for oxygen removal and pH control, and

(2) provides means by which demineralized water makeup can be added to the primary system.

Demineralized make-up water and additives for oxygen removal are introduced through the make-up pump to the shell side of the primary heat exchanger. Hydrazine is added for oxygen removal when the reactor is not operating at significant power. When the reactor is in operation, hydrogen is added to the system for oxygen removal. The make-up pump which is actuated by low liquid level in the pressurizer is capable of delivering 5 gpm at full primary system design pressure with the suction pressure at an atmosphere and 130°F.

(6) Experimental Blowdown Nozzles. Nozzles, equivalent to the cross sectional area of the primary coolant piping, are located on the inlet and outlet primary coolant piping as near as possible to the reactor pressure vessel exterior wall. Space is provided on the dolly for the rupture mechanism and expulsion flow measurement systems. A solid energy dissipator is mounted downstream from the blowdown nozzles to preclude damage to the containment building or experimental equipment.

(7) Primary Piping Ports, Sample, and Instrumentation. Instrumentation ports are provided in the primary coolant piping for measuring the two-phase transient flow during the blowdown experiment. In addition, sample nozzles are provided for insertion and removal of material samples from the system in the TAN hot shop after the completion of the loss of coolant experiment.

4. INSTRUMENTATION (Table VIII)

4.1 Reactor Instrumentation and Control

The reactor control system is less complicated than systems for typical pressurized water reactors essentially because manual control is provided. Instrumentation within the containment shell which must survive test conditions is designed to withstand the environmental conditions within the test area, eg, temperature, pressure, steam, moisture, radiation, and decontamination solutions.

4.11 Reactor Control. Reactivity control is accomplished with mechanical control rods which may be withdrawn from or inserted into the core. Reactor control embraces three phases: (a) manual control to accomplish start-up, criticality, and operational power level, (b) control for adjustment of reactivity to compensate for short-term fluctuations at operational power level, and (c) safety system actions to minimize damage to the facility resulting from operator and equipment malfunctions.

(1) Manual Control. A reactor control console is provided with the necessary instruments and switches to permit a trained operator to control each of the reactivity control devices. Increase of reactivity is manually controlled by interlocks requiring certain monitoring instrumentation to be operating on scale and in proper ranges. Manual control overrides the temperature limiting control but is subject to being overridden by the safety system. Manual control is used to accomplish all long term reactivity adjustments.

(2) Temperature Limiting Control. Average core coolant temperature indication and dead band limiting control are provided for operation at power. The range of dead band control is expected to account for little more than the short term fluctuations and fuel burnup occurring 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 is adequate for maintaining constant power level. The operator selects the "control region" within the core.

4.12 Reactor Power Instrumentation. Instrumentation is 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, ie, start-up, manual operation at power, dead band limiting operation at power, and safety system operation. Nuclear safety instrumentation is shown in Figures 15 and 16.

(1) Start-up Range. The start-up instrumentation consists of two low-level log count-rate channels. Monitoring of the neutron flux is possible in the cold, clean reactor at source level. A minimum count rate is required prior to withdrawing the control rods for reactor start-up. These channels range from source level to within four decades of the power range. The control rods are interlocked to prevent withdrawal unless both channels are on scale.

Provisions are made in the water filled reactor shield tank for inserting the start-up detectors.

(2) 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 power level

TABLE VIII

LOFT EXPERIMENTAL INSTRUMENTATION

Measurement	Detector	Location	Number of Measuring Points	Termination Station(s)
In-Core				
Nuclear	B Current Neutron Detectors	Assembly Centers	114	B,C
Temperature	Pt-Pt-Rh Tc's	Fuel Material (Experimental)	60	B,C
	Pt-Pt-Rh Tc's	Fuel Clad (Experimental)	90	B,C
	Pt-Pt-Rh Tc's	Assemblies Centers	19	B,C
	Pt-Pt-Rh Tc's	Between Assemblies	12	B,C
	Pt-Pt-Rh Tc's	Shroud	8	B,C
	Pt-Pt-Rh Tc's	Control Rods	4	B,C
Pressure	Diaphragm Type	In-Pins - Diameter Diagonal	20	B,C
	Tubes	Coolant in Assemblies	30	B,C
	Tubes	Between Assemblies	18	B,C
Velocity	Probes	Inlet to Core	8	B,C
	Probes	Outlet to Core	8	A
Strain	Strain Gauges	Pins	60	B,C
	Structural	Fuel Grids	21	A,B,C
		Total In-Core Measuring Points	472	
In-Vessel				
Nuclear	Neutron and Gamma Chambers	Above and Below Core	13	A
Temperature	Pt-Pt-Rh and Chromel-Alumel Tc's	Core Flow Skirt	12	A,B,C
		Inner and Outer Thermal Shield	24	A,B,C
		Upper and Lower Grid Support Barrel	16	A,B,C
		Upper and Lower Support Grid	18	A,B,C
		Instrument Grid	8	B,C
		Water Baffles	6	A
		Vessel Walls, Head and Bottom	62	A,B,C
Pressure	Tubes - Transducer	Axial Positioning	4	A,B,C
Strain	Strain Gauges	Skirt	12	A,B,C
		Instrument Grid	4	A,B,C
		Upper and Lower Support Barrel	18	A,B,C
		Inner and Outer Thermal Shield	6	A,B,C
		Vessel Walls, Head and Bottom	24	A,B,C
Vapor	Calorimeter	Top and Bottom	3	A,B,C
Coolant Density		Top and Bottom	3	A,B,C
Liquid Level	Conductivity Probes	Vessel Bottom	2	A
Condensation		Vessel Bottom	2	B,C
		Total In-Vessel Measuring Points	237	
In-Primary Coolant System				
Nuclear	Gamma and Neutron Chambers	Reactor Shield Tank	3	A
Temperature	Chromel-Alumel Tc's or Resistance Thermometers	Inlet and Outlet of Reactor Vessel	12	A
		Shield Wall	12	A
		Reactor Vessel Outer Wall	90	A
		Insulation	12	A
		Coolant Piping	12	A
Strain	Strain Gauges	Vessel Outer Surface	90	A
Pressure	Transducers in Tubes	Coolant Piping	20	A
Velocity	Hot Wire Anemometer	In-Coolant Piping	30	A
Mass Flow	Orifice	In-Pipe Space	2	A
	Load Cell	At Rupture	4	A
		Water Tank - Intr. Outlet Nozzle	36	A
		Along Pipe Length Outlet to Vessel	10	A
Coolant Density		At Rupture - Inlet and Outlet	3	A
Thrust	Load Cell	At Rupture - Inlet and Outlet	3	A
Vapor State	Calorimeter	In-Pipe Space	4	A
Condensation	Electrical Contact	In-Pipe Space	4	A
Convection	Anemometer	In-Pipe Space	2	A
		Total In-Primary Coolant System Measuring Points	349	
Inside Containment Shell				
Nuclear	Neutron and Gamma Chambers	X, Y, and Z Axes	20	D,E,F,G
Temperature	Copper Constantan	Walls - Dome - Concrete	200	D,E,F,G
Pressure	Microphones	X, Y, and Z Axes	26	D,E,F,G
Strain	Strain Gauges	Flatcar	14	A
Visual	TV and Frame Cameras	Penetrations of Walls	4	D,E,F,G
Convection	Anemometer	X, Y, and Z Axes	4	D,E,F,G
		Total Inside Containment Shell Measuring Points	264	
External to Containment Shell				
Pressure	Microphones	Radial to 300 ft	6	L,M
Strain	Strain Gauges	Vessel Wall and Dome - Door	137	K
Temperature	Copper Constantan	Walls - Dome	250	K
		Total External Containment Shell	393	

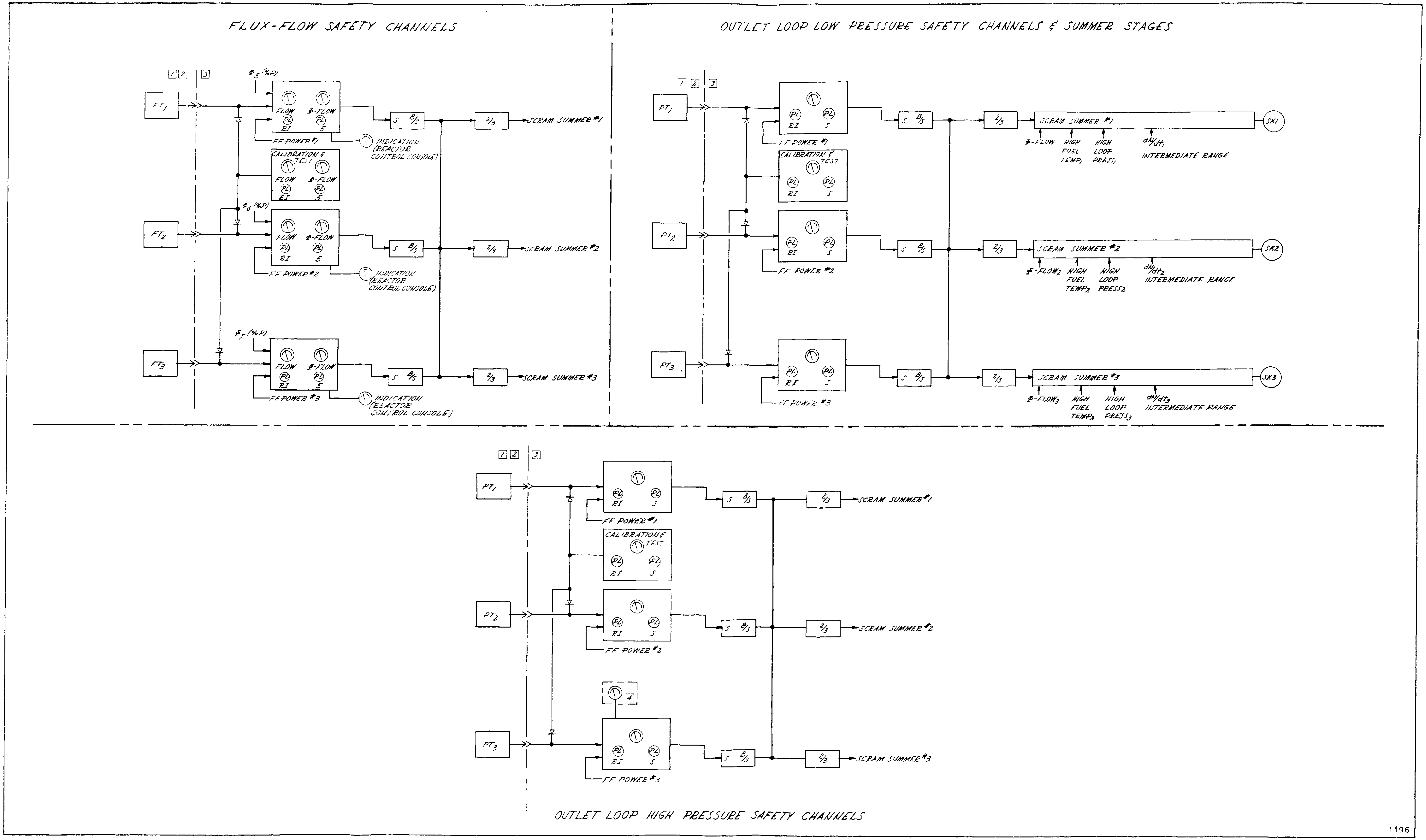


FIG. 15 REACTOR SAFETY INSTRUMENTATION DIAGRAM.

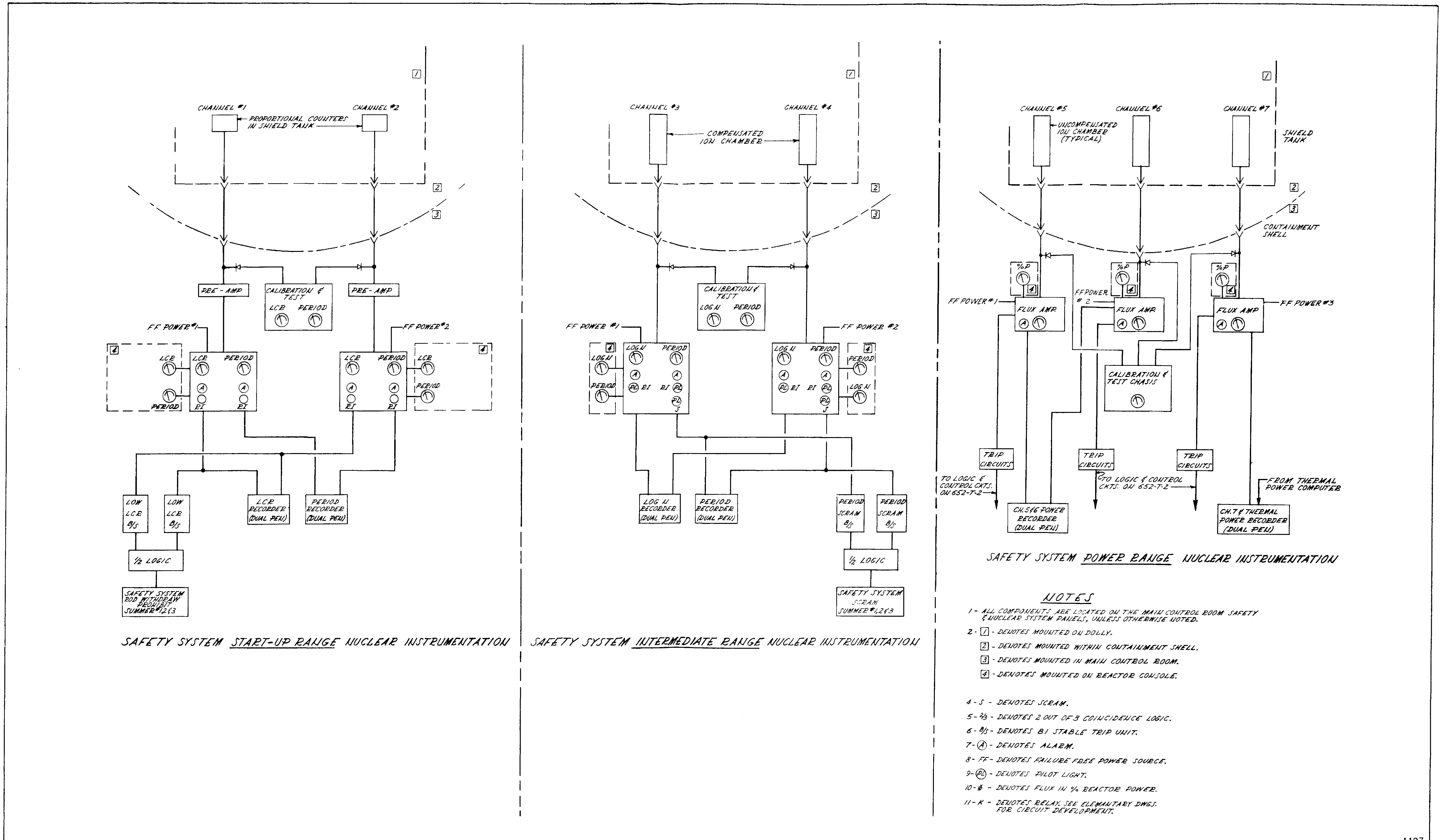


FIG. 16 REACTOR NUCLEAR AND CONTROL INSTRUMENTATION DIAGRAM.

coverage for approximately seven decades and overlap the upper portion of the source range and the power range. A boron-lined compensated ionization chamber is used for the detection of thermal neutrons. The log current and period computer converts 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 will be transmitted to indicators and recorders in the control room. Installation of intermediate range channels is similar to start-up channel placement.

(3) Power Range. This system is composed of three identical channels that monitor the neutron flux level from 1 to 300 percent 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 reactor shield tank. The installation of power range channels is similar to that for the start-up and intermediate range channels.

Calorimetric instrumentation provides a reference measurement of reactor power. Coolant flow is monitored together with reactor vessel inlet and outlet temperatures. From this information, power level may be manually computed and used to calibrate power range neutron flux instrumentation.

All of these indications as well as level and period information from the safety instrumentation are at the disposal of the console operator to facilitate start-up and attainment of operational power levels.

4.13 Reactor Safety System. The safety system monitors all system parameters which relate to the safety of personnel and facilities, and automatically takes appropriate action when required. All operational safety system instrumentation is assured uninterrupted performance by providing multiple power sources. At least two independent indications are available for each measured parameter. Nuclear safety parameters include power level, period of e-fold power multiplication, and core temperature. Process parameters include coolant temperatures, coolant pressures, and coolant flows. Action of the safety system is appropriate to protect the facility as completely as possible against all eventualities except the "planned accidents" for which it is designed. This reactor is typical of power reactors which are generally difficult to restart without refueling if a shutdown occurs in an advanced stage of core burnup. For this reason, the protective 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 potential hazards will first be communicated by audible and visible alarms to alert the reactor operator. The safety system incorporates instrumentation to provide nuclear information, process information, and miscellaneous information that is not dependent upon inputs from other monitoring systems.

Neutron pulse chambers and gamma-compensated ionization chambers described above 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 percent 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 of which the outputs are proportional to the derivatives of the logarithmic outputs. Trip set points are provided to give alarms, interlocks, and scrams at various unsafe levels of power and period.

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

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

4.2 Process Instrumentation

Remote operation of the reactor from the control and equipment building is essential. Therefore, the plant instrumentation is designed to permit operation of certain plant equipment and control of certain process variables from a main control room. The equipment operated from the main control room is that necessary for safe nuclear plant operation and includes radiation detection, containment isolation circuits, and the primary and secondary coolant system controls. Other process variables are recorded and displayed on local panels at appropriate locations. Since the auxiliary or process instrumentation is fairly standard and straight forward as related to reactor operation and safety, it is not detailed here.

4.3 Experimental Instrumentation

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

(1) amplitude accuracy

(2) event-time accuracy

(3) time-history accuracy

(4) dynamic range

(5) spatial gradient coverage.

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

(1) transducing accuracy

(2) response time

(3) dynamic or useful range

(4) environmental sensitivies.

The information contained in the following text and in the applicable drawings reflects the preceding discussion as applied to a preliminary review of the needs for the LOFT experiment, specifically with regard to the number and type of cables and termination facilities required.

The wide variation of conditions that will be imposed upon the detectors during normal operations and blowdowns requires that the occurrency and dynamic range of detectors be adapted to the particular measurements that are to be made. The measurements are divided into five major categories according to the location of the sensing device. These categories are: (a) in-core, (b) in-vessel, (c) primary coolant system, (d) inside the containment shell, and (e) exterior to the containment shell. To provide optimum flexibility in the final allocation and installation of all measurement detectors, termination stations are at various locations on the flatcar and in various areas about the test area. Figure 17 shows stations which provide for a variety of cable types and numbers to accommodate the nuclear, temperature, pressure, strain, and flow detectors in the local area of the station. From these stations the appropriate cables run to a master "pressure barrier termination", and from there to the Experimental Instrumentation room in the Control and Equipment building.

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

A description of the detector types, number of detectors, and the location of the operational and experimental measurement instrumentation is presented in the following sections. A summary of the operational and experimental measurements is presented in Table VIII and the termination stations in Figure 17.

4.32 In-Core Measurements

(1) Nuclear. In-core nuclear measurements are made during start-up, operational power runs, and during the loss of coolant accident. Power, period, and flux measurements are taken. Individual neutron detectors are located at sufficient positions throughout the core to obtain radial and axial flux profiles. The radial positioning of the detectors completely maps one octant, one diameter, and one diagonal. Spot detectors are located at selected positions in the remaining sections of the core to ascertain the extent of symmetry, to provide multiple checkpoints for the smallest section of symmetry, and to give a flux measurement in each quadrant of the core.

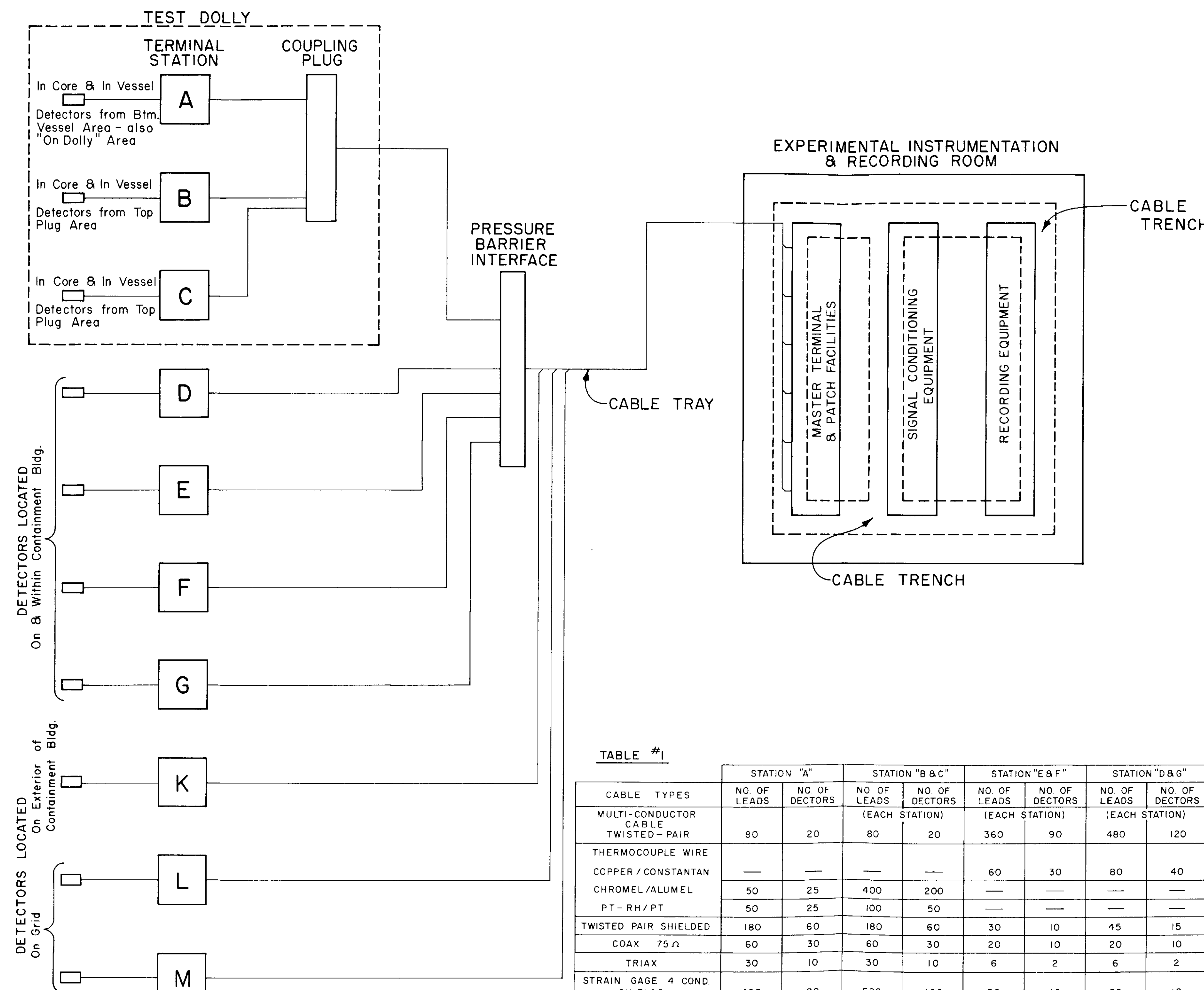


TABLE #1

CABLE TYPES	STATION "A" NO. OF LEADS	STATION "A" NO. OF DECTORS	STATION "B & C" NO. OF LEADS	STATION "B & C" NO. OF DECTORS	STATION "E & F" NO. OF LEADS	STATION "E & F" NO. OF DECTORS	STATION "D & G" NO. OF LEADS	STATION "D & G" NO. OF DECTORS	STATION "K" NO. OF LEADS	STATION "K" NO. OF DECTORS	STATION "L & M" NO. OF LEADS	STATION "L & M" NO. OF DECTORS
			(EACH STATION)		(EACH STATION)		(EACH STATION)				(EACH STATION)	
MULTI-CONDUCTOR CABLE TWISTED-PAIR	80	20	80	20	360	90	480	120	800	200	80	20
THERMOCOUPLE WIRE												
COPPER / CONSTANTAN	—	—	—	—	60	30	80	40	180	90	20	10
CHROMEL / ALUMEL	50	25	400	200	—	—	—	—	—	—	—	—
PT-RH / PT	50	25	100	50	—	—	—	—	—	—	—	—
TWISTED PAIR SHIELDED	180	60	180	60	30	10	45	15	15	5	60	20
COAX 75 Ω	60	30	60	30	20	10	20	10	4	2	20	10
TRIAX	30	10	30	10	6	2	6	2	9	3	—	—
STRAIN GAGE 4 COND. SHIELDED	400	80	500	100	50	10	50	10	—	—	—	—
TOTAL	850	250	1350	470	526	152	681	197	1008	300	180	60

GENERAL NOTES:-

1. EACH TERMINAL STATION SHALL BE A WEATHER-TIGHT ENCLOSURE WITH A DRY AIR PURGE TO MAINTAIN POSITIVE DIFFERENTIAL PRESSURE WITH RESPECT TO CONTAINMENT BLDG. PRESSURE.
2. EACH TERMINAL STATION SHALL BE SIZED TO ACCOMODATE SEVERAL SIGNAL PREAMPLIFIERS IN ADDITION TO THE CABLE TERMINATIONS REQUIRED FOR CABLES LISTED IN TABLE #1.
3. 28 VOLT D.C. POWER FOR LOW WATTAGE HEATERS WILL BE REQUIRED IN EACH TERMINAL STATION.

REFERENCE DRAWINGS:-

TAN-E-0636 TAN LOFT STEP - EXPERIMENTATION INSTRUMENTATION - TERMINATION STATION LOCATION SECTION "B-B".

TAN-E-0637 TAN LOFT STEP - EXPERIMENTATION INSTRUMENTATION - TERMINATION STATION LOCATION SECTION "C-C".

TAN-E-0638 TAN LOFT STEP - PRESSURE BARRIER & INSTRUMENTATION LOCATION - SECTION "D-D".

0635

FIG. 17 EXPERIMENTAL TEST INSTRUMENTATION.

The cross-arms of the cruciform control rods are slightly shorter than the width of the adjacent fuel assemblies so that a space exists of sufficient size to accommodate small nuclear chambers which are inserted in these spaces.

(2) Temperature. Experimental fuel, clad, coolant, and structural temperature measurements are made throughout the core. Experimental temperature measurements are made in the fuel at vertical positions diagonally across the core and on the cladding at vertical positions along one diameter, one-half diagonal, and at selected points in each quadrant. Coolant temperatures at inlets and outlets of the assemblies and in assemblies are measured. Thermocouples located in assemblies along one diameter are used to determine the coolant temperature in the center of the fuel pin array. Structural temperature measurements are made on the assembly shrouds in spaces provided at the ends of the control rod cross-arm.

Experimental fuel, clad and structural temperature measurements in the molten region ($\approx$ 5000°F) will be attempted by appropriate sheathed developmental tungsten-tungsten-rhenium thermocouples. These measurements are made in positions similar to those for previously mentioned experimental temperature measurements and additionally in the region of the bottom head.

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

(4) Strain. In-core strain measurements are made by strain gauges positioned on pins located in assemblies along one diameter, along one diagonal, and at selected points in each quadrant. Several gauges are attached to the assembly shrouds. Structural strain measurements are made on the upper and lower grids to determine the vibrational and thermal stresses due to blowdown and heat-up.

(5) Velocity. Coolant velocities due to blowdown and to transient heat-up are determined at inlet and outlet positions along one diameter and at positions along coolant channels between fuel assemblies. These velocity measurements require the use of detectors which can continue to operate under high temperature conditions. Measurements are made at the inlet and outlet of the core to determine the normal coolant flows before the loss of coolant accident.

4.33 In-Vessel Measurements

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

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

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

Since both operational and experimental instrumentation is provided, the number of thermocouples required becomes significantly large; therefore, the number and positions of the leadout ports are given special consideration.

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

(4) Strain. The strains on the vessel and internal structures resulting from vibrations during blowdown, thermal expansion resulting from the transient heat-up, steam pressure, and the weight of supported components of the core are measured by strain gauges.

Strain gauges are positioned on the flow skirt and on the instrument lead and shock absorber support barrel. They also are located on the upper and lower support barrels, inner and outer thermal shields, vessel walls, and at top and bottom heads. Longitudinal, radial, circumferential, and diagonal stresses are determined at appropriate positions.

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

(6) Coolant Quality. Coolant density measurements will be made to determine the coolant quality.

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

4.34 Primary Coolant System

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

(2) Temperature. Thermocouples are attached to the outer vessel surface, vessel insulation, in the water shielding, and on the reactor shield tank at center, top, and bottom positions. The coolant piping is instrumented with thermocouples along the entire length of the piping at positions on the inner and outer surfaces, and on system components including valves and pumps. Other temperatures are taken at the inlet and outlets to the reactor pressure vessel and on both sides of the rupture.

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

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

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

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

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

4.35 Inside Containment Vessel

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

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

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

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

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

4.36 Exterior to the Containment Vessel

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

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

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

4.4 Radiation Monitoring

Radiation monitoring provisions in the LOFT facility include fission break monitoring, stack gas monitoring, health physics instrumentation, and high range gamma dose-rate monitors. The fission break monitor measures direct radiation increase from a side stream of the primary coolant. A technique which measures the increase in count rate is satisfactory for this measurement, as compared with the integrating technique to be 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, also are available. There also are area radiation monitors to measure and indicate the radiation intensities at strategic locations throughout the LOFT plant area.

Approximately six high range gamma dose-rate monitors are installed inside the containment shell to indicate the general radiation level inside the building as a function of time. The outputs of the detectors are recorded inside the reactor control room. These detectors are intended to provide an early indication of the magnitude of the fission product release from the reactor pressure vessel.

4.5 Containment Vessel Isolation

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

VII. FACILITY DESCRIPTION

1. SITE DESCRIPTION

1.1 General

The Test Area North (TAN) facilities are located within the National Reactor Test Station (NRTS) boundary and are approximately 27 miles north-northeast of the Central Facilities area (see Figures 18 and 19). The Flight Engine Test (FET) facility is located approximately one mile northwest of the Technical Support Facilties (TSF). The Loss of Fluid Test (LOFT) containment and services buildings adjoin the FET control and equipment building at the latter's southeast wall.

1.2 Environmental Conditions

1.21 Geography [16]. The NRTS is located in southeastern Idaho on the Snake River Plain at the foot of the Lemhi, Lost River, and Beaverhead Centennial mountain ranges. From north to south the station extends 34 miles, and from east to west it extends 29 miles.

The general topography of the NRTS is essentially flat with a one percent slope from southwest to northeast at an average elevation of 5000 feet above sea level. Along the western boundary the mountains rise to an elevation of 10,000 feet. The elevation is lower along the Snake River to the west and southwest except for a gentle 600-foot rise midway between the site and the river. This rise runs parallel to the Snake River (see Figure 20).

From the valleys between the mountain ranges to the northwest of the area, the courses of the Big Lost River, Little Lost River, and Birch Creek run into playas or sinks, located in the north central section of the NRTS. The mountains encircle the Snake River Plain, and rise as much as 10,000 to 11,000 feet 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.

1.22 Population Distribution

(1) On-Site[17]. The working force at each of the NRTS facilities as shown in Table IX is variable, depending on the construction work in progress. The closest major facility to TAN is NRF which is 22 miles south-southwest and has a daytime population of 1533 with 200 nighttime personnel. The daytime population of TAN is approximately 365 with 50 nighttime personnel.

(2) Off-Site [18]. Populations of outlying communities are shown in Table X. Towns not appearing in this table are unincorporated and no census figures are available.

TABLE IX

POPULATION DISTRIBUTION OF NRTS

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

1.23 Meteorology [19, 20, 21]. The meteorological conditions at NRTS are described in detail in previous reports. These reports are used as a basis for the following data:

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

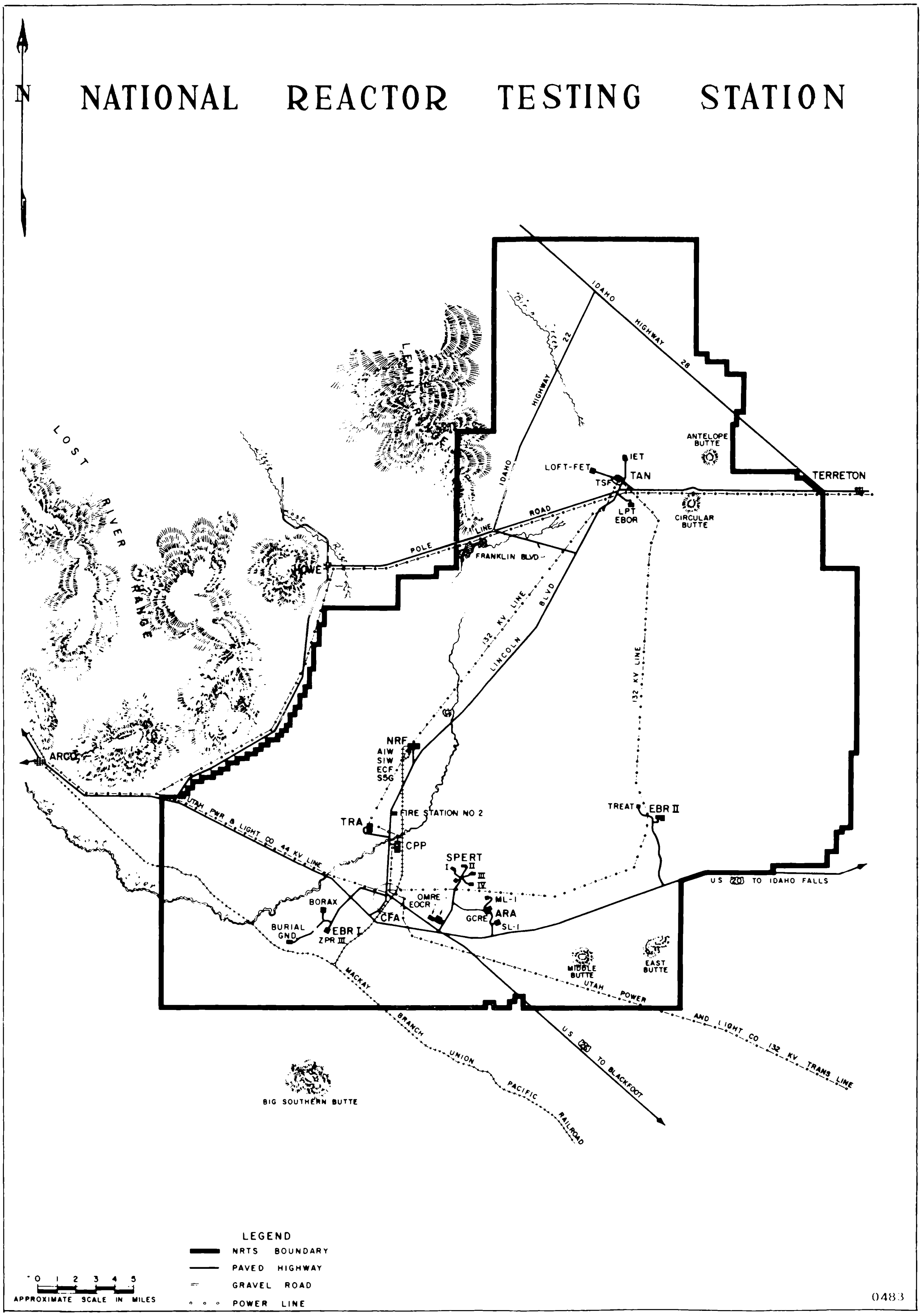


FIG. 18 NATIONAL REACTOR TESTING STATION.

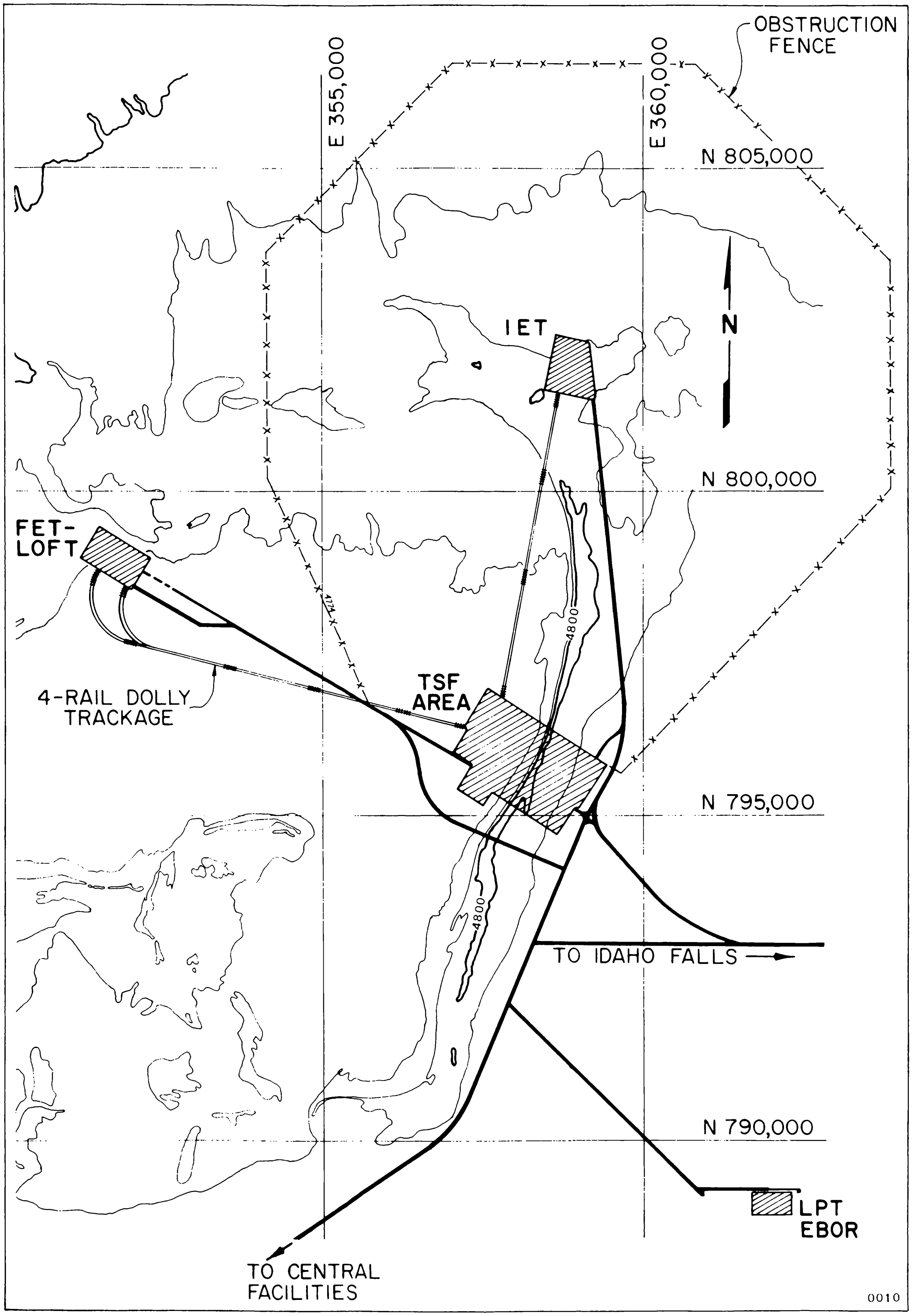


FIG. 19 TEST AREA NORTH, NRTS.

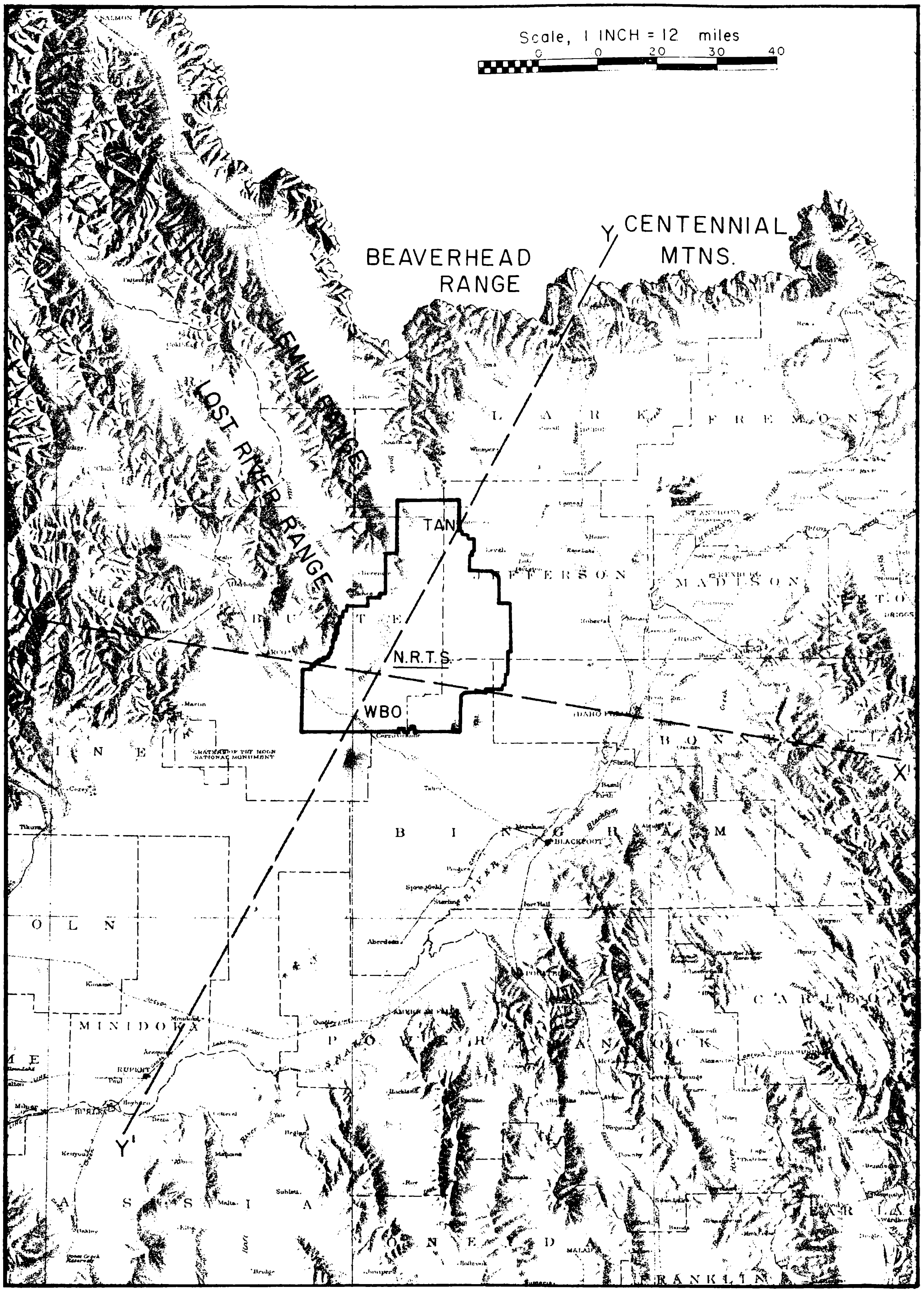


FIG. 20 NRTS AND SOUTHEASTERN IDAHO.

TABLE X

OFF-SITE POPULATION DISTRIBUTION

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

TABLE X (Cont.)

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

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

temperature of 103°F was recorded in July 1960. A minimum temperature at TAN was recorded at -43°F in January 1960.

(2) Precipitation. Precipitation for the NRTS, in general, averages 7.69 inches and appears in the form of rain, snow, sleet, and hail. As to the rain-snow distribution, snow has been experienced in every month except July, and rain can be observed at any time during the year. The greatest 24-hour fall of rain was 1.78 inches in May 1963, and that of snow, 8.5 inches in January 1957. The maximum rainfall recorded for any 24-hour period was 1.33 inches in July 1953.

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

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

(4) 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 more than 60 consecutive hours.

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

1.24 Geology [22]. This 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 three-inch aggregate.

The NRTS has no well-defined, integrated, surface-water drainage system, and it is not crossed by perennial streams. However, the NRTS overlays a natural underground reservoir of water, having an estimated lateral flow of not less than 500 cu ft/sec, or about 323 million gal/day.

The main sources of water for this reservoir are the streams that originate in the mountains to the north and disappear into the porous soils of the NRTS area. These sources of water include the Big Lost River, Little Lost River, Birch Creek, and also an additional source from Mud Lake Basin.

The altitude of the water table ranges from 4580 feet above sea level in the northern part of the station to about 4400 feet near the southwest corner. The water table in the TAN area is at a depth of 200 feet, 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.

1.25 Seismology [23]. The NRTS site is located in a region which "The Pacific Coast Uniform Building Code" (1949) designated as a zone 2 area. Although many recorded earthquakes have been felt in Idaho, none were of sufficient intensity to cause more than minor damage to buildings. Of the 14 recorded earthquakes with an epicenter within the state, seven had their epicenters within 100 miles of the TAN site, four to the southest 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 Arco area, the distances are so great that a zone 2 designation has been considered safe.

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

1.3 FET Facility Description

1.31 General. The Flight Engine Test (FET) facility (Figure 19), located approximately one mile northwest of the Technical Services Facilities (TSF), was selected by the Commission as the area in which to conduct the Loss of Fluid Test (LOFT) program. The LOFT containment building is integrated into the FET complex to utilize buildings, services, and utilities previously constructed for the Aeronautical Nuclear Propulsion (ANP) programs. In order to accurately describe the LOFT-FET test area and delineate LOFT utilization of the original facility, the following FET summary descriptions are presented. Additional buildings and services required in the LOFT program are described in the facility and utility section of this report.

The FET consists primarily of the test building and the underground control and equipment building (Figures 21 and 22). Other minor FET structures include two deep well pump houses, a tank house, and a 2-bay cooling tower. The test building is a reinforced concrete arched structure measuring 320 feet wide x 234 feet long x 99 feet high at the midpoint of the arc. Overlapping sliding doors 60 feet high open to a total width of 240 feet at either end of the hangar building. Four rail dolly trackage enters the building through the southwest door.

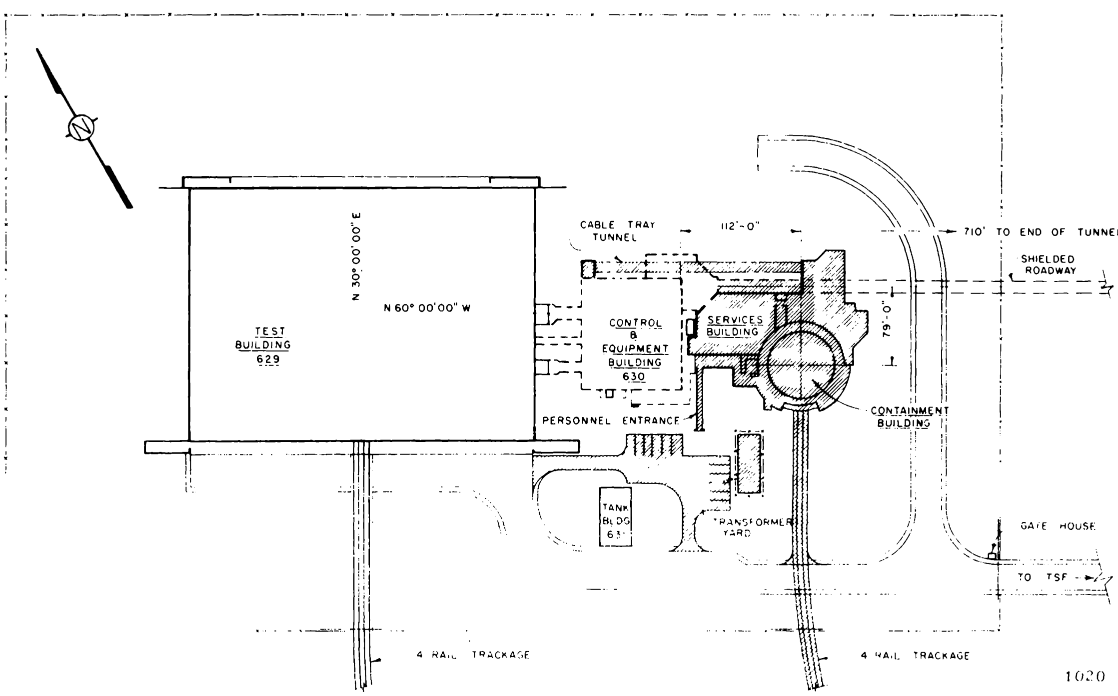


FIG. 21 LOFT-FET PLOT PLAN.

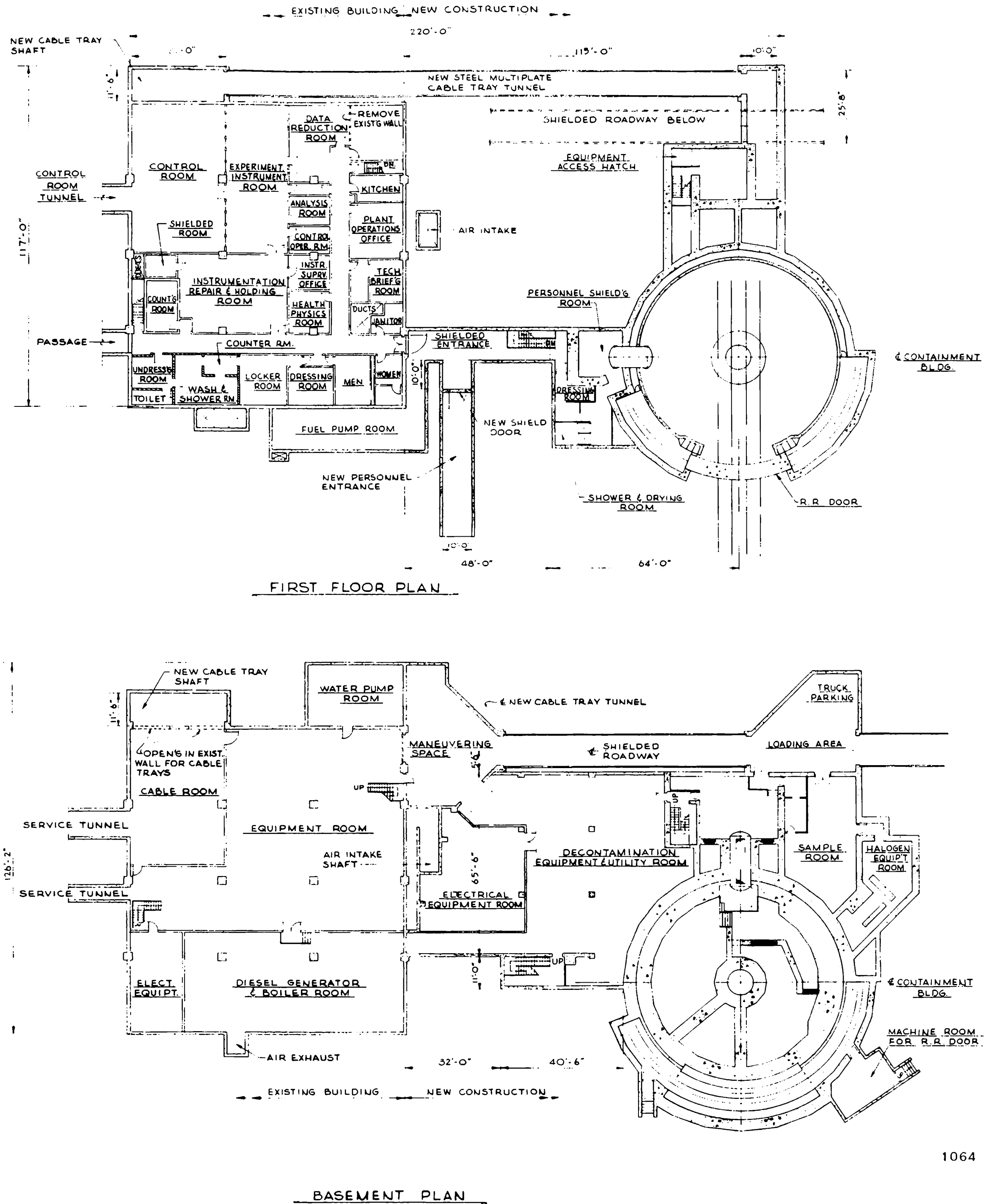


FIG. 22 LOFT SERVICES BUILDING.

The control and equipment building is a reinforced concrete building built underground and immediately adjacent to the test building. The building measures approximately 105 x 94 feet with walls and roof approximately one foot thick. There is eight feet of earth cover over the entire building.

The FET control and equipment building (Figures 21 and 22) houses all the support facilities and utilities for the operation of the FET test building. Located in various portions of the building are the control room and console, the necessary offices and laboratories, water treatment equipment, water circulation pumps, air compressor equipment, electrical switchgear, a diesel driven emergency electrical generator, and such other facilities as are needed to make the FET an integrally independent test facility.

Two large service and utility tunnels extend from the control and equipment building to a common tunnel under the floor of the test building. Air, water, electricity, and other utilities are carried through these tunnels for distribution throughout the facility.

1.32 FET Utilities

(1) Electrical

(a) Commercial Power. Commerical power is provided to the FET area by an overhead 13.8 kV transmission line one mile in length from the TSF area. Two transformers, one 1500 kVA and the other 225 kVA reduce this voltage to 480 V for distribution throughout FET. Circuit breakers and other control equipment are located in the motor control center in the control and equipment building. Transformers are located throughout the facility to further reduce the 480 to 120/208 volts for lighting and receptacle circuits.

(b) Emergency Power. Emergency power is provided by a 1000 kVA, automatic starting, diesel-driven generator which is located in the equipment building. The generator is connected to the instrument bus and other loads critical to the test in progress.

(2) Water

(a) Raw Water. Raw water is provided to the FET area by two-1000 gpm, 125 hp deep well pumps. These pumps supply water, which is automatically chlorinated, to a 500,000 gallon, surface mounted steel storage tank. It is then treated and distributed throughout the control and equipment and test buildings.

(b) Potable and Service Water. Potable and service water is provided to the FET area by tapping the 20-inch main from the storage tank. Two pumps rated at 500 and 100 gpm, respectively, transfer the raw water to a pneumatic pressurizing tank which maintains system pressure at approximately 125 psi regardless of pump cycling. Water from the pressurizer tank is piped throughout the plant area to be utilized as potable or service water and to pressurize the fire loop during nonemergency conditions.

(c) Fire Water. Fire water also is provided to the FET by tapping the 20-inch main from the storage tank. Four fire pumps, each rated at 1000 gpm, 75 hp, are connected in parallel to maintain fire water pressure and flow. These pumps are sequentially started by pressure switches as necessary, and are connected to both the "normal" and "emergency" power buses. Under normal conditions pressure is maintained in the fire loop by the pressurizer tank as described above. When the fire loop pressure falls to preset levels, the four fire pumps start sequentially.

(d) Cooling Water. Cooling water is provided for FET dolly mounted experimental equipment and for certain utility machinery. Items of equipment supplied include the diesel generator lube oil cooler, diesel jacket water cooler, air compressor intercooler and aftercooler, and the air conditioning refrigeration condensers. Additional makeup water is added from service water to the cooling water system automatically to maintain the liquid level in the surge tank.

(3) Air. Compressed air is provided to the FET by a two stage compressor driven by a 250 hp synchronous motor, and rate at one lb/sec at 200 psig. The output air from the compressor passes through an aftercooler, reserve tank, oil vapor remover, and an air dryer before utilization. Various pressure reducing valves reduce the pressure as follows:

(1) 40 psig for instrument air

(2) 150 psig for starting of emergency diesel generator

(3) 125 psig for water pressurizing tank

(4) 25 psig for use by the deionizer

(5) 90 psig for use by various remotely operated valves and the air operated coupler mounted in the dolly coupling area.

An auxiliary air compressor rated at 68 cfm at 250 psig and driven by a 15 hp motor is connected in the sytem and automatically starts and stops as the system pressure requires.

(4) Liquid Waste

(a) Hot Liquid Waste. Hot liquid waste is drained from the various areas of the FET complex to a 15,000 gallon hot waste storage tank which is buried south of the tank building. The waste liquid level is monitored and alarmed in the control and equipment building. When the accumulation of liquid waste dictates disposal, a determination is made of the radioactive level. If the waste is radioactive above a predetermined level, it is hauled by tank truck to the liquid waste evaporators at TSF for volume reduction and ultimate permanent storage. If the liquid waste is no longer "hot", it is pumped to grade and allowed to leach into the soil.

(b) Sanitary Waste System. The sanitary waste system at FET is of conventional design and is sized for an average occupancy of 50 people.

(5) Off-Gas. At the present time there is no system for the handling and removal of radioactive off-gas in the FET area.

(6) Heating and Ventilating Systems. Heated or evaporative cooled air is supplied for year-round temperature control to all personnel-occupied areas of the control and equipment building. Static pressures are maintained in the spaces by variable inlet dampers on the supply and return-exhaust air fans. Heat is supplied to the test building by steam-coil unit heaters located on each side of the building.

(7) Borated Water. Borated water is provided at FET for the emergency control of reactors. This system, which was formerly known as "System 33", consists of a 500 gallon additive mixing tank, two 12,000 gallon storage tanks, and interconnecting valves and piping. The solution in the additive tank is maintained at 140°F for maximum solubility. This concentrated solution is fed by a water-jet eductor into the dilute solution which is circulating and is then stored in storage tanks. The storage tanks are maintained at an average temperature of 70°F by steam coils to preclude crystallization. A heat exchanger is provided to maintain the temperature of the solution circulating through the piping and to the experimental dolly. The circulating pump is rate at 100 gpm.

(8) Miscellaneous. Other loops and systems are provided in the FET complex but will not be described in this report since they have no readily apparent value in the loss of coolant test program being discussed herein. Such systems include argon distribution, air sampling system in the test building, combustible vapor detector system, engine fuel system, etc.

1.4 Examination Area

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

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

1.42 Hot Shop. The hot shop is an area 160 feet long by 51 feet wide by 67 feet 6 inches high. It is surrounded by reinforced concrete shielding walls to provide a maximum dose rate of 6.25 mrem/hr in the operating galleries located on both sides of the long areas of the hot shop while containing an experimental assembly emanating 5.7 x 10^6 R/hr gamma. Located in both the upper and lower operating galleries are windows which provide visual observations 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. The door is operated by electric motors located outside the hot shop and controlled from the control console.

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

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

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

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

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

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

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

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

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.

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

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

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

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

1.46 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.

1.47 Hot Cells. The hot cell facility, consisting of four hot cells and miscellaneous work areas, was designed and built to satisfy the need for space in which post-irradiation examination of reactor fuel and mechanical components could be performed. The cell working space is divided into four equal areas, the type and degree of contamination that can be handled in each being slightly different.

1.5 Transportation System

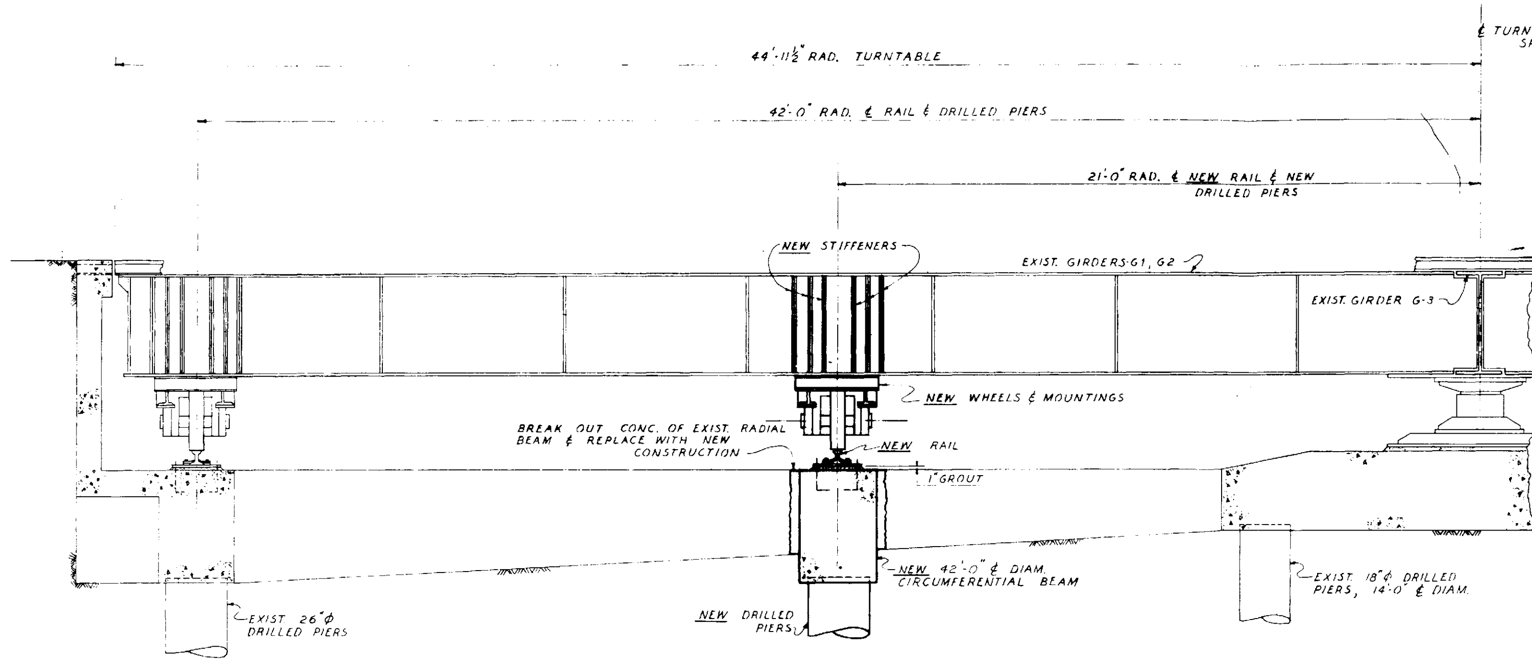
1.51 Trackage. Four-rail trackage connecting the original FET area and the hot shop was extended to the LOFT facility by adding a new turn out into the LOFT containment building. In adding the new section, fill and subgrade were compacted to 95 percent maximum density at optimum moisture content, and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

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

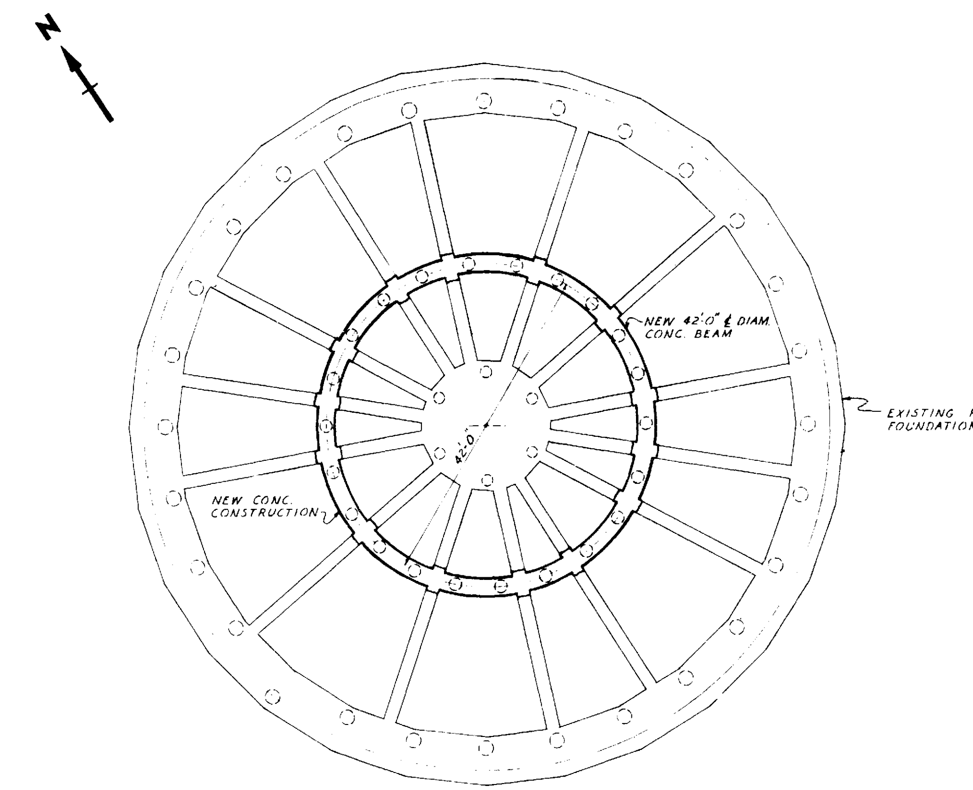
The turntable is a pit-mounted bridge type with full circular deck, approximately 90 feet in diameter with four rails spaced to provide three standard gauge tracks. The built-up plate main girder spans 42 feet from the spindle to the carriage rail.

Calculations indicated that the turntable was not capable of supporting a total load of 800 tons for the LOFT test components and the locomotive concurrently; therefore, the turntable was reinforced to handle the loads required in the LOFT tests by constructing an additional 42-foot diameter circumferential track complete with carriage assemblies (see Figure 23).

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



HALF SECTION THRU TURNTABLE & PIT FOUNDATION



FOUNDATION PLAN SHOWING NEW CONSTRUCTION

1214

FIG. 23 TURNTABLE MODIFICATIONS.

The dolly was designed for movement by the shielded locomotive (Figure 24) located at the TAN complex. Locomotive specifications are as follows:

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

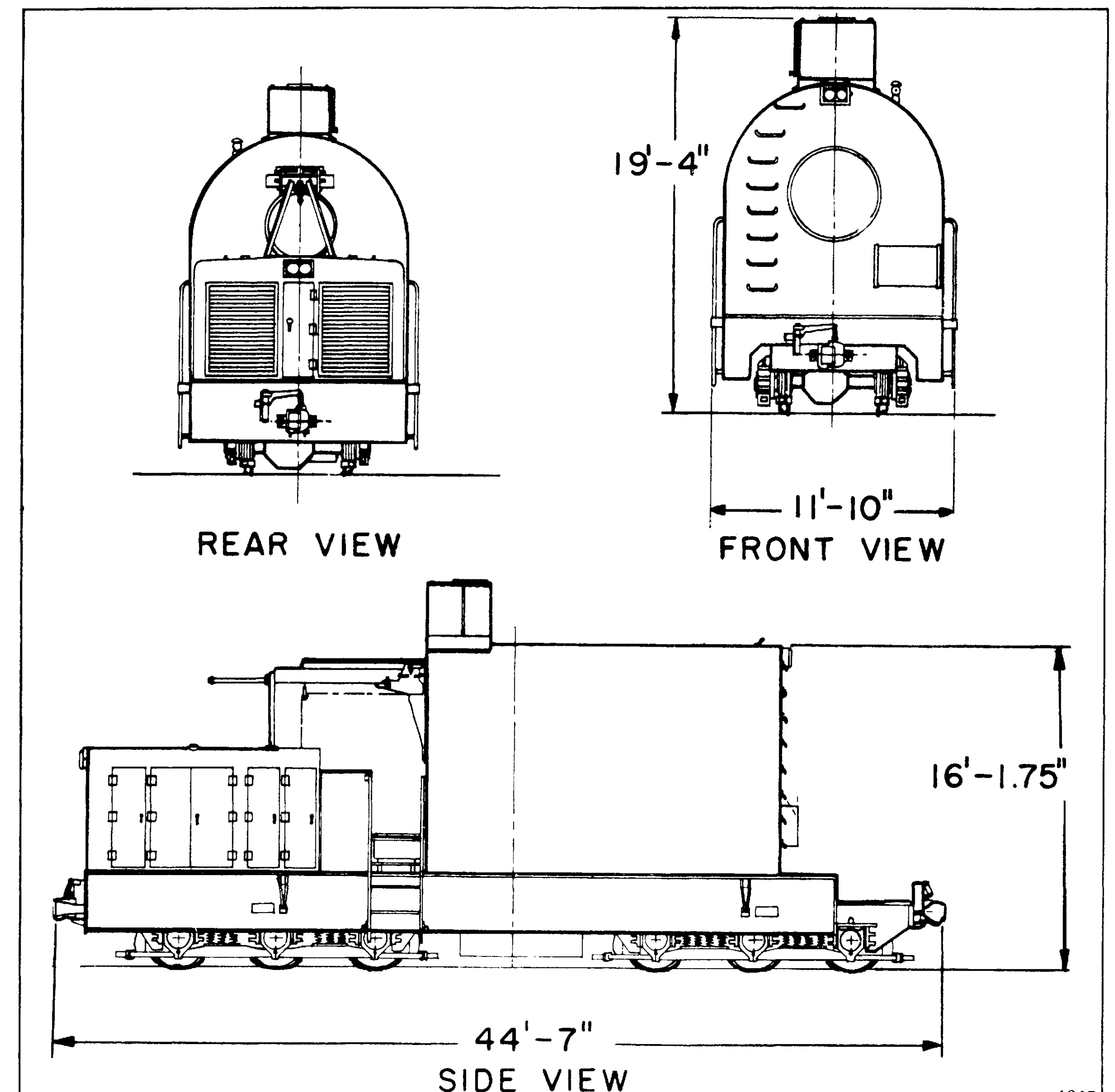


FIG. 24 TAN SHIELDED LOCOMOTIVE.

Ratings

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

Shieldings

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

Dimensions

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

2. LOFT CONTAINMENT BUILDING AND SUPPORTING FACILITIES
(Figures 25, 26, 27, 28, and 29)

2.1 Containment Building

The LOFT containment building is composed of three major sections: the containment vessel which is that part of the building above ground; the basement area which houses miscellaneous equipment; and the remote control room.

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

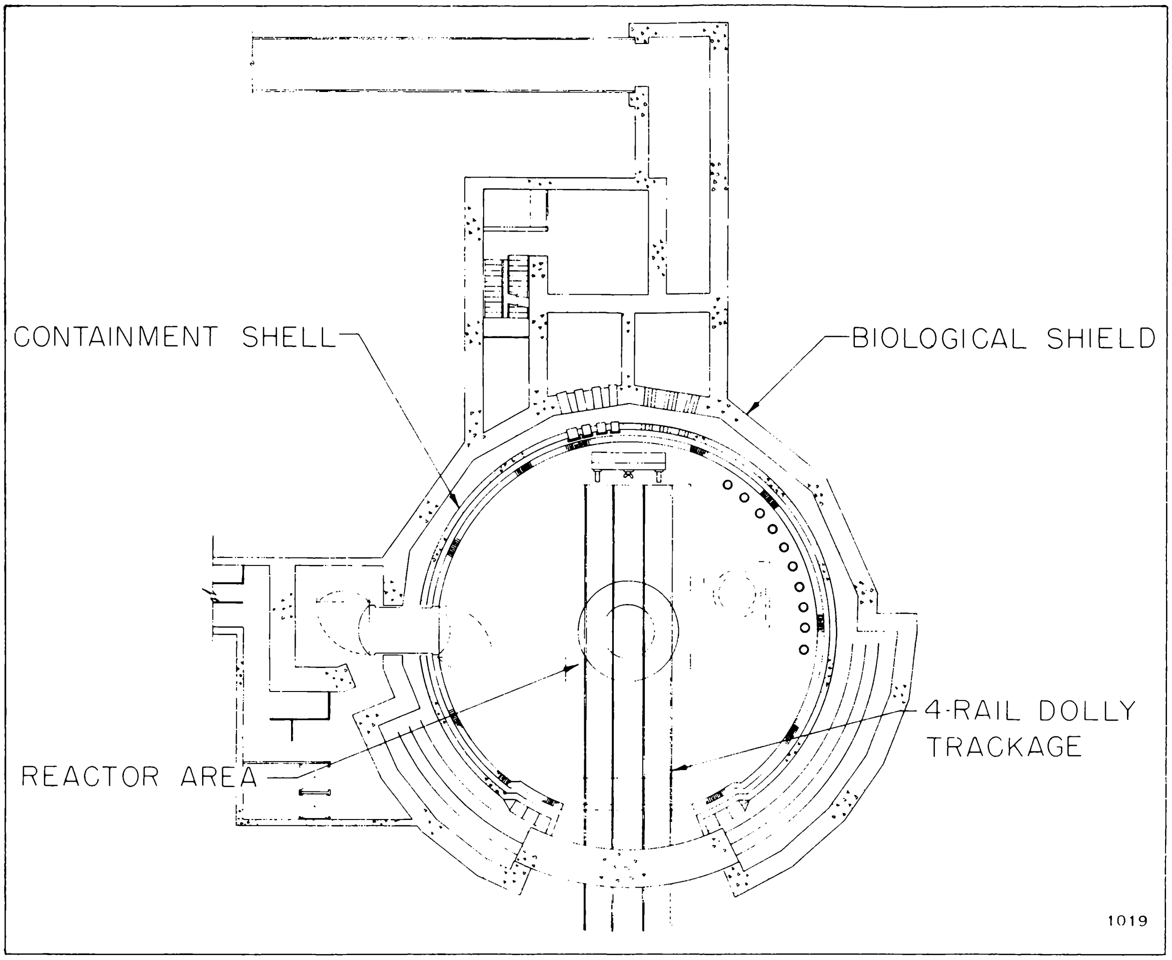


FIG. 25 CONTAINMENT BUILDING -- GROUND FLOOR PLAN.

and the STEP programs will permit maximum utilization of the services and equipment available in the FET.

2.11 Containment Vessel. The LOFT containment vessel is a cylindrical steel pressure vessel with a hemispherical top closure and a 2:1 semi-ellipsoidal bottom closure. The vessel is designed to withstand and contain an internal pressure of 40 psig. The vessel is 70 feet in diameter and measures 97 feet from grade level to the top of the dome. Approximately 32 feet of the containment structure is below grade. The vessel shell plate material conforms to ASTM specification A-300 for ASTM-A-212 Grade B material.

To prevent missiles from penetrating the steel containment vessel, the vertical interface of the vessel is lined with concrete 1 foot 0 inch thick which extends from the operating floor to the overhead crane. The containment vessel dome is lined with 6 inches of shotcrete. All internal surfaces are sealed with a protective coating to allow decontamination of the vessel.

The containment vessel volume and pressure design must simultaneously accommodate the loss of coolant experiment and the contents of the secondary coolant system. This will insure that the containment design pressure will be sufficient to withstand the pressure associated with the coolant blowdown which

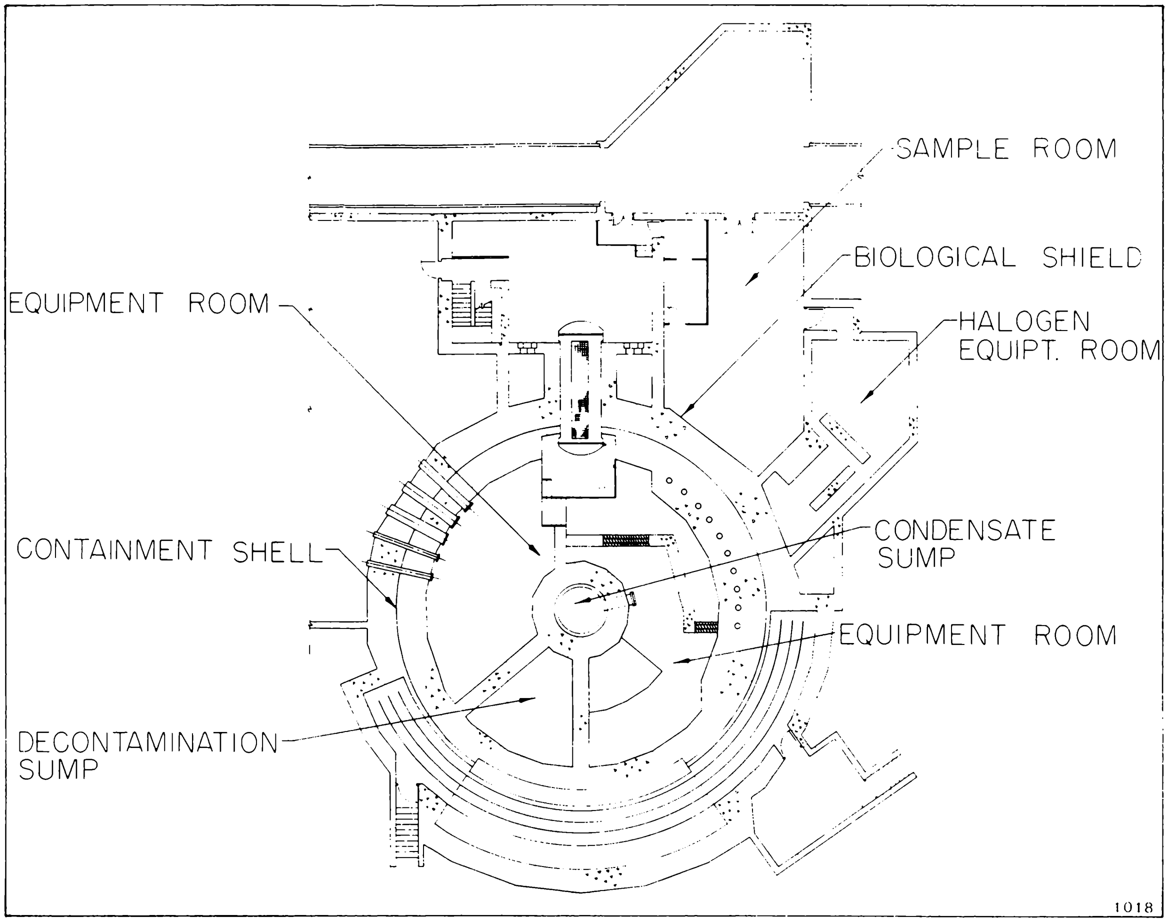


FIG. 26 CONTAINMENT BUILDING -- BASEMENT PLAN.

simultaneously ruptures the steam generator tube sheet from severe hydrodynamic forces.

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

Several openings into the vessel are required for access of the railroad dolly mounted experiment, personnel, miscellaneous equipment, piping, electrical conduits, etc. The largest opening is the 22-foot wide by 33-foot high railroad door opening (Figure 30). 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 4-rail dolly trackage enters the containment vessel at grade level, approximately 32 feet above the lowest point of the bottom closure. The tracks terminate near the wall opposite the railroad door. The railroad door is a steel framed door filled with concrete and designed for the maximum differential pressure.

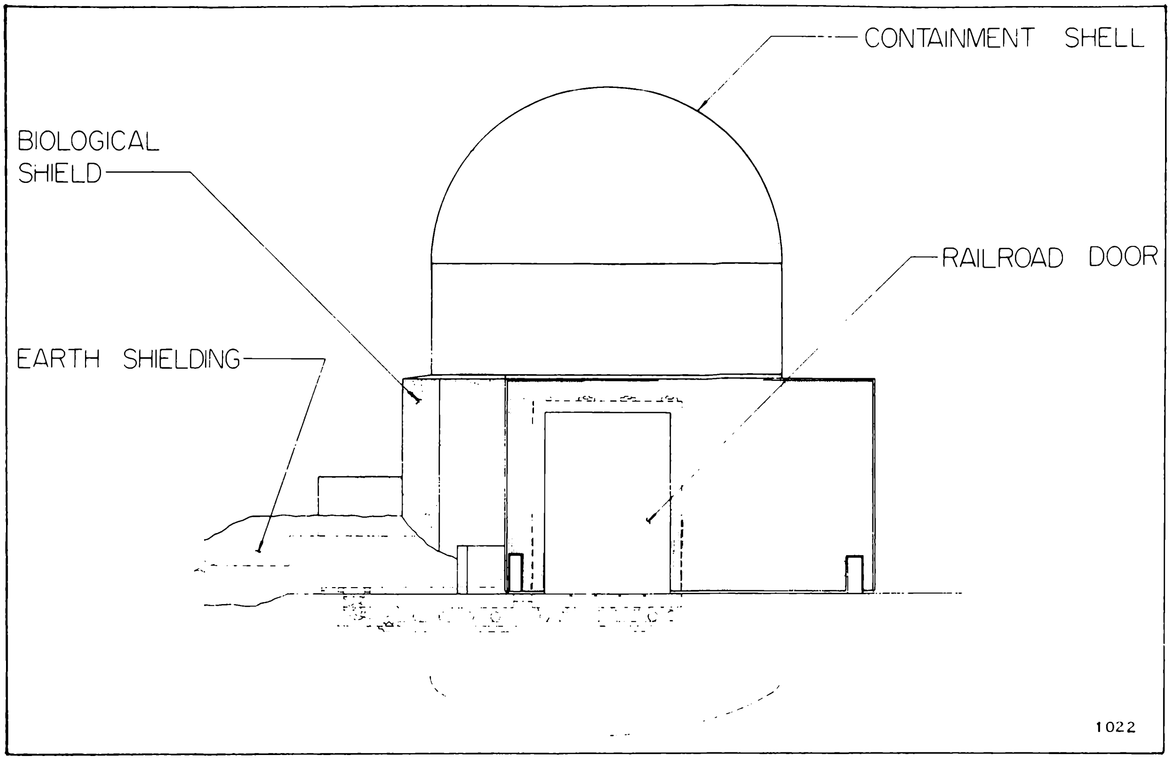


FIG. 27 CONTAINMENT BUILDING -- OUTSIDE ELEVATION.

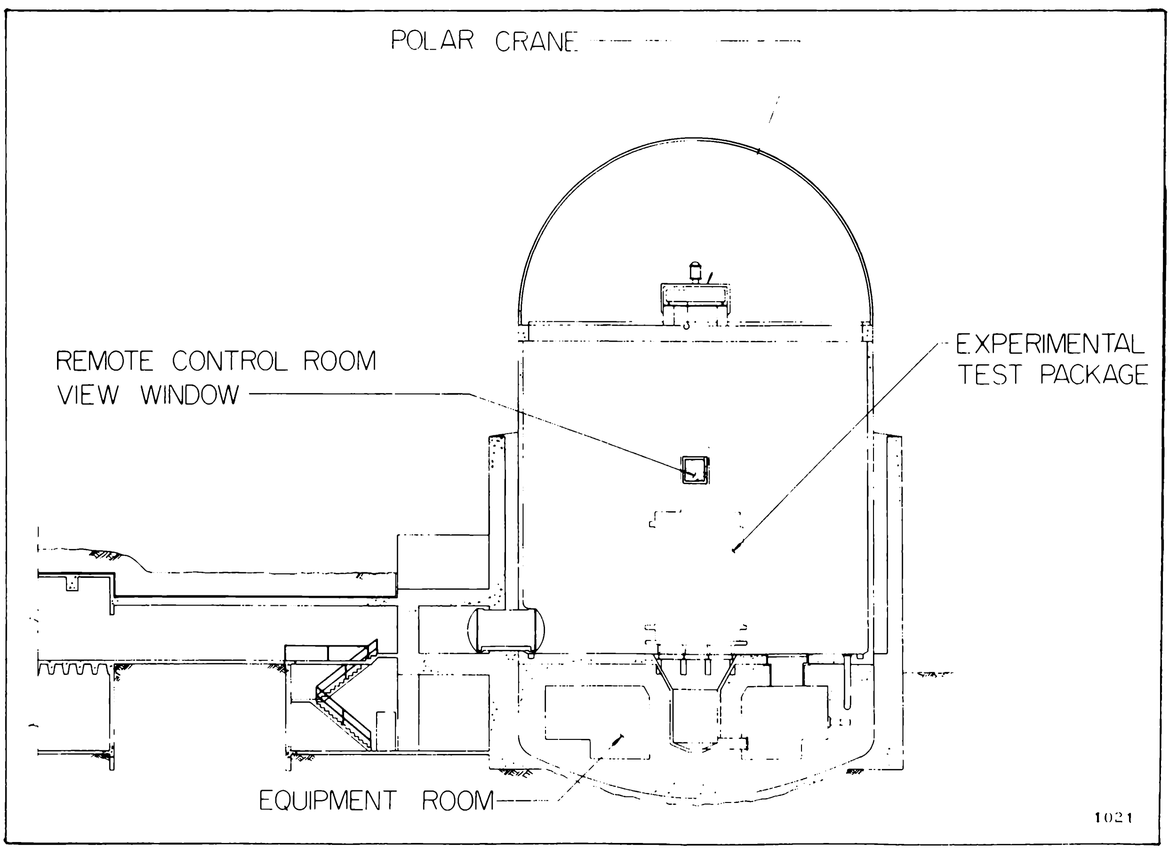


FIG. 28 CONTAINMENT BUILDING -- SOUTH SECTIONAL ELEVATION.

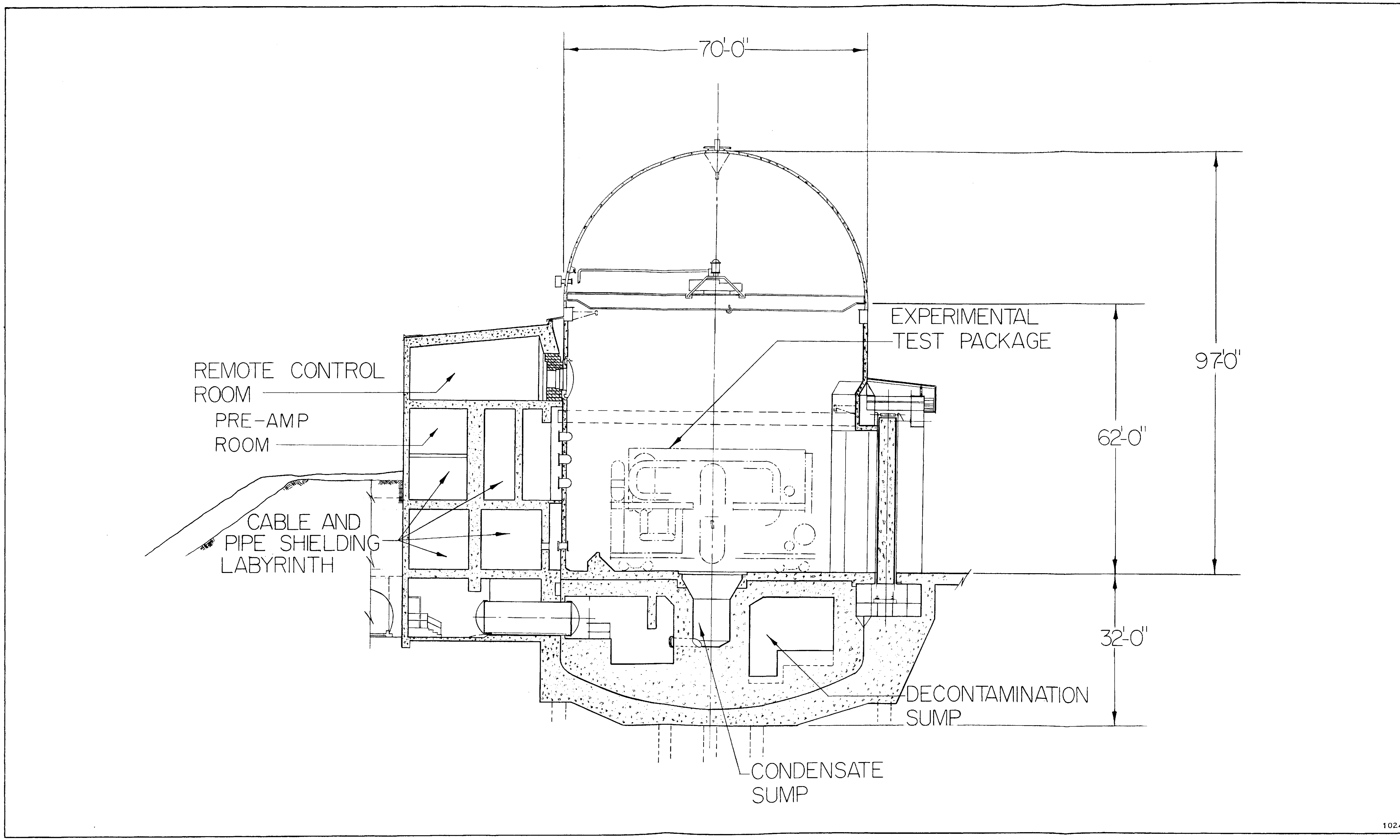


FIG. 29 CONTAINMENT BUILDING -- WEST SECTIONAL ELEVATION.

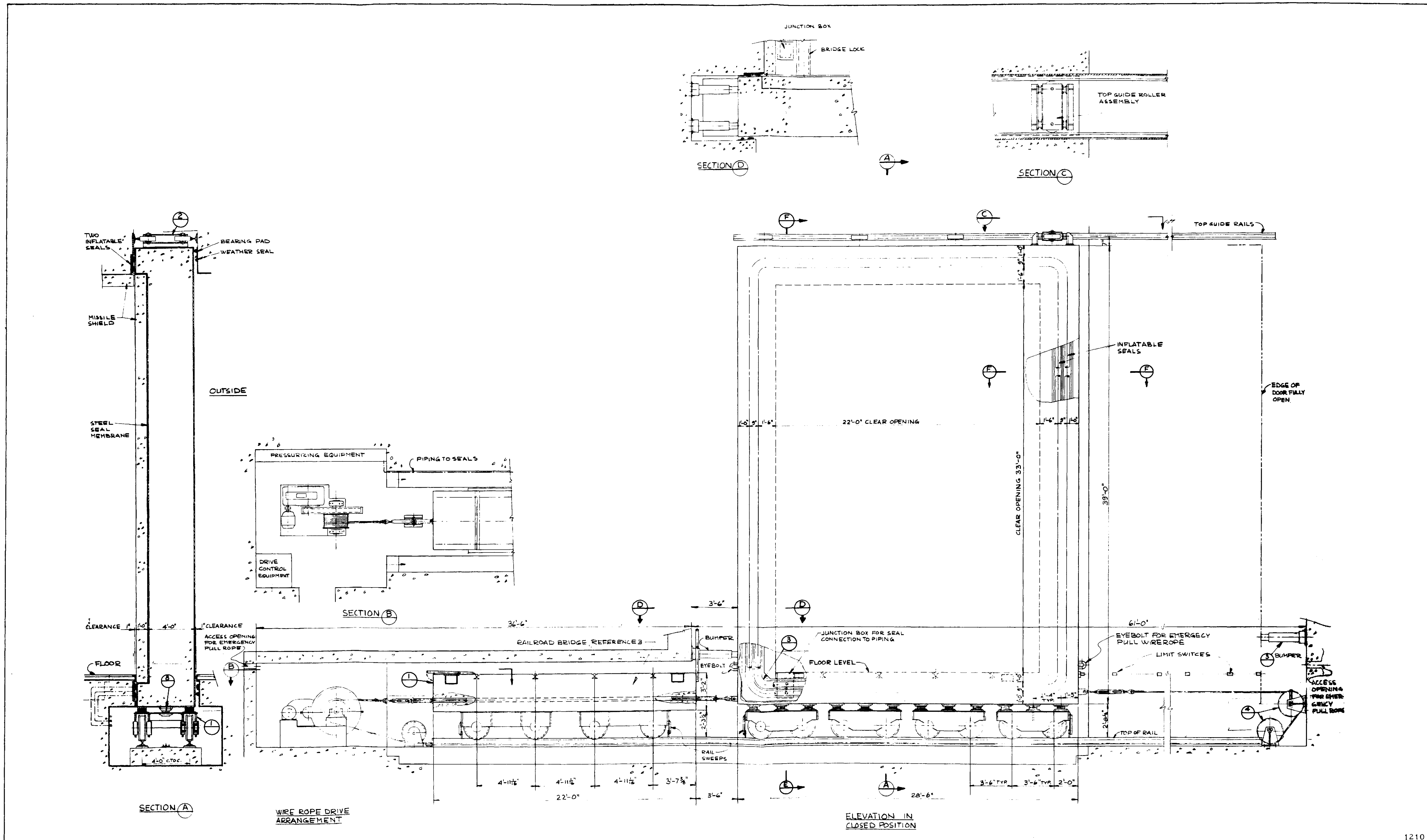


FIG. 30 RAILROAD ACCESS DOOR.

Two parallel and independently inflatable seals are mounted in the inner frame of the vessel railroad door (Figure 31). They maintain a gastight seal between frame and door preventing leakage to atmosphere. The void between the seals is used for pressure testing and monitoring. The door equipment may be controlled both locally and from the remote control room. Limit switches provide signals of position at the control panel. A railroad door follower mobile bridge carrying the 4-rail trackage across the railroad door trench is remotely operated with appropriate position signals and is designed to align the rails precisely and lock into position.

Personnel entry into the containment vessel is through two in-series autoclave doors from the services building. A similar air-lock with two doors permits entry into the basement area. 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 containment building and extend down into the concrete wall adjacent to the basement equipment areas. The stainless steel storage cells measure 12 inches inside diameter by 10 feet deep and are spaced a minimum of 24 inches apart. Each storage cell drains into the hot waste drain system. Each cell is equipped with a closure plug 24 inches long with offsets to preclude radiation shine. All connections to the cells are valved or capped to prevent air leakage from the test chamber.

All service connections are sealed to withstand the pressures expected within the vessel. Quick closing valves on the inlet and exhaust air connections which operate upon loss of primary coolant pressure or an increase in radioactivity of the exhaust air have been installed to effectively seal the building.

Lighting within the containment vessel is maintained at 50 foot-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 150 foot-candle power around the reactor and at the loss of coolant blowdown nozzle to facilitate television coverage during remote operations.

Space is provided in the containment vessel for temporary storage of the reactor vessel head.

Utility and electrical requirements for the reactor package are supplied from the services building through the cable and pipe shielding labyrinth and containment vessel penetrations to the coupling area. All of the utility, electrical, and instrument leads are equipped with connectors located adjacent to the dolly. Jumpers will be used between the facility junction boxes and the reactor package coupling plug by using spool pieces with quick disconnect couplings. The large supply and return secondary coolant pipe lines also are mated to the reactor package lines by remotely operated couplers. Generally, coupling of the process, utility, electrical, and instrument systems is accomplished manually with all disconnect operations to be done automatically.

A remotely controlled top riding, overhead, four-motor, 30/5-ton polar bridge crane is installed in the containment vessel. The clear hook height is approximately 60 feet from the floor. The hoists, bridge, and trolley have five-increment speed ranges with all speeds independent of loads. The ranges are:

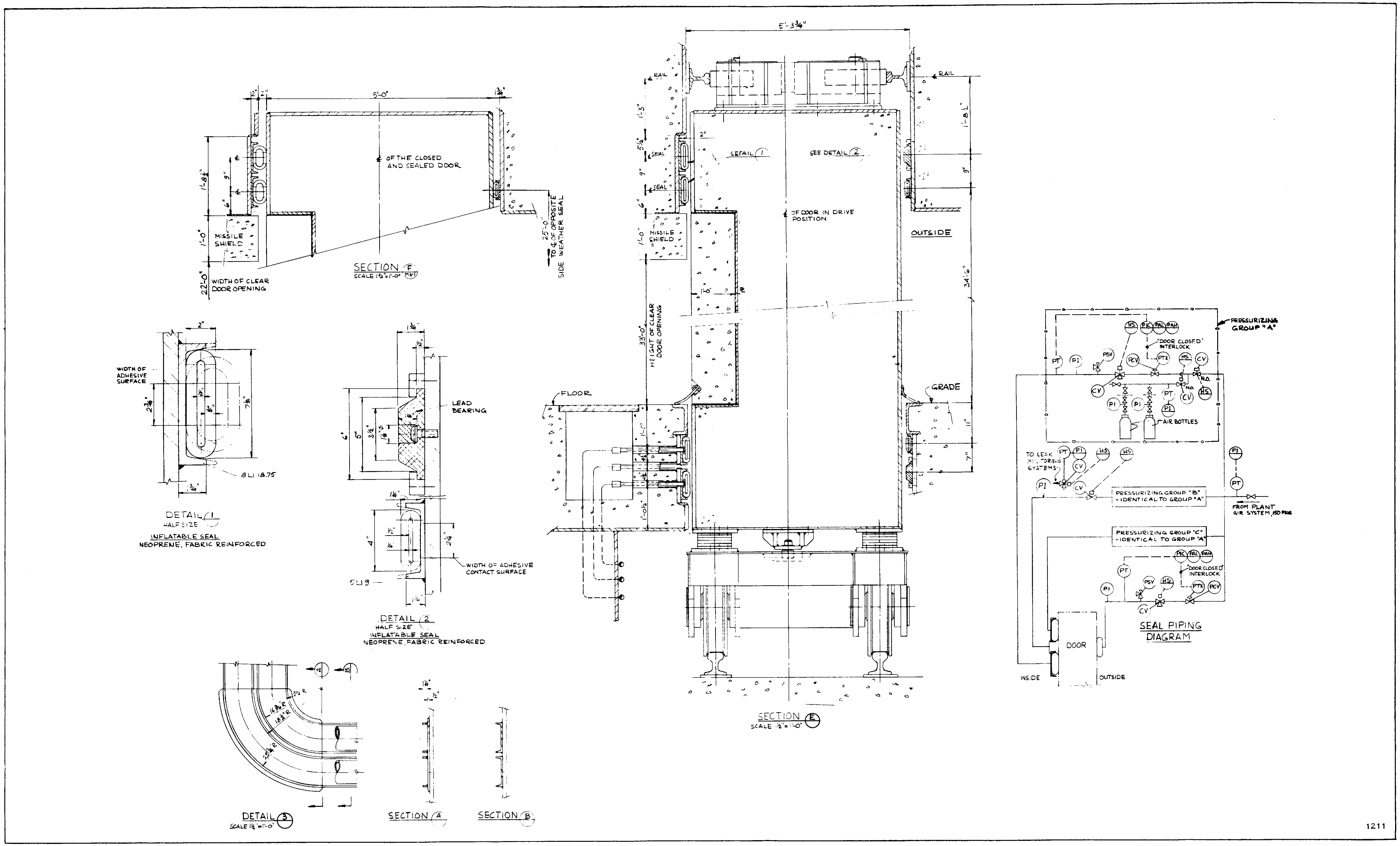


FIG. 31 RAILROAD DOOR SEAL SYSTEM.

(1) hoists - one fpm to approximately 10 fpm maximum, with jogging control to provide incremental steps of approximately 1/8 inch; (2) trolley - maximum speed range of approximately 50 fpm; and (3) bridge - maximum speed range of 100 fpm at the building perimeter.

The crane is controlled from either of two push-button plug-in stations; one located near the floor and one in the remote control room. Operation is aided by visual observation through the viewing window, and by intercom between the operating floor and the remote control room.

Special design criteria include protection of the crane mechanisms, power and control circuits from the environmental conditions of temperature, pressure, humidity, and radiation levels which exist in the containment vessel after a blowdown experiment. The radiation environment was considered in selecting materials, including lubricants and paints, and in providing shielding for components which might be damaged by radiation. In addition, the materials and design facilitate decontamination and provide protection against corrosion by decontamination solutions.

A concrete biological shield is provided around the building to reduce the dose rate to 20 mr/hr at 100 meters from the building during normal reactor operating periods. The limiting dose rate includes the direct radiation and the air scattering contribution through the reactor and containment building dome. The concrete shielding is separated from the containment vessel by approximately three feet.

2.12 Basement Area. The basement area of the containment building is sealed off from the containment vessel test area by a steel membrane imbedded in the first floor and welded to the containment vessel. The basement is shielded from the test area to permit personnel access into the basement (100 mrem/hr at 30 minutes after a blowdown experiment) if such access is required. The following utilities are provided in the basement (see Figure 26).

(1) Plant and instrument air

(2) Power at 110 and 480 V

(3) Potable water

(4) Floor drains

(5) Intercom stations

(6) Heating and ventilating

2.13 Remote Control Room (Figures 28 and 29). The remote control room, which is a part of the containment building, is located opposite the railroad door and above the cable and pipe shielding labyrinths. Controls for the crane, railroad door, and remote handling systems are located in the remote control room.

The remote control room is designed to allow personnel access during periods of high radiation inside the containment vessel test area. A shielding wall between this area and the containment vessel test area limits the maximum weekly dose to 100 mrem (assuming occupancy for 8 hr/day, 5 days/week) following a destructive test with 15 percent of the fission products released to the containment vessel dome. Approximately 90 percent of the maximum allowable accumulated dose occurs during the first 24 hours following the blowdown experiment. This wall is approximately five feet six inches thick, is constructed of high-density concrete, and has a shielding window designed to provide an overall view of the containment vessel test area. This window, measuring five feet wide by three feet high on the inside, takes into consideration missile protection, removal of condensation, thermal shock, and leakage requirements of the containment vessel.

2.2 Services Building (Figures 21 and 22)

The LOFT services building is an area of approximately 4800 square feet located underground between the original FET equipment and control room and the LOFT containment building. It adjoins the east wall of the FET Control and Equipment building. The services building houses miscellaneous equipment including that required for air conditioning, heating and ventilating, and decontamination, and supplements the equipment located in the original FET control and equipment building.

In order to integrate the LOFT containment and services building into the original FET complex, the services building was located immediately south of the FET shielded roadway adjacent to the east wall of the FET control and equipment (C&E) building (see Figures 21 and 22).

3. EXPERIMENTAL FACILITIES SYSTEMS

3.1 Secondary Coolant System (Figures 32 and 33)

3.11 General. The secondary coolant system is an intermediate heat transfer loop that is utilized to transfer heat from the reactor primary coolant to the atmosphere. The heat transfer medium used in the secondary coolant system is high purity water in a closed-loop boiling-condensing cycle.

The secondary coolant system corresponds to that of the usual pressurized water reactor, particularly with respect to the heat exchanger. High purity condensate is pumped into the heat exchanger where it absorbs heat from the primary coolant and evaporates. The steam thus formed flows through a throttling valve to an air-cooled condenser where it gives up its latent heat to air. The condensate is then collected and recirculated by a condensate pump. All components and piping are made of carbon steel and are designed for a pressure of 1000 psig and a maximum temperature of 600°F. Provision is made to remove air and other noncondensables from the secondary system by venting.

Steam is condensed at approximately the same pressure at which it is generated. At part loads, steam is generated at higher temperatures (and pressures) to decrease the log-mean temperature-difference between the steam and the primary reactor coolant in the heat exchanger. This mode of operation allows the reactor coolant system to operate at the same average temperature for the complete range of reactor loads. The secondary coolant system is not utilized for nonnuclear operation.

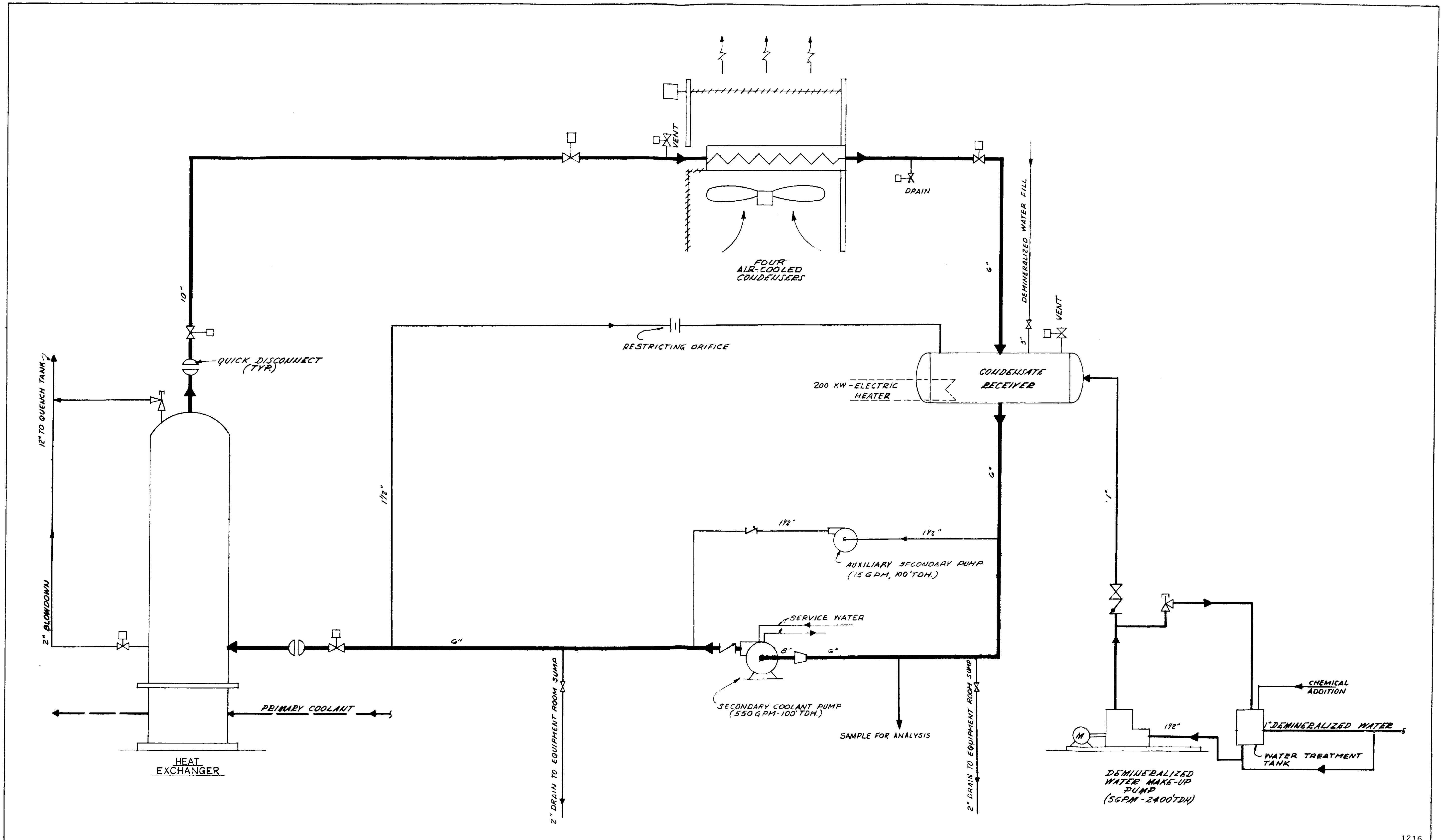


FIG. 32 SECONDARY COOLANT SYSTEM FLOW DIAGRAM.

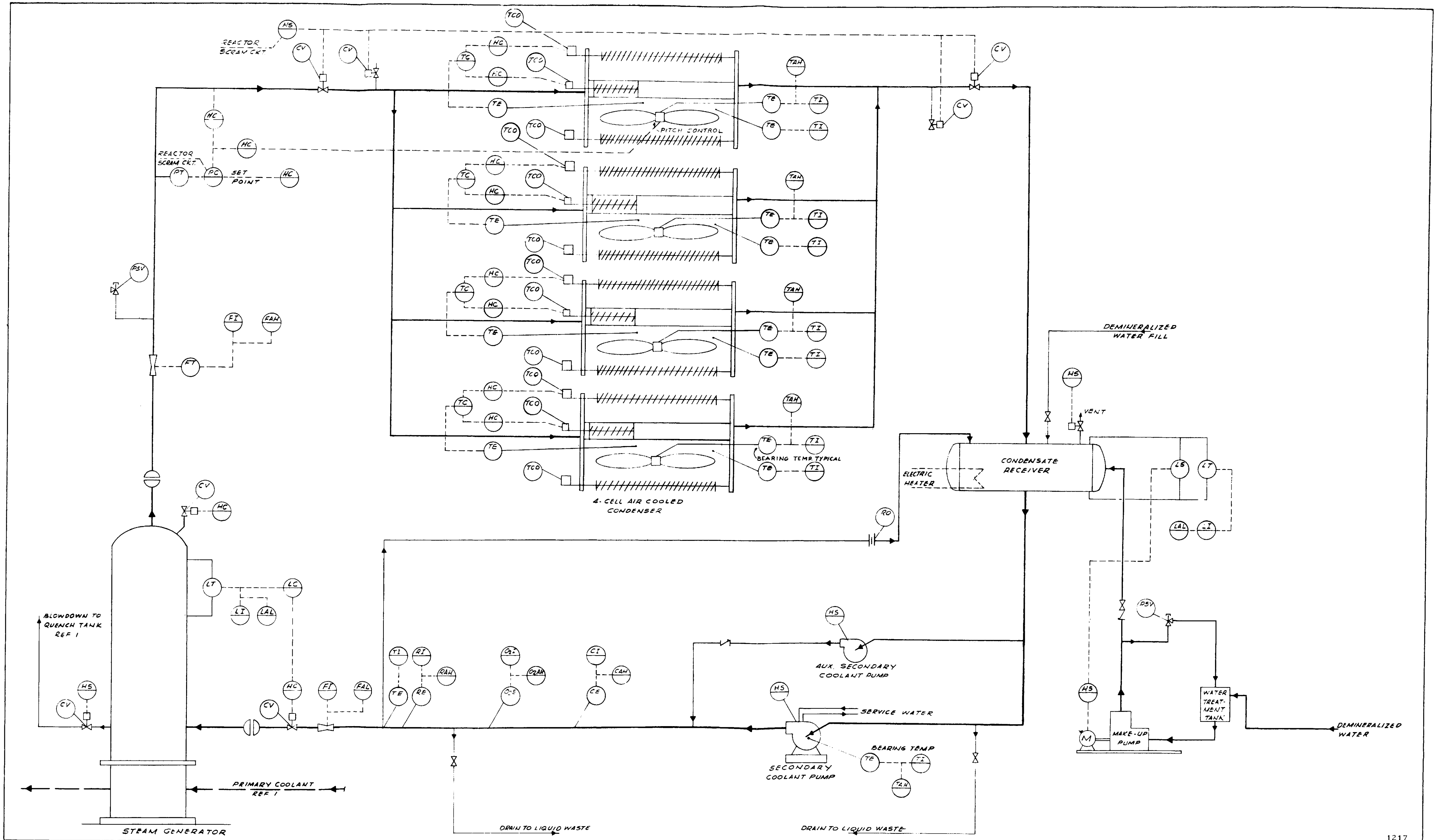


FIG. 33 SECONDARY COOLANT SYSTEM INSTRUMENTATION.

Makeup to the system is supplied from the demineralized water by a high pressure makeup pump. Corrosion in the system is minimized by maintenance of high pH by chemical addition.

3.12 Secondary System Components

(1) Primary Heat Exchanger. Steam is generated on the shell side of the dolly-mounted primary heat exchanger. Only rough separation of entrained water is provided within the exchanger. At full load the steam flow is approximately 214,000 lb/hr at 340 psia and 429°F.

(2) Air-Cooled Condensers. Four multi-celled fan-type, air-cooled condensers with a combined capability of condensing 214,000 lb/hr of saturated steam at 340 psia are installed in parallel. One or all four are operated simultaneously as operating temperatures and pressures demand. Sufficient cooling surfaces are available to remove the reactor decay heat without emergency condenser fans operating.

The air-cooled condensers have approximately 100,000 square feet of extended finned surface area. They are permanently mounted outside the test building with a provision to keep the condensers from freezing in winter when operating at low power or during shutdown. Air vents are provided to remove noncondensable gases.

(3) Condensate Receiver. The 500-gallon capacity receiver is a carbon steel vessel designed for 1000 psig at 600°F. An electrical immersion heater is installed in the receiver to assist in system start-up.

(4) Secondary Coolant Pumps. One secondary coolant pump with a capacity of 550 gpm at 150 to 200 feet TDH and designed for 1000 psig suction pressure at 600°F is provided for normal operation. In addition, an auxiliary pump, arranged to operate continuously in parallel with the main pump, is provided with a capacity of 15 gpm at 150 to 200 feet TDH and 1000 psig suction pressure at 600°F. This pump is connected to the emergency diesel power supply and it assists in system start-up and shutdown and provides for emergency decay heat removal.

(5) Makeup Pump. The demineralized water makeup pump is rated at 5 gpm and 1000 psig discharge pressure. Construction material is carbon steel.

3.2 Sampling Systems

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

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

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

To determine the time-space behavior of fission products such as iodine and the saturation characteristics of the exposed containment vessel and equipment surfaces, plateout samples are required at representative positions throughout the containment volume (Figure 34).

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

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

(1) One station at the center of each 120° sector on the surface of the vessel wall approximately 38 feet above the containment vessel area operating floor.

(2) One station on the operating floor of the containment vessel in each quadrant approximately 12 feet from the periphery of the test area.

(3) One station in each 120° sector on the surface of the dome approximately midway between the center of the containment vessel and the vertical walls.

(4) Two stations on the containment vessel wall in the immediate vicinity of the coupling station. See Figure 34 for location of sampling stations.

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

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

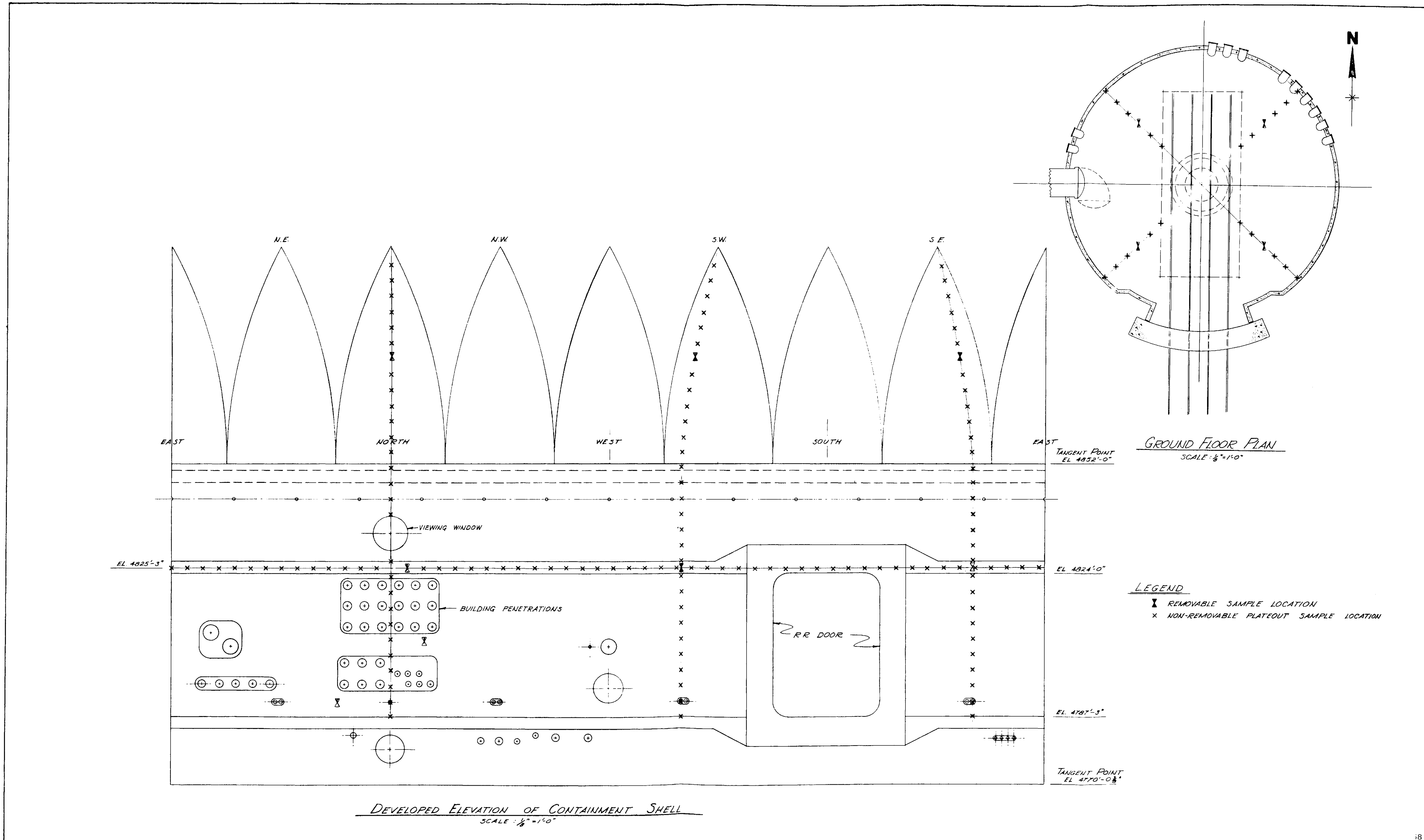


FIG. 34 FISSION PRODUCT SAMPLE LOCATIONS.

(1) At four-foot intervals along the vertical centerline in each 120° sector extending from the containment vessel floor to the top of the dome. All samples are attached to the vessel walls.

(2) At four-foot intervals on the periphery of the containment vessel wall at 38 feet above the floor.

(3) At four-foot intervals on the periphery of the dome at the elevation specified for the removable samples.

(4) At four-foot intervals in each quadrant on the vessel floor, extending from the wall to the projected boundary of the installed experimental railroad dolly.

(5) On each major piece of machinery inside the containment vessel, reactor vessel (4 samples), pressurizer (1 sample), primary heat exchanger (1 sample), primary coolant piping (4 samples), top of railroad dolly (4 samples), underside of railroad dolly (4 samples), overhead crane bridge (4 samples), secondary system piping (4 samples), and other major pieces of equipment ($\approx$ 20 samples).

(6) Inside the reactor pressure vessel and primary coolant system fission product deposition coupons will be used to obtain gross fission product behavior information.

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

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

3.22 Gas Sampling

(1) General. Gas sampling encompasses the collection of representative samples of the containment vessel atmosphere following core meltdown. These samples are used to determine the airborne activities of the noble gases, their daughters, and other volatile nuclides (Xe, Kr, I, Cs, Ba, Te). By obtaining samples as a function of time, information can be obtained pertaining to the time-space behavior of the isotopes and the plateout characteristics as a function of air concentration. In addition, these data assist in evaluating the total quantity of fission products released from the fuel and the concentration available for leakage from the containment vessel as a function of time. Other samples are obtained at the time of maximum concentration of iodine inside the containment vessel. These data are used primarily to determine the plateout behavior as a function of concentration for the various materials inside the containment vessel.

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

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

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

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

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

Solenoid valves at each sampling station are remotely operated from the remote control room.

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

3.23 Particle Sampling

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

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

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

Twelve sampling devices are required, one at each location of the removable plateout samples. However, the sampling tube intakes and exhaust are located at a sufficient distance from the plateout and gas samplers to preclude disturbance of the normal air currents in the building.

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

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

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

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

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

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

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

The recording instruments are located in the control room of the LOFT facility with slave instruments located in the Remote Control Room and the Health Physics monitoring area.

The general location of each detector is as follows:

(1) one above the locomotive entrance door,

(2) one near the shielded sample window,

(3) one in the dome,

(4) one near the personnel entrance to the building,

(5) one approximately 180°F from position 4, and

(6) one approximately five feet above the reactor vessel head.

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

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

3.26 Fission Product Behavior Sample Recovery System. The removable gaseous and plateout sample containers are removed from the containment building as a function of time for analysis in the radio chemistry laboratories at TAN 607 and Idaho Chemical Processing Plant (ICPP). The mechanisms for accomplishing this removal consist of motorized wheels with 24 sample containers attached to the periphery of each wheel. A wheel of fallout samples and a wheel of gaseous samples are located at each of the 12 sample stations within the containment building. At a selected time, an automatic programmer will actuate each wheel. This closes the sample container immediately prior to release into a gravity tube system which leads to an isolation tube penetrating the containment vessel. The isolation tube consists of the tube itself and two valves in series which will accommodate a number of sample containers. These are subjected to high velocity water washing prior to release into shielded casks located in the sample room. The casks are then removed by monorail hoists to the loading area of the shielded roadway for transport to the TSF area. The casks attenuate the direct radiation to acceptable tolerances as outlined in the latest ICC regulations for transport of fissionable materials and consistent with operating contractors health and safety standards.

3.3 Documentary System Cameras

Motion picture cameras are installed in the containment building to record the phenomena associated with the rupture of a high pressure reactor primary water system. Three water-proof and pressure-tight camera housings are provided, each containing one infrared and one conventional black-and-white camera. One camera assembly is mounted immediately under the crane rail and on the approximate centerline of the reactor package. The other assemblies are mounted on the floor on either side of the dolly. With the cameras thus focused on the discharge nozzles and at approximate right angles to each other, discharge flow patterns and experimental system component vibrations and distortions can be recorded and analyzed. These cameras are used only for the nonnuclear blowdown tests and will be manually removed from the camera housings at the termination of each test for film processing.

To record the fission product transport phenomena associated with the nuclear blowdown, pinhole gamma cameras are installed in the floor of the containment vessel and also near the crane rail in positions similar to that described above. Each pinhole camera is shielded by a remotely controlled rotating lead sphere with a viewing window cut in one face. This system protects the film from radiation damage during the rather lengthy cool-down period when direct personnel access to remove the film is prohibited. It is recognized that one frame on each end of the film will be ruined by radiation while the spherical shielding ball is rotating to the optic position.

Pinhole cameras are used to determine the fission product transport and distribution within the containment vessel test area following a core meltdown. A gamma pinhole camera aids in locating areas or objects emitting high levels of gamma radiation. The camera records optical and gamma images on successive frames of highly sensitive 35 mm camera of conventional design. A barrel-type lens assembly protrudes from the body and contains a bakelite slide shutter operated by a solenoid and spring return. The camera frame contains two inches of lead shielding and is housed in a container 12 by 15 by 7 inches. The weight of the camera enclosure is approximately 85 pounds.

The shielded container prevents contamination to which it is exposed within the containment building. Pressure and temperature controls are provided to maintain ambient conditions existing outside the containment building. Two pinhole cameras are located approximately 30 feet from the coolant blowdown nozzle and oriented to provide a direct line of sight at 90 degrees to the blowdown nozzle.

3.4 Pressure Reduction Spray System (See Figure 35)

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

Average wind velocity	U - 2.0 meters per sec
Stability parameter	n - 0.5 (dimensionless)
Virtual diffusion coefficient in the vertical plane	C_z - 0.035 (meters n/2)
Virtual diffusion coefficient in the horizontal plane	C_y - 0.280 (meters n/2)
Uncontrolled leakage rate at 24 psig internal pressure	0.2%/day

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

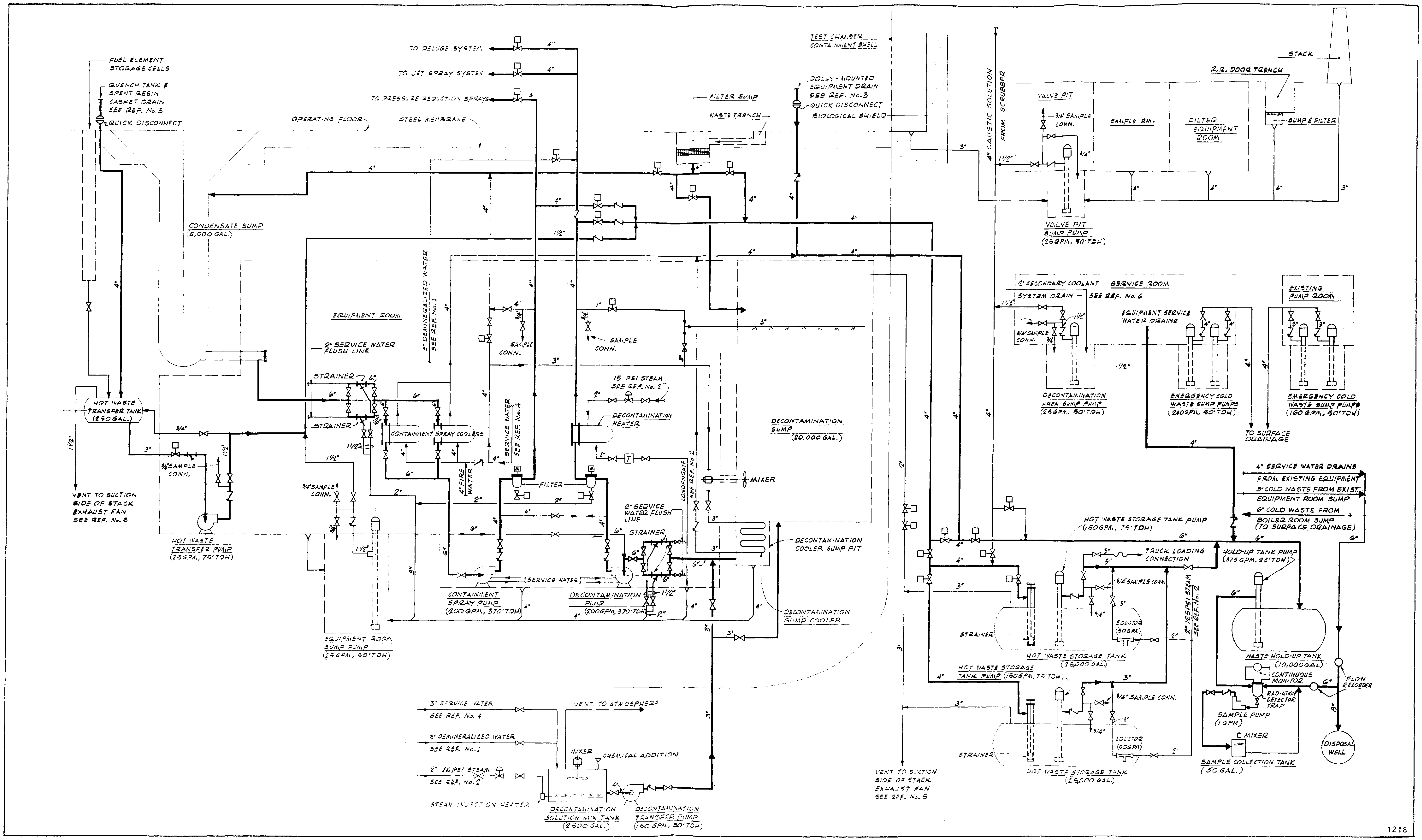


FIG. 35 DECONTAMINATION AND COOLANT SPRAY SYSTEMS FLOW DIAGRAM.

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

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

The electric motor driven spray system pump is supplied from emergency power. The decontamination system pump serves as a standby to the pressure reduction spray system pump. Service water can be supplied to the condensate sump in the event that primary and secondary system liquids are not adequate for the spray system inventory. The spray system flow diagram is shown on Figure 35.

3.5 Decontamination System (See Figure 35)

Systems are provided for decontaminating all exposed surfaces inside the containment vessel. Facilities include a deluge spray system and a high pressure jet spray system. Both are compatible with chemical decontamination solutions and with water.

The decontamination solution makeup system consists of an insulated 2500 gallon stainless steel mix tank with steam heating coils, motorized propeller mixers, and a pump. After the mix tank is filled with demineralized water and heated, chemicals are manually introduced. The chemical solutions of pH 1.5 to 13 are then pumped to the sump, which serves as a feed reservoir for the two spray systems. The single sump located inside the containment vessel has a capacity of 20,000 gallons.

The solutions are pumped from the sump into the headers for the spray systems. After being sprayed from the nozzles, the solutions drain back to the sump for recirculation. Filters installed in the pump inlets remove radioactive contaminants and foreign particulates. This serves the dual purpose of protecting the pumps and spray nozzles from damage due to heavy particles, and preventing the spread of contamination. A removable cartridge-type strainer is installed on the pressure side of the pump. The temperatures of the decontaminating solutions are controlled by a shell and tube heat exchanger, with steam on the shell side.

The deluge spray system, consisting of headers employing banks of spray nozzles ensures that all exposed surfaces are completely washed.

The high pressure jet spray system consists of a bank of rotating head nozzles, each with a capacity of 60 gpm at 115 psig. The nozzles are installed at various locations in the vessel so that all surfaces are no more than 20 feet from a nozzle which ensures complete and thorough coverage. Water is supplied to three jet heads and the deluge system simultaneously by a 200 gpm, 20 hp pump. Water to each group of jet heads is controlled by valves remotely operable from the remote control room.

3.6 Halogen and Particulate Removal System (Figure 36)

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

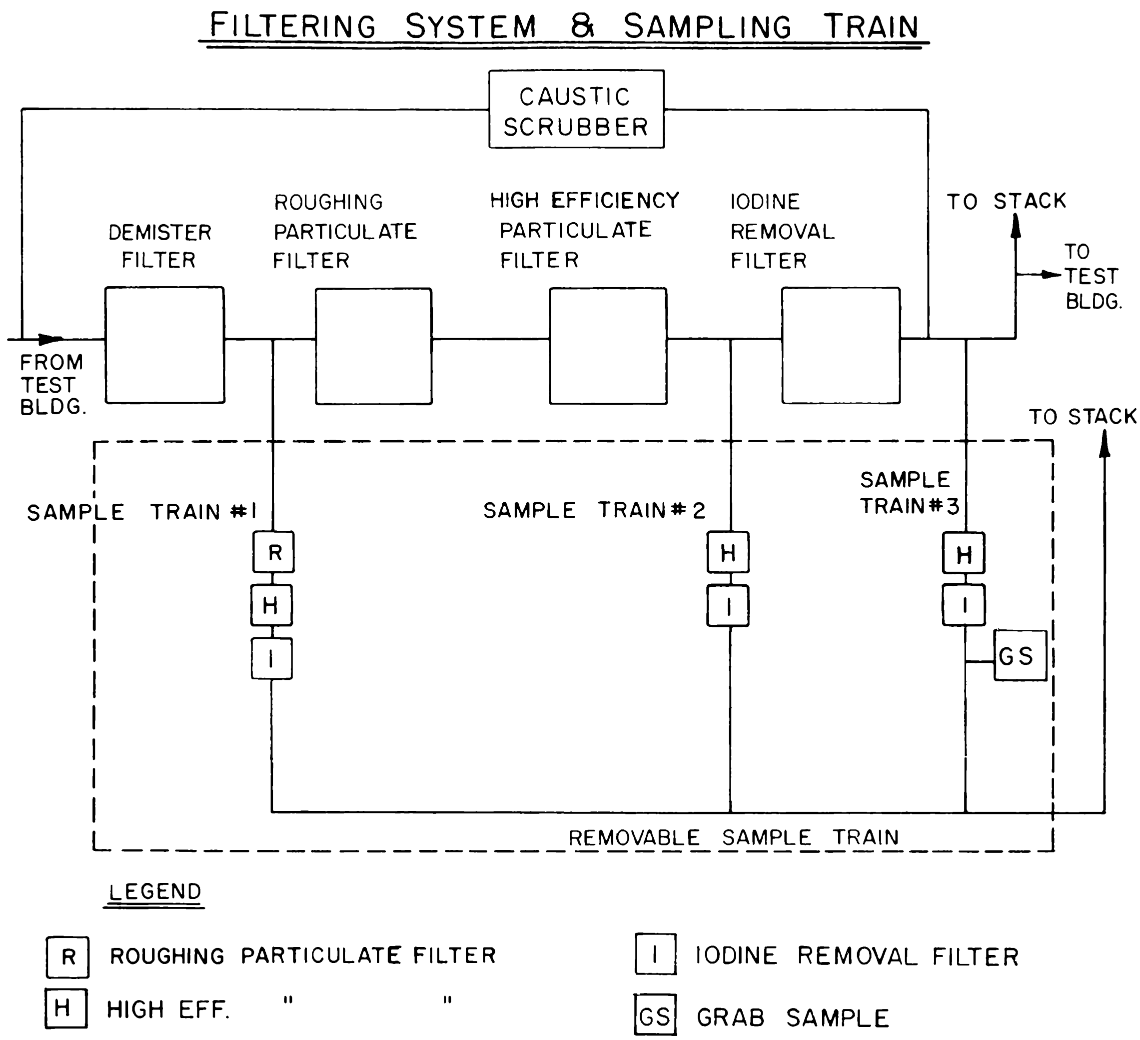


FIG. 36 PARTICULATE AND HALOGEN REMOVAL SYSTEM.

Operation of the halogen and particulate removal system requires that the containment building pressure be essentially atmospheric. This can be accomplished by use of the pressure reduction spray system in approximately four hours after pressurization of the containment, or by allowing the temperature and pressure to decay through heat loss from the containment building to the atmosphere over a longer period of time.

The halogen and particulate removal system filter train is an addition to the containment building exhaust system. Ducts and dampers allow the filter train to be bypassed during normal operation (see Figure 37). The containment building heating and ventilating system is used for: (a) exhaust and recirculation of air during normal reactor operation bypassing the filter train, (b) a once through filtration exhausted to the stack, or (c) recirculation of the containment building gases through the filters and back to the building.

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

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

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

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

The airborne halogen and particulate removal system operation recirculates the gas in the containment building through the filter train and back to the building at the rate of 10,000 cfm. Filter system operation is continued for 20 hours which reduces halogen activity to approximately 2.5×10^{-7} μc/cc. The principle airborne radioactivity present in the containment building subsequent to the 20-hour filter operation is essentially noble gases. These gases are discharged through the filters and up the stack at a controlled release rate.

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

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

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

3.7 Radiological Grid

To assess the hazards resulting from a loss of coolant accident, extensive radiological measurements are required external to the containment building.

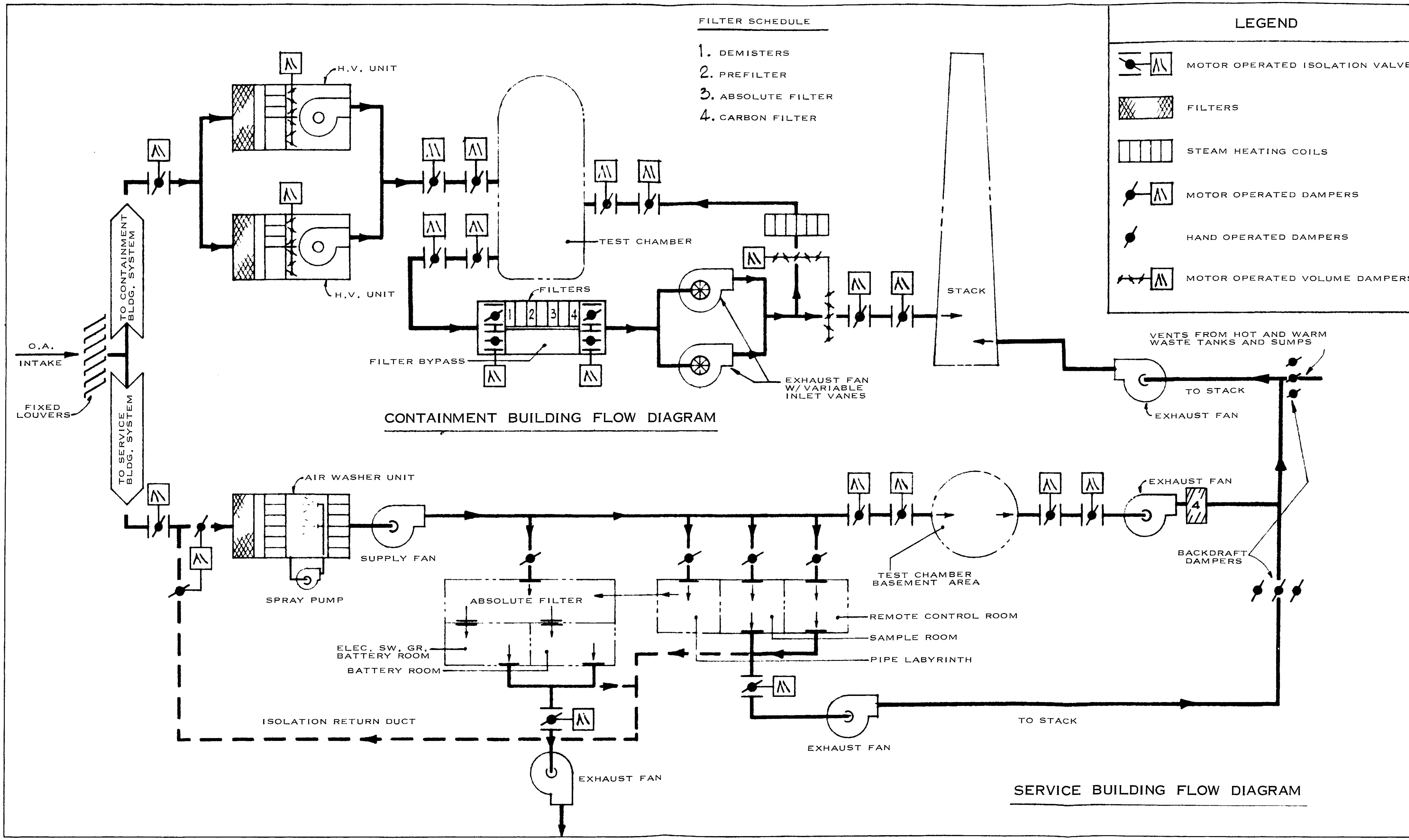


FIG. 37 HEATING AND VENTILATING SYSTEM FLOW DIAGRAM.

These measurements are designed to evaluate both direct radiation from fission products contained in the containment building and the possible radiological doses resulting from fission products leaking to the surrounding environment.

The fission product leakage occurs for several hours following the core meltdown. Thus, an experimental technique is employed to obtain data during all conceivable meteorological conditions that may be expected at the site location. To provide this extensive coverage, radiological detection instruments are located on a grid consisting of concentric circles around the building extending from 100 to 6400 meters. Instruments on two 60-degree sectors of the grid are placed at intervals ranging from 3 to 20 degrees apart to provide extensive coverage during both inversion and lapse conditions. The remaining portion of the grid contains instruments located at 15- and 30-degree intervals to provide data during the meteorological change between lapse and inversion conditions. Part of this instrumentation is also used to measure the direct radiation from the containment building caused by the release of radioactive effluent. The general layout of the grid and instrument locations is presented in Figure 38. The type of instruments employed and the information obtained from each is as follows:

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

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

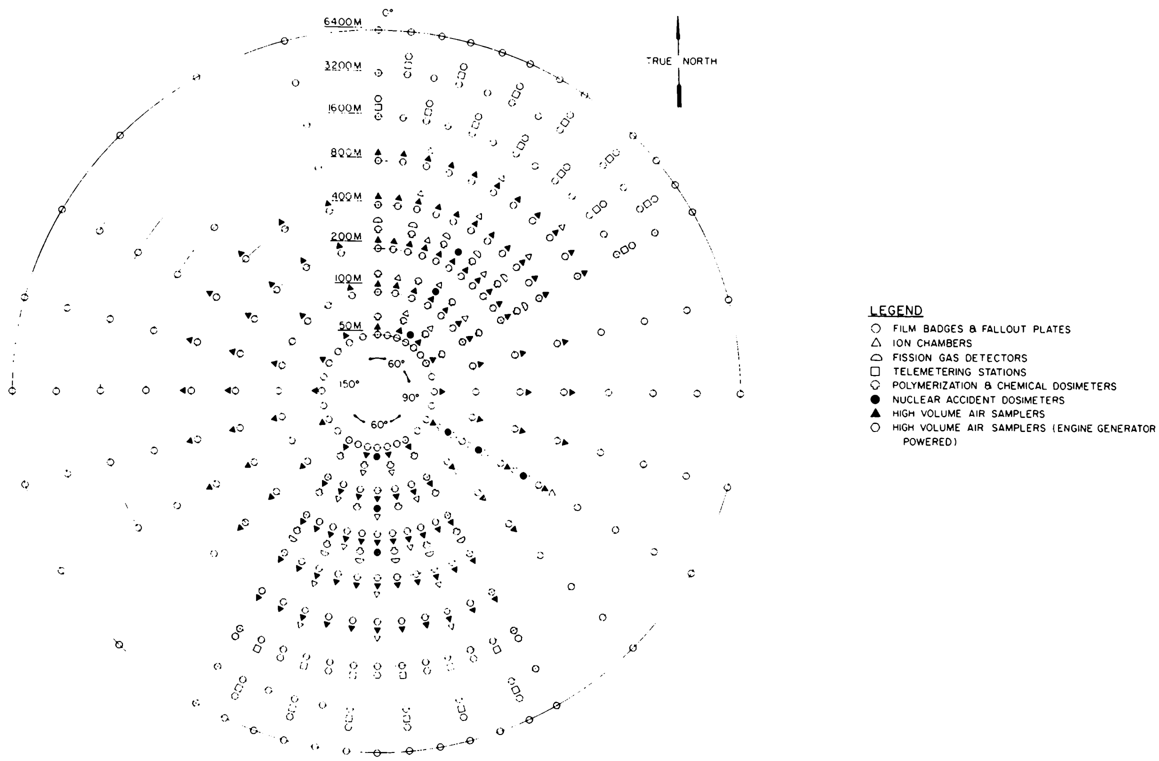


FIG. 38. LOFT RADIOLOGICAL GRID SYSTEM.

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

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

3.8 Controlled Leakage

To provide leakage to the atmosphere commensurate with postulated leakages from typical reactor vessels, the containment vessel includes means for controlled venting to the atmosphere from 0.2 to 5 percent of the free volume per day at internal pressures ranging from 5 psig to the design maximum. This venting system is designed so that the rate decreases as the internal pressure decays away. Provisions are made for measuring and recording the venting rate.

3.9 Leakage Test Systems

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

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

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

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

where

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

$(\Delta p)_o$ = initial pressure difference between container and reference system

W_o = original container air contents

Δw = difference between original and final air contents

Δt = period of test

P_o, P_1 = initial and final absolute pressure, respectively

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

The effect of poor sensitivity in temperature measuring instruments is minimized in the reference system method due to the way the ratio T_o/T_1 enters the leak rate equation.

Temperature gradients in the containment shell may cause many errors in the measurements obtained by the reference system method. For this reason extreme care must be exercised in locating the reference system in the containment vessel and in designing its configuration. It has been found experimentally that temperature gradients are at a minimum from midnight to 8:00 a.m. Data taken during this period are more reliable than data taken at other times.

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

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

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

As with the reference system method, the perfect gas law is assumed, and containment volume changes are neglected. Temperature and pressure errors evident in the reference system method are also present in the latter method; however, these errors may be modified or magnified. Temperature instrument errors are more serious in the absolute method than in the reference system method as indicated by A. H. Heinemon and L. W. Fromm [24] who indicate an error in temperature reading of 0.045°F is equivalent to an error of approximately 0.025% of the container contents.

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

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

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

3.94 Local Leak Test Methods. These methods are more accurate than the integral test methods since they are not affected noticeably by temperature and and pressure variations. The only errors associated with local leak tests are in the measurements of the leakage flow. The accuracy of these measurements is affected by the cross-sectional area of the leak. Babcock and Wilcox [25] give the following limits as the maximum accuracy available:

Soap Bubble and Halide Leak Detector

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

With Mass Spectrometer Added

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

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

(1) Soap Bubble Test. A pressure differential is created across the leakage path to be tested. A soap solution is then brushed over the test area and

the formation of bubbles is observed. The leakage rate is proportional to the rate of growth of the bubbles.

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

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

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

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

4. UTILITY SYSTEMS

4.1 Water Systems

4.11 Supply (Figure 39). Water is supplied by two 1000 gpm, 350-foot TDH, 125 hp deep well pumps and is automatically chlorinated before entering the 500,000 gallon above grade storage tank. The pumps and storage tank were part of the original FET facility, and were adequate to meet the LOFT requirements.

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

4.12 Fire Protection Water and Service Water

(1) Source of Fire Protection Water (Figure 39). The existing 500,000-gallon steel water tank provides storage for fire protection water as well as service and potable water. No separation is made between service water storage and fire protection storage since service water is required at all times during a fire or during a power failure. Fire protection water and service water are supplied through a buried 20-inch diameter pipe from the 500,000-gallon tank to the pump room at the north side of the control and equipment building. The fire pump header is connected to this 20-inch pipe.

(2) Fire Pumps. Four horizontal centrifugal pumps, each rated at 1000 gpm, supply fire protection water to the LOFT facility. Each pump is driven by a 75 hp motor which is connected to the portion of the electrical bus which can be fed from either commercial or diesel power.

(3) Building Drainage. The service building floor is below ground level. This prevents gravity drainage of fire protection water from the building in the

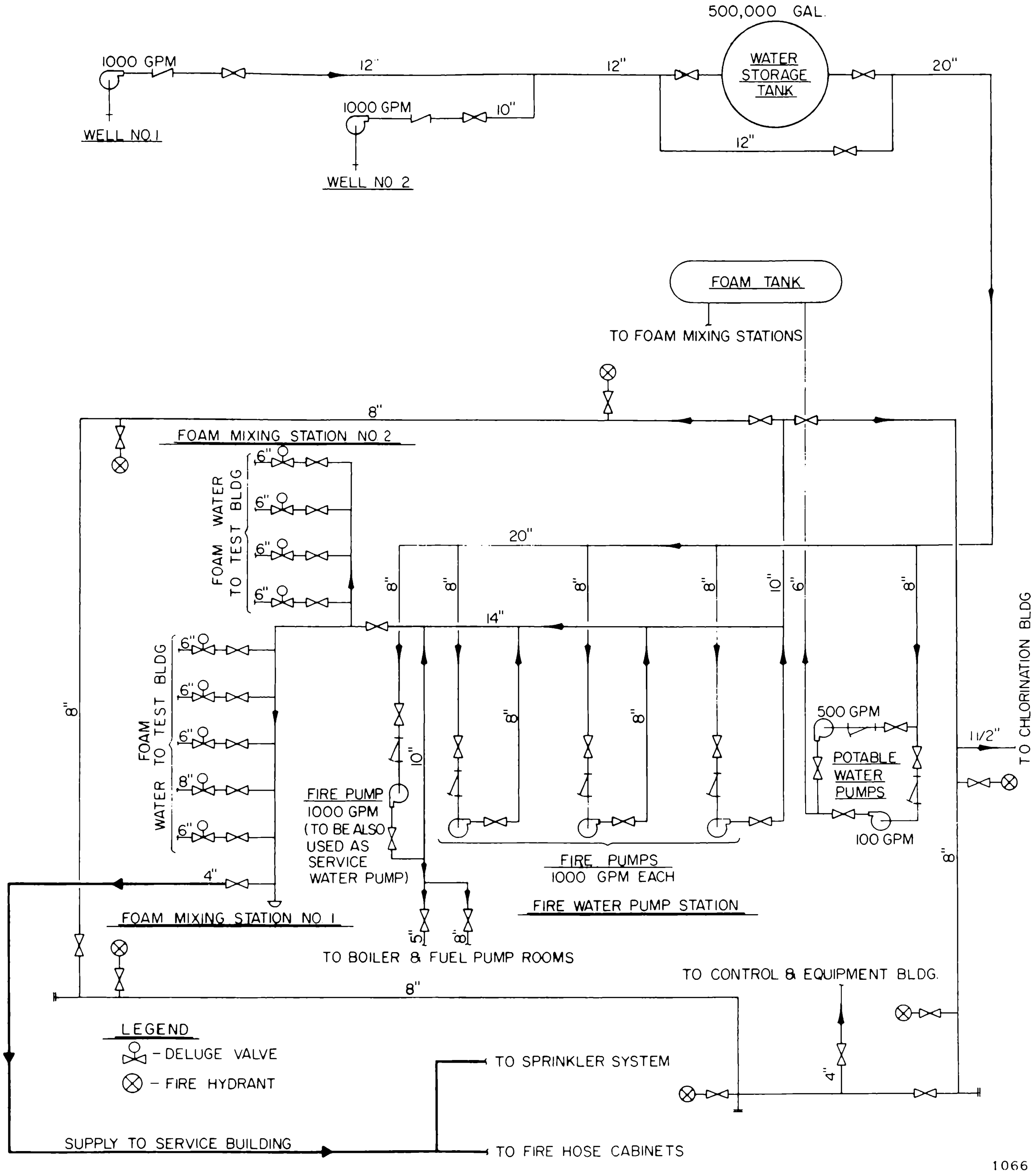


FIG. 39 FIRE PROTECTION WATER FLOW DIAGRAM.

event of fire and it is necessary, therefore, to provide a sump and sump pumps to prevent building flooding at the time of a fire.

(4) Hazard Areas and Fire Protection Features

(a) Fuel Pump Room. The fuel pump room is located at the southeast corner of the FET control and equipment building. The foam extinguishing system installed in the pump room is supplied from a pump header and from a buried foam storage tank north of the control and equipment building.

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

(c) Diesel Generator Area. The diesel generator is protected by fog nozzles connected through a manually controlled valve to the fire header. An automatic high pressure, bottle storage carbon dioxide local application system is provided at the diesel generator. The manually controlled fog nozzles thus provide water backup for the CO_2 primary protection system. CO_2 is the preferred fire protection at the diesel generator since it prevents water damage and avoids electrical hazards.

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

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

4.13 Domestic and Service Water (Figure 40). The original FET facility provided a domestic and service water system which was considered inadequate for LOFT. It included one 500 gpm pump at 255-foot TDH, and one 100 gpm pump at 255-foot TDH, plus a cooling tower and auxiliary equipment. The original FET cooling tower was dismantled, and stored, and a "once through" system was substituted.

The 500 gpm pump is being used in LOFT. The 100 gpm was replaced with two 387 gpm, 150-foot TDH pumps.

4.14 Demineralized Water (See Figure 41). The demineralized water system is sized in accordance with the following criteria:

(1) Two primary coolant system flushes in one hour, followed by filling of the primary coolant system.

(2) Decontamination using 150 gpm of demineralized water for four hours.

(3) Boiler makeup.

(4) Control rod seal and cooling, estimated at approximately six gpm.

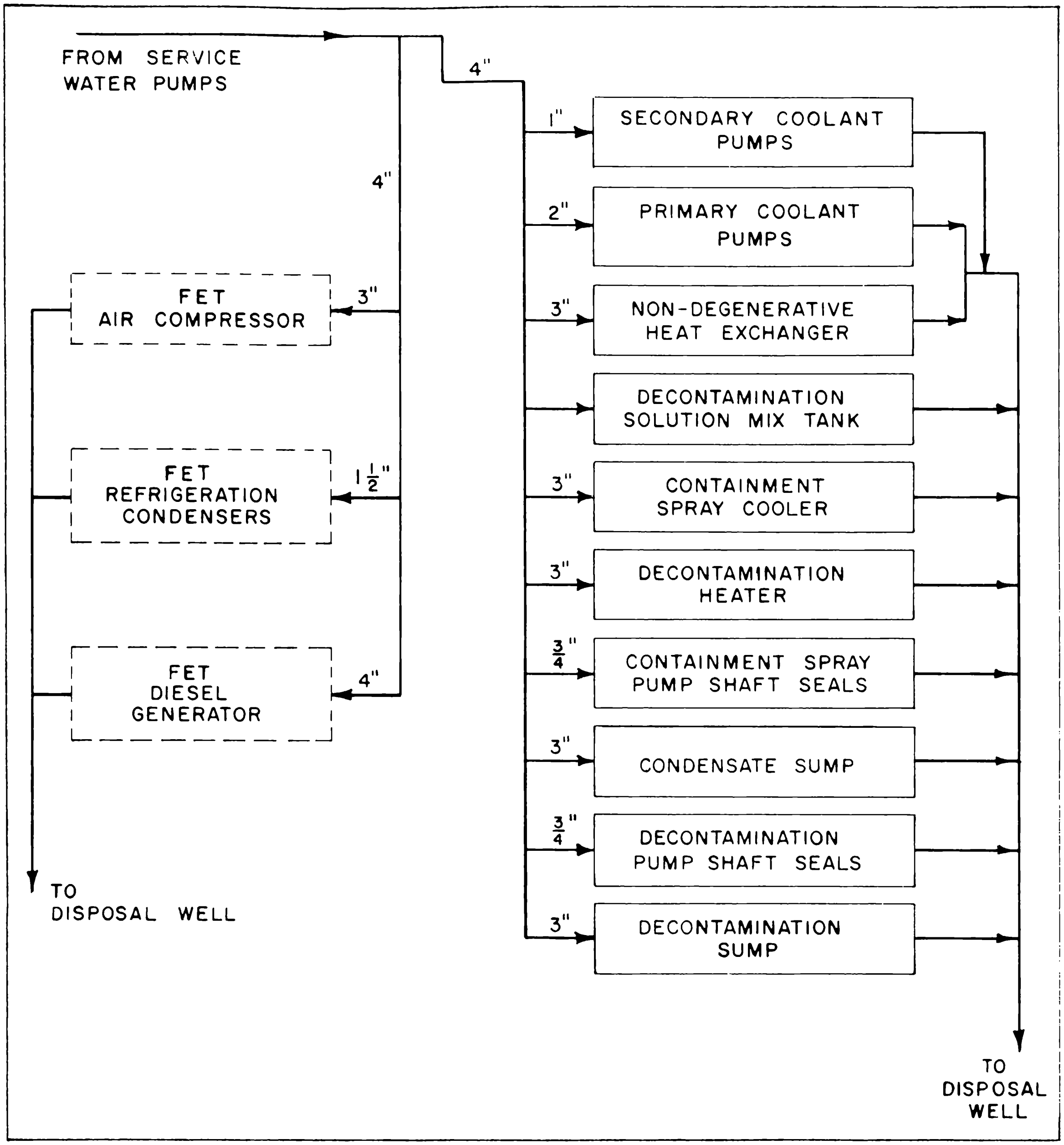


FIG. 40 LOFT SERVICE WATER DISTRIBUTION SCHEMATIC.

(5) Instrumentation ion chamber cooling, estimated at 10 gpm.

(6) Secondary coolant system fill.

(7) Primary and secondary coolant system makeup.

Mixed bed demineralizers are used to produce deionized water with a specific resistance of 500,000 ohms/cm^3. Concentrated sulphuric acid for regeneration of the mixed beds is supplied from a portable 500-gallon carbon steel tank. The acid is transferred to the mix tank by gravity. Flake caustic is mixed with demineralized water to provide caustic for regeneration.

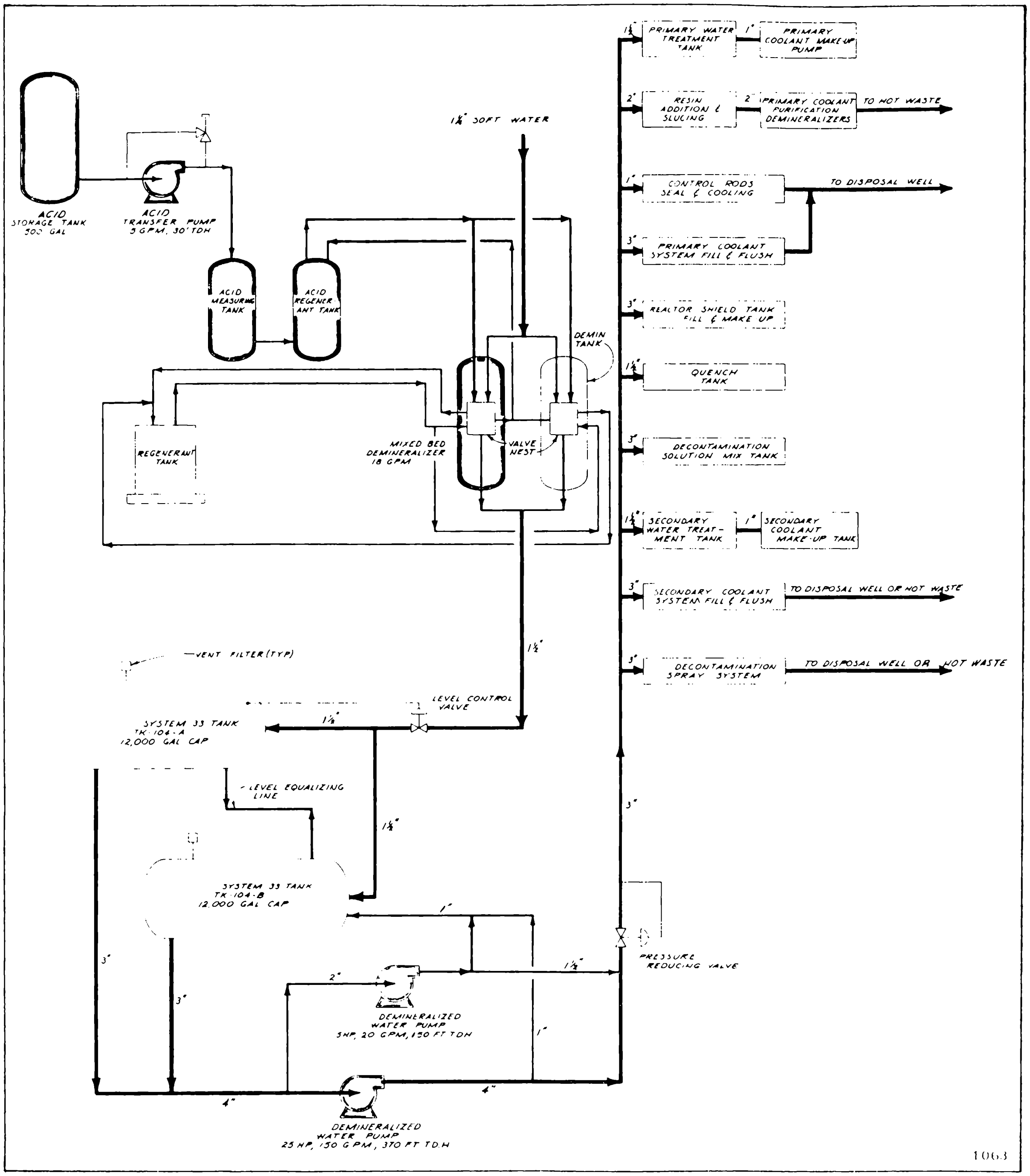


FIG. 41 DEMINERALIZED WATER FLOW DIAGRAM.

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

4.2 Waste Systems

4.21 Liquid Waste Disposal System (See Figure 42)

(1) General. The ID Manual Chapter 0500-7 sets forth policies and procedures to be followed in waste disposal at the National Reactor Testing Station:

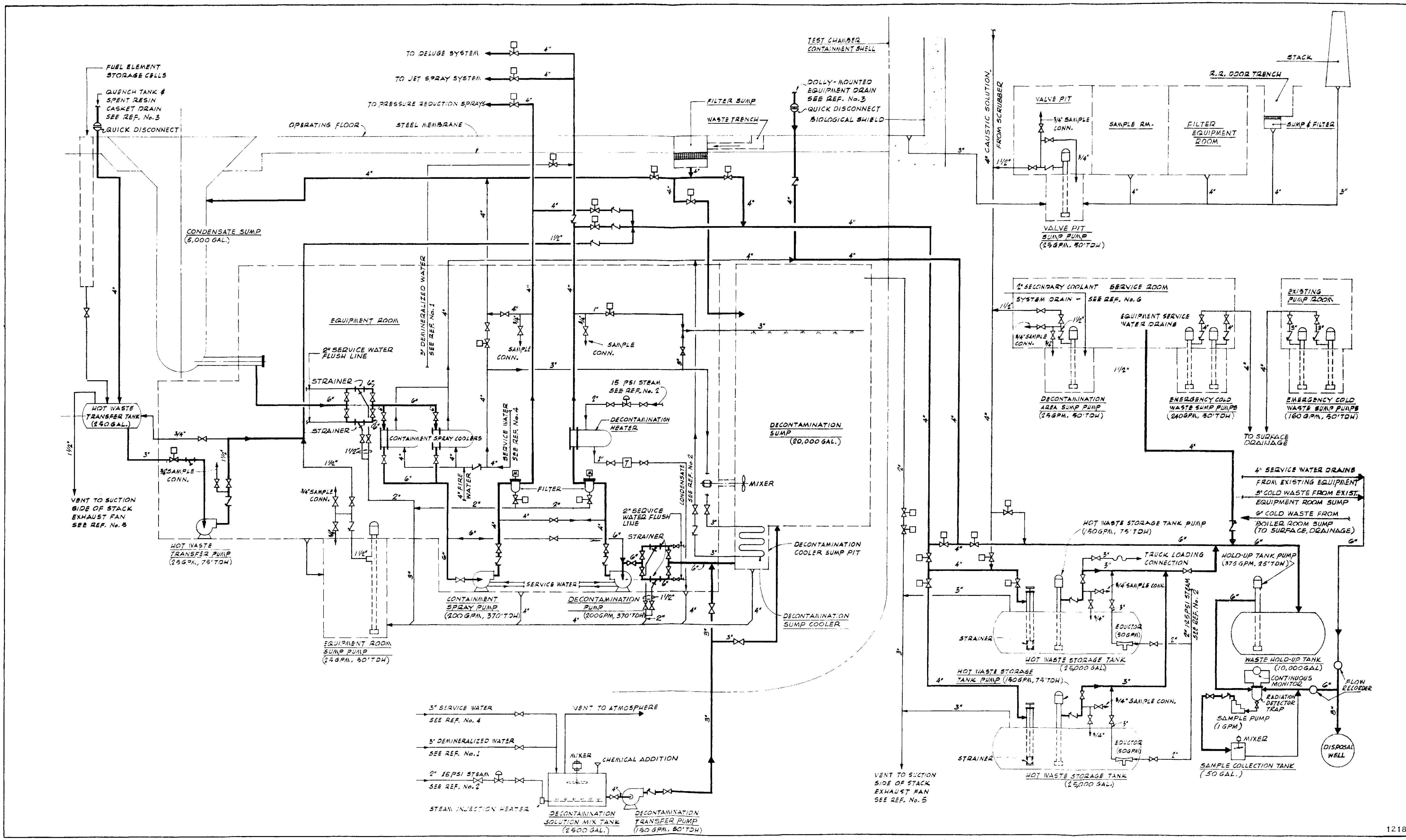


FIG. 42 LOFT LIQUID WASTE SYSTEM FLOW DIAGRAM.

The basic guides for the disposal of the liquid radioactive waste are: "The concentration of liquid radioactive waste discharged to the ground or ground water shall be maintained at levels such that the concentration in water at any point of use shall not exceed a concentration which will result in a dose to individuals in excess of 1/10 of the Radiation Protection Guide (RPG) values."

Figure 43 establishes the concentration limits for liquid waste discharged to the ground at facilities at the NRTS. This chart is a supplement to the provisions of the ID Manual 0500-7. The values obtained by the use of this chart are applicable at the point of discharge, or at the points of last sampling or monitoring. The volume of total wastes is measured before being discharged to the disposal well, and a continuous sample proportional to the volume is taken. Decay, absorption, and dilution have been considered in establishing the above standards. The values are applicable to radioactive liquids for which the isotopic composition is known. If composition is not known, the isotopes will, of necessity, be considered all long-lived (greater than 1000 days).

The four general classes of liquid waste are:

(1) Hot wastes

(2) Process wastes

(3) Corrosive wastes

(4) Sanitary waste

A description of the function of each system including the different conditions of operation follows:

(2) Hot Wastes from Containment Vessel, Remote Control, and Sample Rooms. The operating floor area contains waste trenches located along the containment vessel walls. Floor drains with special fabricated screens are provided in the waste trench. These drains are connected to a common header embedded in the concrete below the operating floor at the point where it penetrates the containment vessel. An isolation valve is provided to maintain the pressure and leakage requirements of the vessel. Outside the containment building, the hot waste discharge header is buried below grade and terminated in one of the two hot waste filter sumps.

The sumps are of concrete construction equipped with a fine mesh screen filter for collecting debris following a destructive test. Normal operation requires only one filter sump with the other as standby; however, both may be operated in parallel. The effluent piping is arranged so that the fluid from either filter sump flows by gravity to either of the 25,000 gallon hot waste storage tanks. Provisions are made for collecting samples from these tanks. A liquid activity monitor is provided to analyze the contamination level of the fluid.

In addition to the floor drains, hot waste drain lines from the following sources are connected to the common collection header:

(1) Reactor pressure vessel and primary system drain.

(2) Quench tank drain.

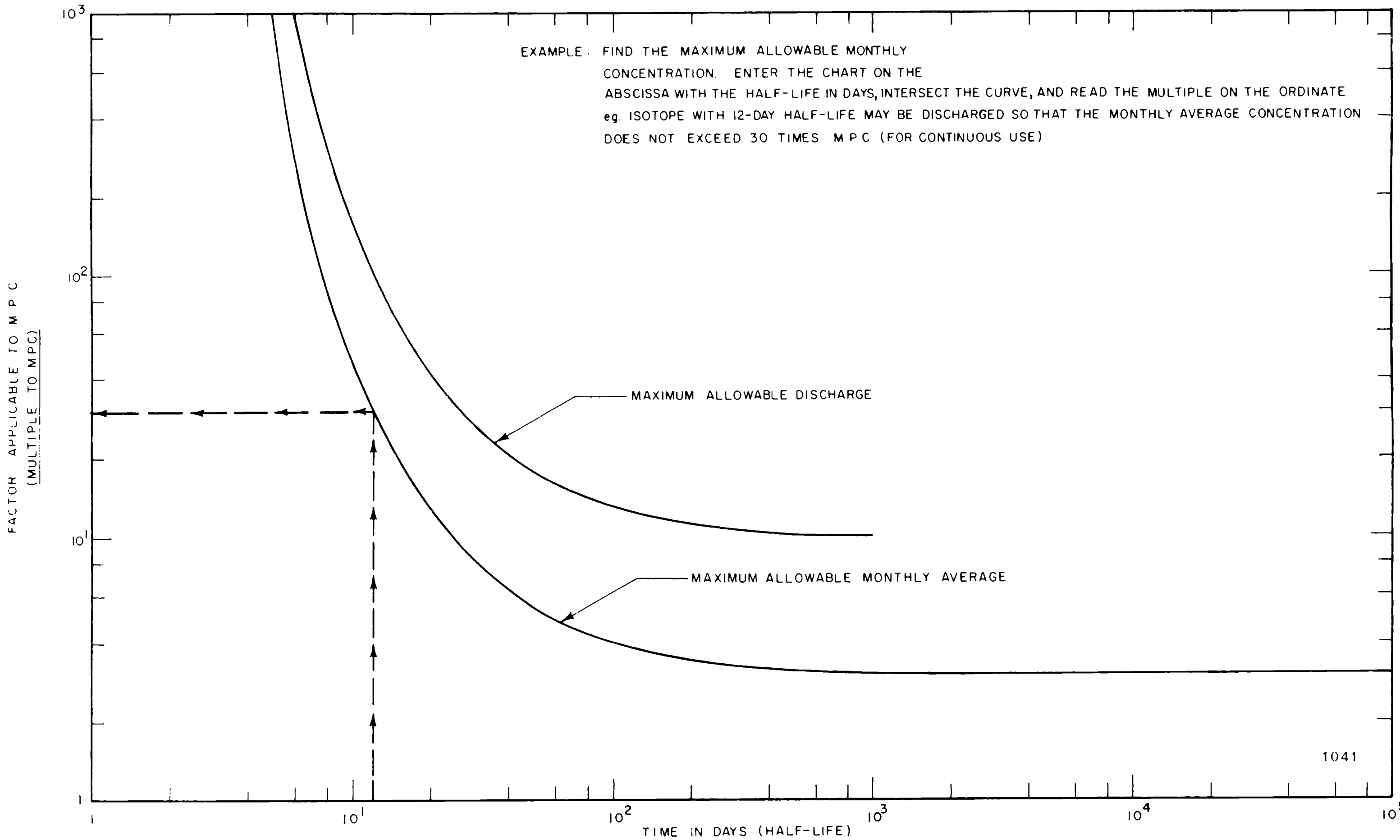


FIG 43. MAXIMUM ALLOWABLE CONCENTRATION OF RADIOISOTOPES IN LIQUID DISCHARGE TO THE GROUND.

(3) Bypass demineralizer spent resin casket drain.

(4) Hot waste sump eductor discharge.

Spray water provided for reducing the containment shell pressure following a loss of coolant test, and decontamination solutions collect in the pit located in the center of the containment vessel. The fluid drains into a hot waste sump located below the pit. In addition, accumulated wastes from the 10 fuel storage cells are discharged to the hot waste sump. A 50 gpm eductor is provided since location of the sump precludes gravity drainage. The discharge piping is embedded in concrete and connected to the main hot waste discharge header. The discharge lines from sample room decontamination drains connect to the main hot waste header outside the containment vessel and downstream from the isolation valve. Corrosion resistant material is used for the hot waste system. Cleanouts and connections for flushing are provided at appropriate locations.

(3) Hot Waste Storage Tanks. Two 25,000 gallon underground hot waste storage tanks, located in the yard adjacent to the containment building, hold the hot waste effluent until it can be analyzed for disposal. One 100 gpm vertical turbine pump mounted on each tank is provided to discharge the hot wastes from the tank. These pumps are remotely operated and operation is governed by the remote level indicating devices. A 50 gpm eductor is located in the tank with blind flanges for possible future hookup to a steam connection.

The discharge piping is manifolded so that the fluid from the tank can be pumped to the disposal well or to the tank loading connection. Connections are provided for sampling the tank contents which dictate whether the liquid is pumped to the disposal well or to the truck connection for transfer outside the LOFT area to liquid waste evaporators. Another provision is a raw water backflush connection.

(4) Containment Building Process Wastes. Cooling water from the primary pumps and the reactor shield tank collects in a common header discharging into the disposal well. An isolation valve is provided in the discharge line outside the test building.

(5) Corrosive Wastes. This system carries the acid and caustic regeneration wastes from the demineralizers in addition to acid wastes from the water makeup and boiler water treatment equipment. Wastes flow by gravity in separate corrosion resistant lines to the leaching pond.

(6) Sanitary Wastes. The sanitary sewage collection and disposal system is designed for three shifts of 20 people each. The system consists of a collecting lateral, a concrete septic tank, and a lift station to pump septic tank effluent to the disposal well.

4.22 Solid Wastes. The NRTS burial ground and necessary handling equipment are adequate for the amount of solid waste expected from LOFT.

4.23 Gaseous Wastes. The hot waste tanks and decontamination sump are vented to the suction side of the exhaust filter equipment. The exhaust filter system handles the exhaust air from the test area within the containment vessel and from the equipment area of the containment building basement. The exhaust filters remove particulates and halogens from the exhaust stream before it is discharged up the stack by the fan. Installed in the 150-foot stack are monitoring probes connected to monitoring equipment located in the filter vault near the base of the stack.

4.3 Plant and Instrument Air System (Figure 44)

Two plant and instrument air compressors with after cooler, air dryer, oil vapor remover and receivers were originally installed in the FET control and equipment building. One of the compressors has a free air capacity of approximately 1000 CFM at 200 psia and the other a free air capacity of approximately 55 CFM at 250 psia. Total receiver capacity is 900 ft^3.

The capacity of this system was more than adequate to fulfill LOFT requirements. A two-inch air line was installed from the FET facility to the LOFT containment vessel to supply plant, instrument, and containment vessel test air.

4.4 Heating and Ventilating Systems

4.41 Control and Equipment Building Heating and Ventilating. The FET control and equipment building heating and ventilating system was adequate, with minor modifications, for the LOFT facility. These modifications were (a) install motor operated dampers in the heating and ventilating air intake and exhaust shafts to isolate the building during periods when radioactive contamination is present in the outside air, (b) supply combustion air to the diesel engine from an outside air duct, (c) provide for shutdown of the steam boilers during periods of isolation, and (d) shutdown the existing exhaust fans and provide isolation valves in the ducts.

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

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

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

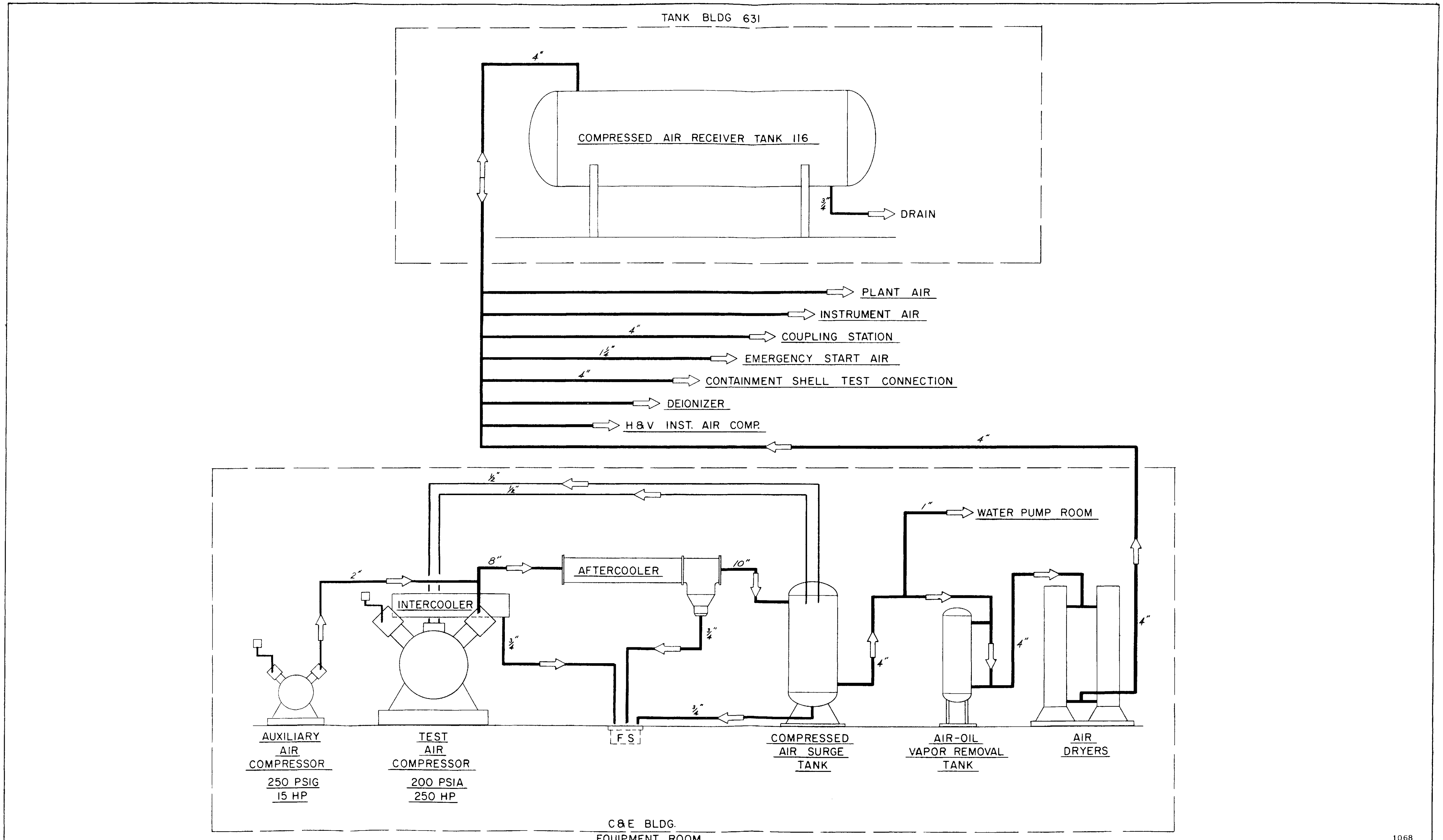


FIG. 44 PLANT AND INSTRUMENT AIR SYSTEM.

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

The exhaust-recirculation ventilation system, as shown on the flow diagram (Figure 37) is designed for operation when high radioactivity in the containment building necessitates closure of the outside air ventilation isolation valves. A 10,000 cfm capacity filter train in the exhaust-recirculation system provides for removal of radioactive contaminants. During normal operation, the filters in the exhaust-recirculation system are bypassed.

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

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

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

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

4.5 Steam Plant (Figure 45)

Two 300 bhp boilers were installed in the original FET facility and were sufficient to meet the 15 psig steam demands for FET and LOFT space heating. Auxiliary equipment includes a feed water tank, motor-driven feed water pumps, and a blowdown tank. A chemical feed system treats the boiler feed water with disodium phosphates for pH control and with sodium sulfite for residual oxygen removal.

Two motor-driven fuel oil transfer pumps (2.5 gpm, 100 psi, 0.5 hp) and one duplex fuel oil strainer are located in the piping tunnel adjacent to the fuel oil storage tank. 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.

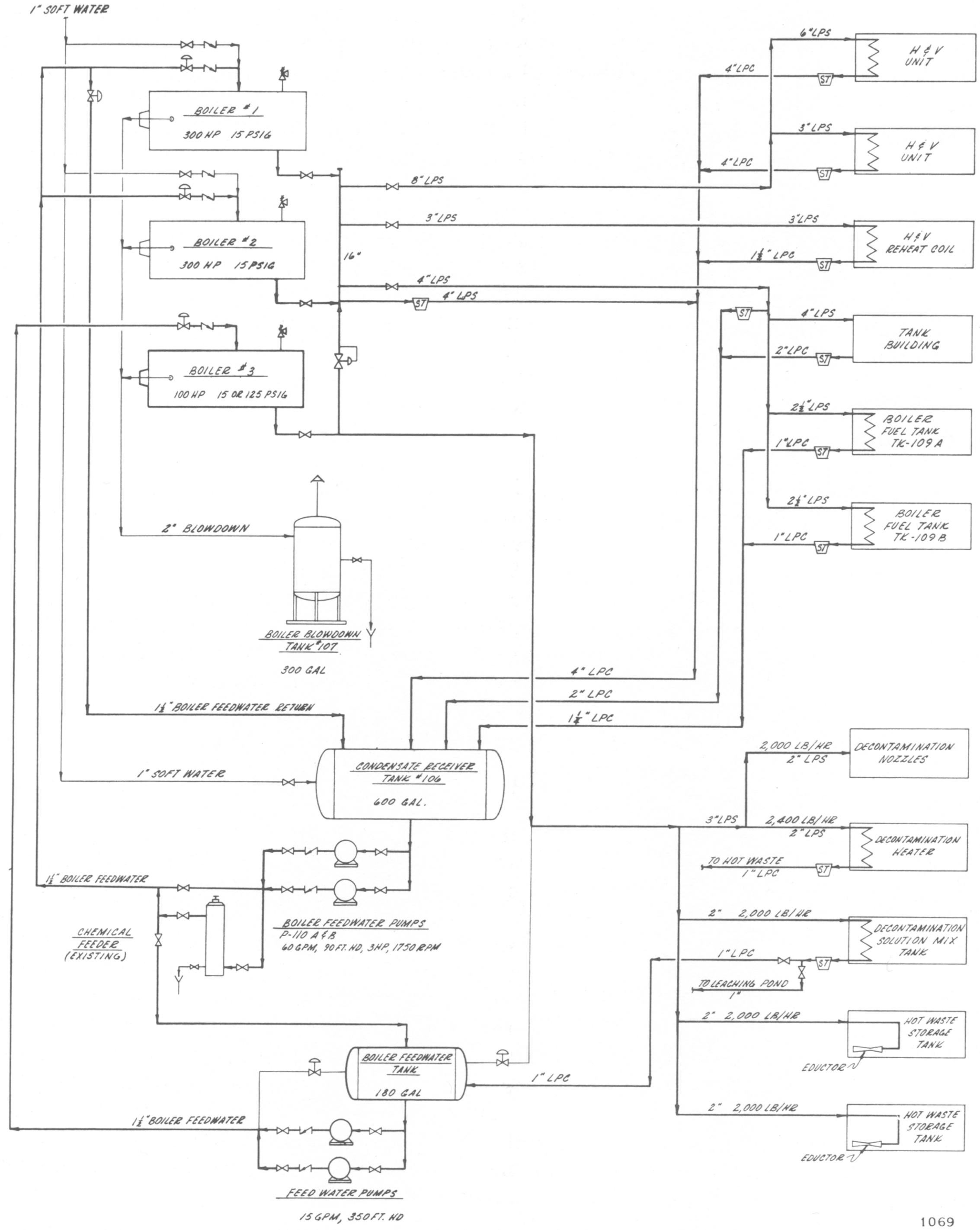


FIG. 45 STEAM SYSTEM FLOW DIAGRAM.

One additional 100 hp, 125 psig boiler was required for LOFT decontamination purposes. The latter was installed in the space formerly reserved for an additional 100 kW diesel generator in the FET control and equipment building.

4.6 Electrical Power

4.61 Transmission and Distribution System. Commercial NRTS electric power is supplied to the combined FET-LOFT loads from the TAN substation over the rebuilt FET 13.8 kV, wood pole, overhead transmission line (Figures 46 and 47). Power is supplied to the overhead transmission line via underground 15 kV cables from cubicle Number 13 in the TAN substation, 13.8 kV, 140 MVA, switchgear. Power is supplied to the new FET-LOFT substation via underground 15 kV cables from the transmission line terminal pole using underground ducts and manholes. Power is supplied to the FET well pump substations over the 13.8 kV overhead tap line.

Expulsion-type, 15 kV lightning arresters are provided on the transmission line at each overhead cable and substation termination and at intervals of 2000 feet along the length of the line. Open-type 15 kV fused disconnect switches are provided in the transmission line ahead of each substation.

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

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

4.62 Normal Power Systems

(1) 2400-Volt Power System. The 2400-V power system starts at the terminals of the LOFT 1000 kVA, 13.8 kV to 2500 V transformer and includes the LOFT 2400 V switchgear and feeders to motors and power loads. This power system supplied power to the LOFT primary coolant pump, the electrically driven fire pump, the radiological grid, and the test dolly photographic lighting system.

(2) 480-Volt System. The 480-volt power system is divided into two subsystems. The first subsystem starts at the terminals of the FET 1500 kVA, 13.8 kV to 480 V transformer and includes the normal power section of the modified FET 480 V switchgear and feeders to motor control centers, motors, lighting transformers, and power panel boards. This system supplies power to all FET loads and those LOFT loads located within the FET building complex which operate from normal power only. This system also supplies normal power for reactor instruments, health physics instruments, test instruments, process instruments and associated controls including the reactor safety system 120 V AC normal power bus.

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

(3) Emergency Power System. The emergency power system starts at the terminals of the FET 1000 kW, 480 V diesel engine-generator unit and includes the emergency power section of the modified, FET 480 V switchgear and feeders to motor control centers, motors, lighting transformers, and power

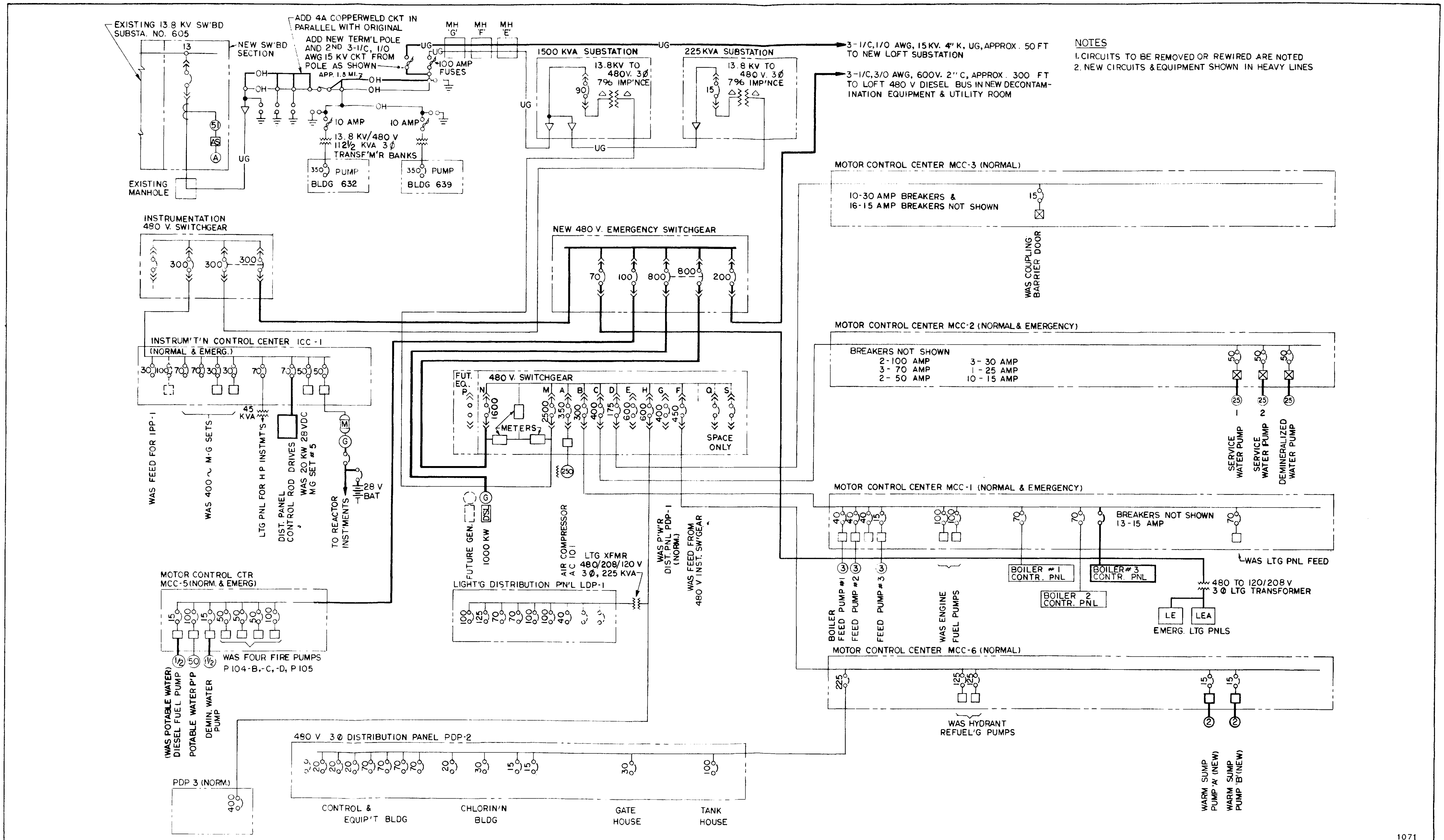


FIG. 46 FET ELECTRICAL DISTRIBUTION MODIFICATIONS.

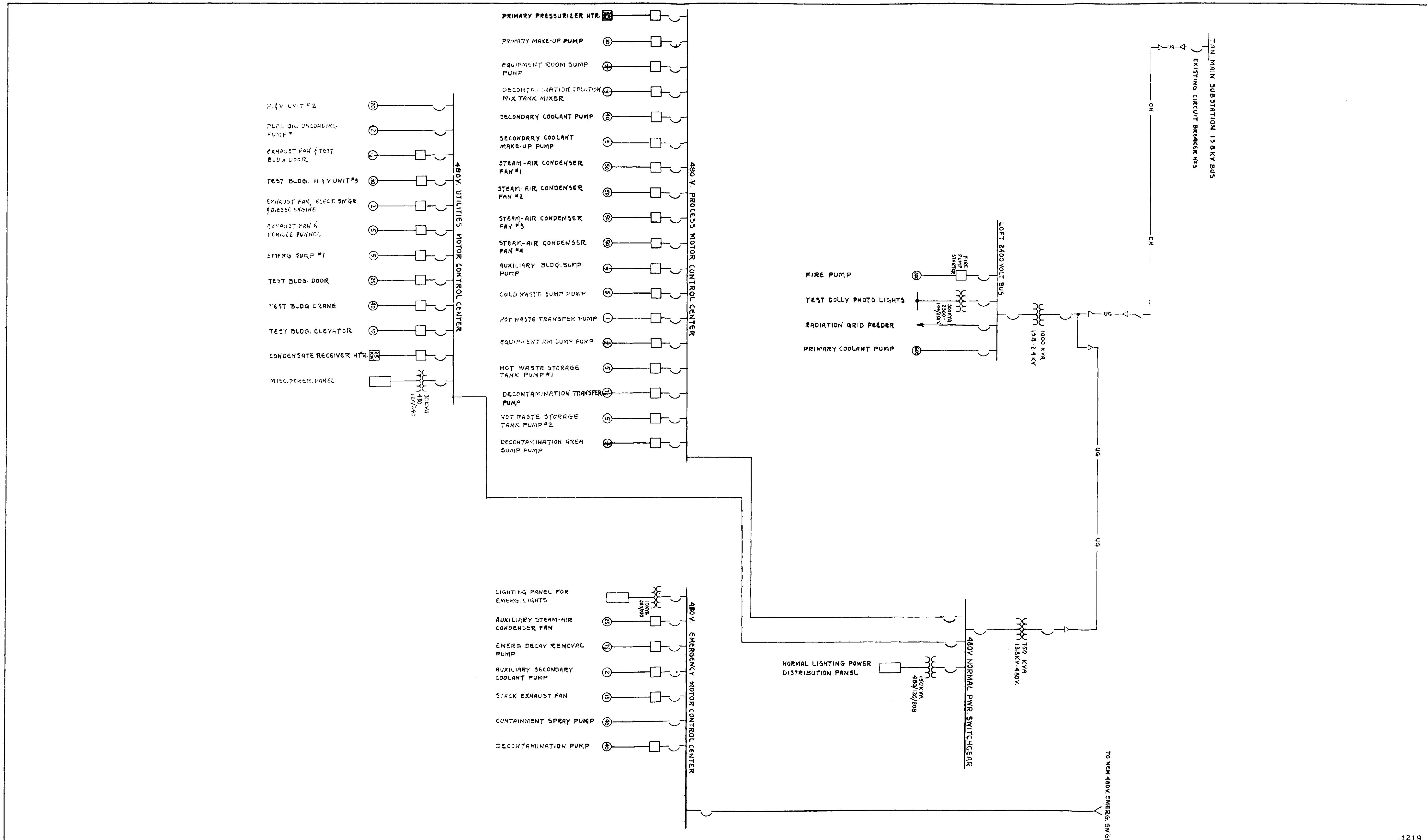


FIG. 47 LOFT ELECTRICAL DISTRIBUTION SCHEMATIC.

panel boards. This system operates continuously and independently of the normal power system during periods when test operations are in progress to supply power to FET and LOFT critical auxiliaries, emergency lights, instruments, and controls which are required for safe shutdown of the test reactor on loss of normal power. The reactor safety system 120 V AC emergency power bus is supplied from this system.

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

4.7 Communications and Alarm Systems

Commercially installed and serviced telephone systems connected to the NRTS commercial system provide for operational and emergency 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 are provided. The master station is capable of speaking to all locations singularly or collectively. Space is provided on power line poles entering the LOFT area for telephone cables.

Manual automatic coding fire alarm systems are strategically located throughout the LOFT-FET area and consist of alarm boxes connected to centralized relay and terminal strip units. The coded fire alarms annunciate at the TAN fire station, at the main TAN security station, and at the NRTS central fire station as well as at the reactor control room. Alarms are actuated through lines in commercial telephone cables.

The evacuation siren located in the test area is actuated from the reactor control room or the TSF area guard house and annunciated at the TSF area and LOFT-FET area.

4.8 Fuel Storage Tanks (Figure 41)

The two 35,000 gallon boiler fuel oil storage tanks and the 10 gpm pumps originally installed in the FET facility were adequate to meet LOFT requirements as was the original diesel fuel oil system.

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