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Summary Report

Nuclear Safety Research
and
Development Program

November 1964

DIVISION OF REACTOR DEVELOPMENT

★ UNITED STATES ★
ATOMIC ENERGY COMMISSION

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Summary Report

NUCLEAR SAFETY RESEARCH
AND
DEVELOPMENT PROGRAM

November 1964

Division of Reactor Development

UNITED STATES ATOMIC ENERGY COMMISSION

FOREWORD

This report was prepared by the members of the Nuclear Safety Group under the direction of Dr. Joseph A. Lieberman, Assistant Director for Nuclear Safety, in the Division of Reactor Development. It provides a current status summary of the coordinated research and development activities under the cognizance of the Nuclear Safety Group, with some coverage of past accomplishments and plans for the future.

This program is being carried out to provide a technological basis for the incorporation of the highest degree of public and personnel protection, consistent with economic and military requirements, in all the varied applications of reactors and other nuclear devices in the over-all program of this Division. This report contributes to the dissemination of information on the Atomic Energy Commission's various programs, which is the basis for a continuing and mutually beneficial exchange of ideas between the Atomic Energy Commission, its advisory bodies (e.g., the Advisory Committee on Reactor Safeguards), and the nuclear industry. An appreciable effort has been expended in maintaining adequate flexibility in the technical direction of the Nuclear Safety Research and Development program to permit response to the recommendations of the ACRS and the changing needs of the industry.

An indication of the high degree of AEC-industry cooperation in this program is shown by the current assignment of industry personnel to four of the principal projects in the program, viz., the Loss-of-Coolant Fluid Test (LOFT), the Special Power Excursion Reactor Test (SPERT), the Nuclear Safety Pilot Plant (NSPP), and the Containment Systems Experiment (CSE).

Principal contributors to this report were P. Cybulskis, H. L. Hamester, and R. R. Newton for Reactor Safety Research and Development; E. H. Davidson, W. K. Kern, and C. P. McCallum for Engineering Field Tests; H. Bernard, W. H. Regan, D. C. Costello, and I. Van der Hoven for Effluent Control Research and Development; J. L. Mershon for Analysis and Evaluation; and I. E. Jackson, Jr.

This report supersedes that bearing the same title dated June 1962.

Frank K. Pittman, Director
Division of Reactor Development
U. S. Atomic Energy Commission

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PART I

INTRODUCTION

I. INTRODUCTION

Recognition of the dominant importance of safety considerations in the location, development, design, construction, and operation of nuclear facilities and the need for a coordinated research and development effort in relevant technological areas provides the incentive for the program described in this report. The increasing scope, variety, and rate of growth of nuclear energy applications in land, marine, and aerospace environments also generates an increasing need for adequate information related to the safety of nuclear systems.

The Nuclear Safety Research and Development Program therefor includes a comprehensive analytical, experimental, and engineering-scale test effort to develop information essential to provide an adequate basis for the safe design, construction, and operation of nuclear facilities. Technological areas covered include reactor kinetics, chemical reactions (e.g., metal-water, metal ignition, fission product phenomena), reactor containment, fast reactor safety; integral engineering-scale testing of terrestrial and aerospace nuclear systems; environmental studies of radioactive effluents, low and intermediate level waste management, high level waste management, gaseous and aerosol effluents; and nuclear safety analysis and evaluation methods and procedures.

This work is being carried out under four major subprograms. Each of these was initiated at a different time as follows:

- (a) Effluent Control Research and Development which was initiated in 1946 as a radioactive waste disposal program under the AEC Sanitary Engineering Program.
- (b) Reactor Safety Research and Development which was initiated in 1954, following the planned nuclear excursion which destroyed BORAX-I, as the SPERT subprogram under the AEC Reactor Engineering Development Program.
- (c) Engineering Field Tests which were initiated in 1959 with the creation of the Aerospace Nuclear Safety Board to prepare a coordinated aerospace system safety program.
- (d) Analysis and Evaluation which was initiated in 1961 concurrently with the establishment of the over-all Nuclear Safety Program.

These origins indicate the evolution of the Nuclear Safety Program as a response to current or anticipated needs in the support of other development programs for nuclear systems.

Significant accomplishments have been attained in the evolution of the older subprograms. For example, activities in Effluent Control Research and Development have provided assurance that an increasingly large output of radioactive wastes is being safely discharged into the environment or

stored. The SPERT program has made available a significant amount of information on the kinetics of water-cooled and moderated reactor systems and has also provided a basis for planning the larger scale engineering field tests. Since the initiation of the coordinated program in 1961, particular emphasis has been placed on engineering scale, integral tests involving potentially destructive energy releases to obtain information related to the possible consequences of postulated severe accidents, including the "maximum credible accident", and the performance and reliability of engineered safeguard devices incorporated in reactor systems. This emphasis is dictated by the siting problems related to the Civilian Power Reactor Program and the complex safety problems related to the launching and controlled flight of aerospace nuclear systems, including re-entry into the earth's atmosphere. Emphasis has also been placed on the effective exchange of information in the Nuclear Safety Program. Important and useful working relationships have been established with the Advisory Committee on Reactor Safeguards through the Subcommittee on Safety Research, the nuclear industry through the Reactor Safety Committee of the Atomic Industrial Forum, and with the regulatory staff.

The following Parts II through V summarize activities under the four major subprograms of the Nuclear Safety Program.

PART II

REACTOR SAFETY RESEARCH AND DEVELOPMENT

II. REACTOR SAFETY RESEARCH AND DEVELOPMENT

The Reactor Safety Research and Development program involves both basic and applied research work in nuclear reactor safety, including studies in reactor kinetics, fuel meltdown phenomena, fission product release, and reactor containment. The data and knowledge obtained is applicable to the design and safety evaluation of thermal as well as fast reactors. Some of the research results are also applicable to the safety analysis of non-reactor accident events including criticality assessment in chemical processing plants, safety in fuel handling and storage, and ignition characteristics of plutonium and other isotopic power sources.

A. Reactor Kinetics

The reactor kinetics research effort is centered around the SPERT (Special Power Excursion Reactor Test) program which was initiated in 1954 following the planned nuclear excursion which destroyed BORAX I and revealed the need for the ability to predict the course and consequences of such excursions. This program is carried out by the Phillips Petroleum Company at the National Reactor Testing Station in Idaho and now includes four SPERT reactors. The characteristics of these reactors and the SPERT program are summarized below.

SPERT I (see Fig. 2-1) is an open tank light water moderated heterogeneous reactor designed to investigate reactor transient behavior through the destructive range. Seven different cores have been tested in SPERT I since its initial operation in July 1955, including high enrichment uranium-aluminum alloy, plate-type cores, a stainless steel clad cermet core, and a low enrichment uranium oxide fuel, stainless steel clad pin-type core. Three destructive tests have been conducted in this facility as follows:

1. In November 1962, the first such test was performed with a high enrichment uranium-aluminum alloy core of plate geometry. The excursion in this test reached a peak power of 2500 megawatts on a period of 3.2 milliseconds with a nuclear energy release of 32 megawatt-seconds. Approximately 35% of the core melted and the core and its support structure were violently disassembled by a multithousand psi pressure pulse. Water was ejected from the open vessel to a height of about 80 feet, and the control rod drive mechanisms were damaged. A top view of this core before the destructive test is shown in Fig. 2-2. The extent of physical damage to the core as a result of this test is shown in Fig. 2-3. The nuclear characteristics of the self-limiting power burst were approximately as predicted. However, the extensive damage observed was produced by the very large pressure excursion on a short-rise-time, which followed about 15 milliseconds after the peak of the self-limiting power burst, and which appears to have been the result of a purely thermal interaction between the molten fuel and water. The extent of metal-water reaction was small. Fission product release to the atmosphere and radioactive contamination from the low-fission-product-inventory core were insignificant and did not impair recovery of the fuel, core components, and instrumentation from the reactor vessel.

Figure 2-1

SPERT I Type "A" U-Al Alloy Plate Core

Figure 2-2

View of SPERT I with Type "D" U-A1
Alloy Plate Core before Test

Figure 2-3

View of SPERT I with Type "D" U-Al
Alloy Plate Core after Test

2. The second destructive test was conducted in November 1963 with a low enrichment pin-type UO_2 fueled core. This core had previously undergone a 3.2 millisecond excursion without suffering any damage thus demonstrating the efficacy of the Doppler effect as a shutdown mechanism. The oxide core destructive test attained a peak power of 20,000 megawatts on a period of 2.1 milliseconds with a total nuclear energy release of 150 megawatt-seconds. The excursion was self-terminating with the Doppler effect being the major shutdown mechanism. Only limited damage was sustained by the core as a result of this test; about 25% of the fuel rods in the core underwent some bowing, two of the rods were split open. The peak pressure during this test was 65 psi.

3. The third test in this destructive series, the second attempted destruction of the low enrichment UO_2 -SS clad pin-type core, was carried out in April 1964. A peak power of 35,000 megawatts was observed on a period of 1.55 milliseconds with a total nuclear energy release during the burst of 165 megawatt-seconds. This excursion was also self-terminating due to the combined effects of Doppler broadening and steam formation, which resulted in a more rapid power reduction and a lower total energy release (by about 50%) than would be expected in a purely Doppler-controlled burst.

These three tests demonstrated the superiority of SS clad UO_2 rod fuel cores over metallic U-alloy plate cores with respect to resistance to damage in nuclear excursions.

SPERT II (see Fig. 2-4) is a medium pressure and temperature (300 psi, 400°F) facility with forced circulation capable of operation with light or heavy water as the moderator-coolant. It was placed in operation in 1960 with the purpose of investigating the effect of the prompt neutron lifetime on reactor kinetic behavior. Transient tests are being conducted on a high enrichment plate-type core with heavy water coolant. The expanded D_2O core experiments have been completed. Due to the substantially longer prompt neutron lifetime of the expanded D_2O core, larger reactivity insertions are required to produce a specified reactor period than in light water moderated cores. Consequently, the self-shutdown effect must also be greater to avoid core damage. Fuel distortion is observed at periods of about 50 milliseconds in heavy water moderated cores whereas periods down to 10 milliseconds did not yield any damage with similar fuels in light water moderator. Currently excursion tests with a close-packed D_2O core are in progress in SPERT II facility. At the conclusion of this test series, about September 1964, the heavy water moderated reactor kinetics program will be terminated.

SPERT III (see Figs. 2-5 and 2-6), completed in December 1958, is a high pressure and temperature (2500 psi, 650°F) plant with forced circulation (20,000 gpm) to 60 MWT for short periods of time. After an extended shutdown due to pressurizer vessel failure, the SPERT III facility resumed operation during 1963. The present core consists of fully enriched UO_2 dispersed in stainless steel and clad in stainless steel. Kinetic experiments from high initial powers and flows will be followed by stability and loss of flow

Figure 2-4

SPERT II Reactor Vessel Assembly

Figure 2-5

View of SPERT III Facility

II-7

Figure 2-6

SPERT III Reactor Vessel Assembly

experiments. A low enrichment uranium oxide, stainless steel clad core of the type used in power reactors, will be installed during the fall of 1964.

SPERT IV shown in Fig. 2-12 is a large open pool facility with controlled flow which started operation in June 1962. It is currently investigating the effects of coolant flow rate, initial temperature, and hydrostatic head on the onset of reactor instability. In the fall of 1964, a low enrichment oxide core with a central exposure facility will be installed; it will be utilized to conduct capsule experiments on the phenomena associated with the rapid meltdown of reactor fuel samples. This program will be a precursor to the more extensive subassembly meltdown program planned for the Power Burst Facility.

The Power Burst Facility (PBF) is a planned transient test facility designed to generate power bursts with initial asymptotic periods as short as 1 msec, producing energy releases large enough to destroy entire fuel subassemblies placed in a capsule or flow loop within the reactor, all without damage to the reactor itself. It would be used to evaluate the consequences and hazards of very rapid destructive accidents in power reactors. It would also be used to carry out detailed studies of non-destructive reactivity feedback mechanisms in the short-period domain. The conceptual design as well as a considerable amount of optimization have been completed; construction is planned to start in early 1965 and is expected to take about 20 months to complete.

Under the Kinetic Studies of Heterogeneous Water Reactors project at Space Technology Laboratories, single channel, out-of-pile experiments are conducted to study power-void, and void-flow transfer functions in order to determine the criteria for stability in boiling water reactors. The results of this project have been successfully applied to the explanation of the instabilities observed in the early SPERT I tests. The current effort is aimed at extending the theories to geometries and thermodynamic conditions representative of power reactors. Another aspect of this project is the detailed study of void growth and the mechanisms of pressure generation during short period reactor excursions. Both in-pile capsules in KEWB as well as out-of-pile laboratory studies are being utilized for this purpose.

The dynamics of void formation during high heat flux transients are under investigation at the University of California (Berkeley). The objective of this work is the quantitative prediction of the steam void volume during power excursions in boiling and pressurized water reactors. The variables investigated include temperature, pressure, flow, and rate of energy addition.

The Doppler coefficient project at Atomics International is aimed at providing a firm experimental basis for the calculation of the Doppler coefficient of reactivity. Initial effort is being devoted to the measurement of the Doppler coefficient of simple resonance absorbers in an idealized $1/E$ neutron spectrum by both the activation and reactivity

oscillator methods. The results of these measurements will be compared with the best theoretical calculations available. After the experimental techniques have been proven out, measurements will be extended to various fuel materials up to temperatures of 1000°C.

In addition to the above-described active projects, a number of kinetics studies have been undertaken and have been completed or terminated. These include the experimental study of the kinetics of aqueous homogeneous reactors (KEWB); study of pressure transients, flow stability, and void formation in organic coolants; survey of safety problems applicable to gas cooled reactors; criticality hazards in fuel handling; analytic studies of space dependent reactor kinetics, etc.

B. Metal-Water Reactions

Exothermic metal-water reactions would occur as a result of a major reactor accident in which molten reactor core metals are dispersed into the cooling water. The energy released by such reactions could exceed that released by the fission process during a nuclear excursion. A three-phase program (see Table I) is being carried out at ANL to provide a practical means for estimating both the rate and extent of metal-water reactions that will occur during postulated reactor accidents. The first phase involves the experimental determination of the isothermal rate laws and rate constants. The second phase involves experimental observation (including in-pile tests) of the non-isothermal reaction with small particles in order to simulate accident situations under idealized conditions. This serves as a check on computed results and detects any physical limitations to the reaction rate not apparent in isothermal experiments. In addition, the in-pile experiments in the TREAT facility using test assemblies as shown in Fig. 2-7 yield information as to the nature and degree of fragmentation of fuel element specimens under excursion accident conditions. Fig. 2-8 shows the original configuration of a typical specimen before test, and the final configurations of two specimens which have experienced a TREAT excursion, although it should be noted that the major portion of the metal-water reaction occurs with small particles which are not shown. Such information is necessary for the third phase of the program which is the development of a mathematical model which combines isothermal rate laws with equations of heat and mass transfer so that the course of the reaction can be computed under any set of initial conditions. Table I shows that studies are being made on zirconium, aluminum, stainless steel, and uranium.

Experimental results for the zirconium-water reaction indicate that both the extent and rate of reaction is determined largely by the size of the reacting particles. Only thin oxide films are formed on the larger specimens while extensive reaction (sometimes at explosive rates due to self-heating) occurs when finely divided metal is used. Analysis indicates that the reaction is initially controlled by the rate of gaseous diffusion of water vapor toward the hot metal particles, through the hydrogen generated by the reaction. As the reaction proceeds, it follows the parabolic rate law, indicating that the oxide film is protective; the failure to achieve

TABLE I

STATUS OF METAL-WATER REACTIONS RESEARCH PROGRAM

Metal	Phase I Determination of Isothermal Reaction Rate Law				Phase II Study of Non-isothermal Reactions			Phase III Theoretical Calculation of Rate and Extent of Reaction ^{1/}
	Pressure Pulse Method	Volumetric Method	Levitation Melting Method	Other Institution	Condenser Discharge Method	Laser Heating Method	TREAT Studies	
Zirconium	-	-	-	Available 1000-1850°C	Completed	Will be attempted	Work in progress	Some work successfully done
Aluminum	Completed 800-1200°C	-	Completed 1200-1800°C	-	Not applicable	Will be attempted	Work in progress	Will be attempted
Stainless Steel	-	Work in progress ^{2/}	-	-	Completed	Will be attempted	Work in progress	Will be attempted
Uranium	-	Completed 400-1600°C	-	-	Completed	Will be attempted	Work in progress	Work in progress

^{1/} It is planned to extend these calculations to cases of full-scale reactor accidents.

^{2/} Volumetric method not completely satisfactory and further method development work may be necessary.

Figure 2-7

Typical TREAT Test Assembly
for Metal-Water Reaction Studies

Figure 2-8

TREAT Metal-Water Meltdown Specimens

II-13

100% reaction by any experimental method other than for very fine particles suggests that this protective coating is retained even at temperatures beyond the melting point of the oxide (2700°C). At these temperatures, however, larger particles increase in their tendency to fragment causing an over-all increase in reaction rate. No important differences between zirconium and the Zircaloy-series alloys have been observed. Further, variation of steam pressure between 200 and 1000 psig do not alter the results to any large extent. A marked dependence, however, is observed on the extent and rate of reaction as a function of initial (bulk) water temperature. Explosive reactions are found with particles smaller than 1 mm in heated water (up to 300°C), but only for particles smaller than 0.5 mm in room temperature water. This is attributed to the increased rate of gas-phase diffusion resulting from the violent motion of such small particles. A similar result was found in the uranium-water reaction.

Results for the uranium-water reaction indicate a rapid linear reaction at 400°C, but a nearly parabolic rate between 500 and 1200°C. (Experiments have recently been extended to 1600°C.) Meltdown tests of uranium in the TREAT facility indicate that three factors control the extent of the uranium-water reaction: the total fission energy input, the reactor period, and the metal surface area created by fragmentation. For a given energy input, more reaction occurs at shorter periods since the reduced time for heat loss to occur results in higher peak temperatures. This effect is not as important in oxide fuel meltdowns because of the lag in transferring heat from the oxide to the cladding.

The results of aluminum-water reaction studies suggest that the peak aluminum temperature determines the nature of the reaction. Isothermal studies show that the oxide film is very protective up to 1300°C. The reaction at temperatures above 1750°C, however, is more vigorous than the metal-water reactions of either zirconium or uranium; the violence of the high temperature aluminum-water reaction is attributed to vapor-phase combustion. Meltdown tests in TREAT have been performed with aluminum, aluminum-clad fuel specimens, and aluminum-uranium alloys. The data of these tests can be correlated well with the isothermal studies. Further, less metal-water reaction occurred in tests with aluminum-clad mixed oxide fuel specimens than with Zircaloy-clad mixed oxide fuel specimens; it seems evident that the aluminum-clad did not reach 1750°C, where ignition and burning in water could occur even though the mixed oxide received energies sufficient for melting the ceramic (2600°C). No violent pressure surges have been observed in the TREAT experiments, indicating that the over-all reaction rate (measured by the increase in pressure) was slow. Hence, it appears that the fine particles formed in the TREAT tests were formed over a period of time so that the over-all reaction was relatively slow even though each fine particle reacted rapidly. Current studies indicate that steam pressure (water temperature) causes a marked increase in the amount of aluminum-water reaction. For example, in two TREAT experiments in which the initial water temperature was 285°C (saturation pressure 1000 psi), the extent of metal-water reaction was 76% and 89%. When the nuclear energy input was sufficient to bring the specimen temperature somewhat above

1200°C; for similar experiments in room temperature water, less than 20% metal-water reaction was observed.

Condenser-discharge experiments with stainless steel wires show a limited amount of reaction for small particles (260 microns) even at stainless steel temperatures as high as 3700°C in heated water (the metal-water reactions for zirconium and uranium under similar conditions was more extensive). Further, the stainless steel reaction was not much more extensive with heated water than with room temperature water. This indicates that the reaction rate is controlled predominantly by a chemical rate law, even at very high temperatures (unlike the zirconium-water reaction which is controlled by the gaseous diffusion of water vapor through the evolving hydrogen). TREAT tests have been performed with specimens of unclad stainless steel- UO_2 cermet fuel pins and plates, and with stainless steel-clad mixed oxide and pure UO_2 fuel pins.

Future plans call for isothermal rate studies for the stainless steel-water reaction and for the aluminum-water reaction as a function of pressure. A new apparatus is under construction for this purpose. Additional TREAT runs will be made; one variable which is being studied is the effect of burnup on metal-water reaction rates. As more experimental results are obtained, the data will be applied to postulated (and real) reactor accidents. In this connection, calculations of the extent of aluminum-water reaction occurring in the SL-1 and SPERT ID destructive transients have been made; the calculations are in good agreement with the experimental analysis of the reactor debris in both cases.

C. Metal Ignition and Burning

Numerous incidents of reactor metals igniting (both in connection with reactor operations and during fabrication and storage) have required studies of the factors influencing the ignition of these metals. Such studies, carried out at ANL, have concerned three aspects: isothermal oxidation kinetics, ignition, and burning of metals. It is expected that these studies will be completed in July 1964. Studies on the air oxidation of zirconium, uranium, and plutonium (See Table II) indicate that the most important parameter is particle size.

The ignition of single pieces of zirconium has been studied by shielded ignition tests, in which an inert atmosphere is used during sample heating and then rapidly replaced by air. Ignition does not occur if the zirconium is heated slowly in the presence of air or oxygen, due to the low temperature formation of a tenacious, protective oxide coating which prevents oxidation at higher temperatures. Results suggest that zirconium ignitions are thermal explosions resulting from more heat being generated than can be dissipated. Computed ignition curves agree well with experimental values, and indicate that samples of high specific area (say 500 cm^2/g) ignite rapidly if the initial ambient temperature is above a critical value (about 425°C), but experience little self-heating if the temperature is below that value. Specimens of low specific area (say 5 cm^2/g) ignite

TABLE II
STATUS OF METAL OXIDATION AND IGNITION KINETICS PROGRAM

	Phase I Isothermal Oxidation Rates ^{1/}	Phase II Ignition				Phase III Air Burning			
		Single Pieces ^{1/}		Powder		Temperature Measurement		Propogation Rates	
		Exp.	Theory	Exp.	Theory	Exp.	Theory	Exp.	Theory
Zirconium	400-900°C in O ₂ completed	Completed	Completed ^{2/}	Completed	-	Completed	Completed ^{3/}	Completed	Completed ^{3/}
Uranium	125-625°C in O ₂ completed	Unirradiated completed	Completed ^{2/}	Completed	Completed ^{2/}	Completed	Completed ^{3/}	Completed	Completed ^{3/}
	600-1000°C in air in progress	Irradiated in progress							
Plutonium	140-450°C in O ₂ completed	In progress	-	-	-	-	-	In progress	-
	Above 450°C in O ₂ in progress								

^{1/} Includes studies with a large number of binary alloys.

^{2/} Ignition temperatures calculated from isothermal rate data.

^{3/} Burning temperatures and propagation rates calculated from heat transfer theory.

slowly above a critical temperature (about 1175°C); however, self-heating is considerable in samples that do not ignite. The effect of alloying additions on the ignition temperature of zirconium have been measured. The data for zirconium-titanium alloys is particularly striking; a 14.9% titanium-zirconium alloy ignited at 372°C, while a similarly-sized pure zirconium sample ignited at 786°C. In general, aggregates and powder compacts of metals ignite at lower temperatures than single pieces even when compared at equal specific areas. This is attributed to the decreased heat losses from particles in the center of the aggregate. Hence, a sometimes hidden hazard can exist in massive zirconium parts prepared by powder metallurgy if sintering has been inadequate. Theoretical values of burning temperatures of zirconium in air and rates of propagation of burning, computed on the basis that the reaction rate is controlled by the rate of gas-phase diffusion of oxygen through a nitrogen-rich boundary layer, are in reasonable agreement with measured values.

The ignition temperature of single uranium pieces in air has been measured by rising temperature ignition tests, the data showing a marked dependence on specific area. The results suggest that uranium ignitions result simply from an accumulation of heat generated by oxidation. The reaction appears to accelerate slightly between 300 and 450°C, and decelerate slightly above 450°C. Alloying additions of many different elements can have profound effects on the oxidation and ignition of uranium. For example, the addition of 1130 ppm (1 atom %) of aluminum reduced the ignition temperature from 595°C to 355°C and increased the oxidation rate by over a factor of 20. Apparently the aluminum tends to keep the accelerated reaction going at higher temperatures before the decelerating reaction (observed at 450°C in pure uranium) takes over. Copper, on the other hand, renders the post-transition oxide more protective and therefore decreases the oxidation rate above 500°C. Several alloys show either one of these effects, and some show both; 24 different elements were tested. Isothermal experiments of the uranium oxidation in air indicate a linear rate law at 500°C, but a parabolic (initially parabolic and later linear) law at 600, 700, 800, 900, and 1000°C; at the two higher temperatures, the parabolic rate was followed for much longer times. The data suggest that the reaction rate is controlled by an adsorption equilibrium at the oxide-gas interface and by the diffusion of oxygen ions through the oxide film. The transition from a parabolic to a linear reaction probably occurs when the protective oxide film begins to crack. Aggregation of uranium foils decreases the ignition temperature because of decreased heat losses (tests were run on both tightly packed foils and foils spaced 2.6 mm apart). In studies of the ignition of spherical uranium powders, ignition temperatures were found to decrease with increasing stack heights; nearly constant values were found at relatively small values of sample heights. Such limiting ignition temperatures were found to increase with specific surface area. Theoretical maximum burning temperatures and rates of propagation along foils and wires have been found to agree with experimental values, and are close to those found for zirconium.

The complexity of the metallurgy of plutonium metal (six stable phases at temperatures below its melting point) is reflected in its oxidation

kinetics. Isothermal oxidation studies have been carried out at temperatures corresponding to five of these solid metal phases. In the alpha phase (stable below 120°C), a protective film is rapidly formed. In both the beta phase (stable between 120°C and 210°C) and the gamma phase (stable between 210°C and 320°C), the oxidation kinetics initially follow a parabolic (parabolic and then linear) rate law. Appropriate analysis of the data shows that the same initial rate-controlling process is in effect over the temperature range from 140-300°C. In this range, oxidation follows the parabolic law as a protective oxide film of approximately one micron is developed. This initial layer remains compact and adherent as subsequent oxidation proceeds at an essentially linear rate. In the isothermal oxidation of beta-phase plutonium, an additional linear rate (both faster and temperature-independent) was observed to take place after the initial parabolic oxidation. This faster oxidation, which gave rise to a loose oxide powder, was probably caused by the presence of a small amount of water vapor in the gas phase. There is a change in the oxidation kinetics in the vicinity of 300°C which results in a sharp drop in the isothermal oxidation rate as the temperature increases from 350° to a minimum rate at 420°C. This change is attributed to a phase change in the oxide at 300°C, rather than phase changes in the metal phase (gamma-delta transition at 320°C and the delta-delta' transition at 452°C). Apparently the oxide(s) formed above 300°C are more protective than those formed at lower temperatures. Above 420°C, the isothermal oxidation rate again increases despite the presence of the more protective oxide(s); this is attributed to the increased diffusion rate of oxide ions through the oxide film at higher temperatures. Experiments indicate that once the protective oxide film is formed at the higher temperatures (above 420°C), further oxidation at room temperature is reduced by factors ranging between 9 and 50. Thus the hazards associated with preparing plutonium samples for storage could be reduced considerably by pre-oxidizing them at temperatures above 420°C. Ignition temperatures in air for pure plutonium cubes, rods, and foil specimens and for a series of binary plutonium alloys have also been measured. The plutonium ignition temperatures fall into two regimes, a high temperature regime at approximately 500°C for samples thicker than 1 mm and a low temperature regime at 300°C for thinner foils. This behavior is consistent with the oxidation behavior of plutonium mentioned above. Evidently, the rate of heat generation by oxidation in the region of 300°C is just sufficient to cause ignition of specimens having the larger specific areas. Specimens with smaller specific areas, however, have greater thermal inertia and the temperature increase caused by self-heating occurs more slowly. This allows formation of the more protective oxides at temperatures above 300°C. The resulting decreased oxidation rate thus prevents ignition from occurring until the sample temperature is increased by external heating beyond 450°C where the oxidation rate again rapidly increases with increasing temperature. Experiments with binary plutonium alloys indicate that the addition of aluminum and silicon increase the resistance to ignition and oxidation in the high-temperature regime, while carbon, manganese, and cobalt decrease this resistance; chromium, iron, nickel, cerium, and uranium had no effect. One ternary alloy which shows promising ignition characteristics has a uranium-plutonium ratio of about 2:1 with about 6 a/o molybdenum. This alloy did not ignite until a temperature of 603°C

was reached, even in the presence of moisture. Burning characteristics of plutonium appear to be very similar to those of zirconium and uranium.

D. Fission-Product Release, Characterization, and Behavior

Fission-product release, characterization, and behavior studies are being conducted at Atomics International, Brookhaven National Laboratory, and Oak Ridge National Laboratory. The original out-of-pile studies identified the major parameters which control the extent of fission-product release. Such studies permitted effective planning in the higher cost in-pile tests; the chemical and physical forms of the fission products are being identified in these in-pile experiments (and in additional out-of-pile tests). These fission-product forms are, in turn, simulated in more complex out-of-pile experiments (e.g., NSPP and the Hot Cell Containment Mockup) which investigate fission-product transport and deposition phenomena. These programs will indicate the parameters which govern fission-product transport behavior and will develop various measurement techniques. Such results and techniques will then be available for the effective planning of a fission-product transport behavior program in which fission products are released under conditions closely simulating reactor accidents, i.e., on the destruction of fuel-pin bundles by gamma decay and/or fission heating in appropriate environments. A few facility (described in the last paragraph of this subsection) is being considered for this purpose.

Results of out-of-pile experiments at BNL and ORNL indicate that the following parameters influence both the extent and the character of fission-product release -- fuel type, cladding, atmosphere, peak temperature, time at temperature, and burnup. Variation of these parameters indicates three release mechanisms - via diffusion and subsequent vaporization of the fission products, on fuel oxidation, and on fuel melting. The fission products which are released generally fall into one of three categories. These are the volatiles (such as the noble gases and the halogens which are invariably released to a larger extent than other fission products), the volatile metals and oxides (such as cesium and ruthenium which show variable releases), and the non-volatiles (such as barium and strontium which are invariably released to a small extent). Typical results indicate that release via diffusion from UO_2 in inert atmospheres follows the diffusion laws up to 1700°C , but accelerates at higher temperatures. This acceleration has been attributed to grain-growth. For example, 12, 17, and 42% of the fission-product iodine were released from trace-irradiated UO_2 fuels via diffusion over 5.5 hours at 1780, 1900, and 1980°C , respectively; at the latter temperature, 12% of the rare gases, 76% of the tellurium, 15% of the cesium, and 13% of the ruthenium were also released. Fission-product release on oxidation by air is significant at temperatures below 1500°C . The complete air oxidation of trace-irradiated UO_2 (to U_3O_8 over a 1.5 hour period) at 1400°C , for example, released 9% of the rare gases, 22% of the iodine, 29% of the tellurium, 3% of the cesium, and 99% of the ruthenium. (The high release of ruthenium on oxidation of UO_2 is attributed to formation of the volatile oxide RuO_4). Such a release mechanism would be important in a reactor accident for those portions of the core in which cladding melting temperatures but not fuel melting temperatures were reached; in this case, a substantial fission-product release could occur as the fuel

temperatures cooled from 1500°C to lower temperatures. The effect of steam on release via the air oxidation of UO_2 is currently being investigated. Maximum fission-product release occurs on fuel melting and vaporization. Extensive UO_2 melting experiments in helium (melting time, approximately 2 minutes) show nearly quantitative release of the rare gases, halogens, tellurium, and cesium, a large release of ruthenium, and small (generally less than 1%) of the barium, strontium, and rare earths. The data listed above for release via diffusion and UO_2 oxidation are somewhat misleading in that the numbers refer to trace-irradiated fuel; the extent of release via these mechanisms goes up markedly as the burnup level is increased. For example, the iodine release via diffusion is 12%, 25%, and 59% at 1780°C over 5.5 hours for tracer, 1000 Mwd/T, and 4000 Mwd/T burnup levels, respectively. The releases on the complete air oxidation at 1200°C of UO_2 pre-irradiated to 4000 Mwd/T were 80% of the rare gases, 94% of the iodine, 66% of the tellurium, 6% of the cesium and 99% of the ruthenium. It should also be noted that these release values are for relatively small samples; results show that the release rates decrease somewhat with sample size. This result is not unexpected inasmuch as the specific area generally decreases with sample size. The largest effects were associated with cesium and the rare earths. Yet another parameter which influences the extent of release is the type of cladding used. On melting clad UO_2 specimens in helium, stainless steel appears to segregate from the UO_2 , whereas zirconium seems to interact with it. The zirconium, under these conditions, increases the strontium release by a factor of ten (presumably by removing residual oxygen, thus stopping the formation of the non-volatile strontium oxide). Out-of-pile release studies have also been conducted on U-Al alloy fuels and U-Mo alloy fuels. Currently studies are being made on both UO_2 and UC fuels. In another investigation carried out at Atomic International, attention has been focused on the retention ability of molten sodium for iodine as a function of temperature. Here fission products are released via diffusion and recoil from pre-irradiated uranium fuel disks into molten sodium (900-1400°C) under a flowing helium atmosphere.

ORNL personnel are carrying out in-pile meltdowns of miniature stainless steel clad UO_2 fuel pins in both the ORR (see Fig. 2-9) and TREAT facilities to simulate the loss-of-coolant accident and nuclear excursion accident, respectively. Past experiments have been carried out in inert atmospheres; present experiments are being carried out in air, air-steam, and water environments. Both the extent of release and the chemical and physical form of the fission products are measured via diffusional deposition, by hold-up on particulate filters, and by adsorption on charcoal beds; parameters which will be investigated include atmosphere, cladding, peak temperature, time at temperature, and burnup. Typical data for the meltdown (and partial vaporization) of a trace-irradiated stainless steel clad UO_2 specimen (molten period, 5 minutes) in flowing helium show the release of over 90% of the ruthenium, tellurium, cesium, and iodine, and over 70% of the strontium, zirconium, barium, cerium, and UO_2 from the original fuel pin. The data, however, indicate that less than 1% of the strontium, zirconium, cerium, barium, and UO_2 , 4% of the ruthenium, 80% of the iodine, 89% of the tellurium, and 58% of the cesium were released from the "high-temperature zone" (minimum temperature, 1000°C) to cooler portions of the experimental

Figure 2-9

Furnace for ORR Meltdown Experiments

assembly. The chemical and physical form of many of the fission products are characterized by measuring their distributions among the various regions of the experimental assembly. Such data is interpreted in terms of fractionation processes, i.e., the alteration of radionuclide composition occurring during the experiment which result in a sample composition different from that predicated for normal fission yields. Such analyses permit one to establish whether the release of a given fission product, say strontium, took place as the metal, the oxide, or associated with a particulate. The results of experiments to date indicate that an increase in sweep gas (helium) flow rate changed the character but not the extent of release; the majority of fission products were released as elemental or oxide vapors rather than being associated with particles at the higher flow rates. A similar set of experiments has been carried out in the ORR on the air burning of pyrolytic-carbon-coated UC_2 particles embedded in a graphite matrix. In another investigation, carried out by AI, pre-irradiated uranium fuel disks are immersed in a capsule of molten sodium (at various bulk sodium temperatures). The fuel is then melted by subjecting the capsule to a neutron burst in the KEWB reactor facility. Results to date indicate that from 1-10% of the pre-burst iodine inventory was released to the sodium, and less than 0.02% of the iodine in the sodium was released to the flowing helium cover gas.

Out-of-pile fission-product characterization studies are being carried out at BNL and ORNL. Studies at BNL have included measurement of the condensation temperature of specific fission products (such as iodine, tellurium, cerium, and barium) released on melting metallic U fuel and U-3.5% Mo alloy fuels in various atmospheres. These studies show that the majority of the iodine released in helium on melting pre-irradiated U and U-Mo alloy fuels deposited at 250-350°C; evidence indicates that such deposits are made up of uranium iodides. On melting these fuels in air, however, over 90% of the iodine deposited at temperatures below 50°C; such deposits have been identified as molecular iodine. Further, the melting of pre-irradiated U fuel in steam at various burnup levels resulted in 85-90% of the iodine in the aqueous phase as hydrogen iodide. This is attributed to a reaction between iodine and hydrogen; the latter is formed by the U-steam reaction. Current studies of a somewhat similar nature are being carried out at BNL on UO_2 fuels in addition to studies on the diffusion-release of fission products from UC fuel. At ORNL, progress has been made in the study of the anomalous behavior of iodine at low concentrations. Much emphasis is placed on the study of iodine since the combination of the large fission-product iodine release (expected in a major reactor accident) and the extreme biological hazards associated with radioiodine, currently determine the siting criteria for power reactors. The release of fission-product iodine to a containment shell even as the result of a major loss-of-coolant accident would, however, result in low iodine concentrations over extended periods of time, inasmuch as the release would be spread out over long times. Low concentration iodine has been prepared in clean laboratory systems under a variety of conditions; in each case, similar anomalous fractions are found. The relative amount of the

anomalous forms increases as the total iodine concentration decreases (approximately 30% at 1 ug/m³ in both air and helium). The chemical nature of these species is currently being investigated, and the retention ability of various activated charcoals for these species is being measured. Preliminary experiments indicate that the introduction of non-purified (industrial) atmospheres increases the fraction of anomalous iodine species. This has been attributed to interaction of the iodine with organic impurities.

The chemical and physical forms of the fission products which have been found in the experiments discussed above (both in-pile and out-of-pile) are simulated in the complex out-of-pile experiments which investigate fission-product transport and deposition phenomena. In the ORNL Hot Cell Containment Mockup, U-Al alloy fuels (and more recently stainless steel clad UO₂ fuels) are melted (or oxidized) by either induction or resistance heating, depending on the fuel and ambient atmosphere. The released fission-product cloud is transferred to a 180 liter tank (containment mockup) where aging effects take place. The time and space dependent behavior of the fission products (e.g., deposition and aerosol agglomeration, etc.) are determined. Special emphasis is being placed on the adsorption-desorption behavior of fission-product iodine on both containment walls and particulates. In addition, some quantitative information of anomalous iodine species is obtained, and the retention efficiencies of various filter-charcoal combinations is checked. Parameters which will be varied in these studies are cladding (stainless steel vs. zirconium), peak fuel temperature, burnup, and the pressure, temperature, ionizing radiation level, and surface composition of the containment tank. Current studies emphasize clad UO₂ fuel specimens in air-steam atmospheres. Larger scale experiments of a similar nature will be run shortly in the NSPP. Here fuel destruction will be brought about by heating with a plasma torch - a tool which is flexible enough to generate many of the physical and chemical forms of fission products found in the in-pile release experiments.

A new facility is being considered to further investigate fission-product behavior under conditions which closely simulate a major reactor accident. Such a facility would produce the meltdown of bundles of fuel pins within a reactor loop, driven by either fission heat or gamma decay heat depending upon the release mechanism to be investigated (i.e., diffusion, oxidation, or melting). The reactor core surrounding the loop would both produce the fission-product inventory, and supply the power necessary for the melting experiments. Loop atmospheres would include realistic steam-air mixtures. Further, the products of release would be fully characterized by sampling both in the vicinity of the initial release (simulating the reactor vessel region), and, after transport, in a simulated containment which would be sized to produce realistic activity concentrations. The realistic fission-product releases would allow the testing of filters under realistic accident conditions, and would yield information on the deposition of fission products in both the reactor core simulant and containment simulant. Such an experimental program would aid in extrapolating the results of experiments such as the NSPP and CSE to real reactor systems, because it

would produce realistic fission-product inventories. Further, it would help make possible the extrapolation of the single-point LOFT experiment to other reactors and different reactor conditions since the fission products would be released over a wide range of conditions.

E. Reactor Containment

Research and development programs on reactor containment are directed toward a fundamental understanding of the mechanisms of the various steps in hypothesized maximum accidents, the hazards from various credible accidents, and the means of restricting those hazards to the immediate reactor vicinity. These programs include both older work, based on a philosophy of complete standard containment of the upper limit accident and newer work based on refinements of the maximum credible accident and alternative types of containment (e.g., pressure suppression, pressure relief, and others). The older work on upper limit accident phenomena (such as shock loading) is being de-emphasized and newer studies directed to demonstrating the actual course of accidents and the engineering safeguards and countermeasures which can be employed to reduce the consequences of any accident are being intensified. Through such study, it is anticipated that containment costs may ultimately be reduced and reactor siting improved and made more economical without compromising safety. Theoretical analysis and scale-model experimentation at small and large scale are being used with major emphasis on defining (1) the likelihood and extent of a failure of a reactor primary system conducive to a serious accident; (2) the energy releases associated with the most credible accidents; (3) the various resulting pressure, shock, and thermodynamic loadings which must be effectively resisted by the containment system without significant loss of integrity; (4) the ultimate strength, load and leakage resistance, and energy absorption capacity of containment devices; and (5) the post-accident transport and behavior of fission products constituting the major safety problem, and the efficiency of containment devices which can be employed to reduce the mobility of radioactivity and its escape from various containment schemes.

Older Studies of Upper Limit Accident Phenomena

Naval Ordnance Laboratory (NOL) has completed a study to determine the response of reactor primary vessels to very high transient internal pressures, which could be generated by a reactor accident such as a violent excursion. It was postulated that the maximum resistance of such vessels was a strong function of the plastic strain-energy absorbing potential of the vessel material. A generalized mathematical technique for computing the plastic strain-energy of idealized dilated cylindrical vessels with rigid end caps and no penetrations was developed, and tests of small model vessels at several scales were conducted to define the maximum energy (released at near-explosive rates) that a vessel could be expected to contain without losing its integrity. It was found that about twice as much energy release can be sustained by a vessel capable of large plastic strain as by one with small capability. In addition, in a vessel with large capability, about 16% more energy can be sustained

dynamically than statically. A method of predicting the maximum energy-retention of cylinders has been developed which is based on material properties, vessel size and thickness, high strain rate, scatter in test data, maximum strains that can be accepted without impairing vessel function, and the effect of idealized penetrations. NOL has also completed 1/30 and 1/15 scale model tests to determine the response of the top shield plug of the Fermi reactor to an upper limit accident (meltdown and supercritical re-assembly in the reactor) leading to energy releases equivalent in rate and amount to 1000 pounds of TNT. The effects of various natural mechanisms (e.g., strain energy, temperature rise, kinetic energy generation) which tend to absorb portions of accident-released energy and thus reduce loadings to be withstood by reactor components were assessed. It was qualitatively found that such absorption is quite significant and can be increased by lower initial temperatures in the models or by increasing the amount of material present to act as a heat sink. Because of the difficulties in quantitatively assessing effects of the individual energy absorption mechanism, the tests studied the already conservatively defined accident so as to predict an upper limit plug response. However, the result was still less than previous estimates and, as a result, the reactor design could incorporate a safety device capable of accepting the more accurate upper limit response.

Aberdeen Proving Ground has completed model tests and analyses of the response of thin cylindrical and spherical shells to high level transient internal pressures to determine upper limit load capacities in the elastic and plastic ranges of dilation. The elastic vibrational modes have been predicted and confirmed by tests, and an empirical attempt has been made to slow the plastic response to spherically symmetric and eccentric loading. Tests conducted on buried and submerged shells to show the effects of external support on shell response showed only a limited contribution to strength. The ability of shell linings to attenuate or change the characteristics of transient loadings to increase the structural resistance of the shell was qualitatively shown.

Stanford Research Institute has completed a study of missiles that might be generated as a result of violent, upper limit nuclear excursions which could rupture reactor vessels and surrounding structures. Chemical explosives were used in small models at 1/24, 1/12, and 1/6 scales to simulate excursions in thermal reactors ^{down} to about a 2 millisecond period. Data gathered on concrete missiles indicated a threshold level of energy release below which no missiles would be expected. Above the threshold, the variation of number, size, and kinetic energy of concrete fragments was obtained as a function of simulated excursion energy as a basis for predicting the possible missile hazards to containment structures. The ability to extrapolate model data to larger systems was found uncertain because of a tendency of test results to diverge at different scales. Within the limitation of the probable missile characteristics from model data, the study was able to describe the ability of a given missile to penetrate steel shells of various thicknesses. Energy per unit diameter of steel and concrete missiles of various shapes required to

penetrate shells was expressed in a simple formula which includes effects of shell material, thickness, and shell bracing and support.

Illinois Institute of Technology Research Institute has studied the transmission and attenuation of high level loadings (such as air shocks) which could conceivably be imposed upon reactor vessels as a result of upper limit accidents such as violent excursions. A computer program has been developed to analyze the transmission of such shocks through generalized layered media. Application of this program was made to the design of blast shields constructed of materials which absorb energy by crushing. Tests were also made to determine the pressure vs. specific volume characteristics of the crushable materials under high pressure loadings in uniaxial compression, and the actual material characteristics were factored into the program as were the yielding characteristics of the structure being protected by the shield. Other IIT studies cover the analyses of gross plastic dilation of vessels under large dynamic and semi-static loading, defining the ultimate resistance of idealized vessels without penetrations and with various end conditions such as rigid or hemispherical caps. Various length-to-radius ratios and material characteristics were studied and families of curves of dilation vs. load were developed. In addition, the plastic stress and strain fields around simple idealized penetrations were studied so as to be able to predict the weakening effect of such penetrations during gross vessel dilation and the amount of reinforcing needed to restore the vessel strength to its unpenetrated level. Given the amount of gross deformation which is acceptable in terms of maintaining structural integrity and operating functions, the upper limit strength of realistic vessels could be assessed by application of the combined results of studies such as IIT, NOL, and Aberdeen. IIT is also conducting studies of the magnitude of upper limit loading that could be expected as a result of the maximum credible (loss-of coolant) accident in water cooled reactors. These studies involve the sudden one-dimensional and three-dimensional decompression of water under conditions simulating (at small scale) the gross rupture of primary vessels in pressurized and boiling water reactor systems. Air shocks and jet impact loads are determined as well as decompression phenomena and loading in the simulated primary systems.

Newer Studies of Actual Accident Phenomena, Engineering Safeguards and Improved Containment

The maximum credible accident currently postulated for water reactor systems is loss-of-coolant initiated by a gross pipe rupture in the primary system. Several studies are underway to assess the probability and extent of such a failure and the consequent rate of flow (blowdown) from the primary coolant system.

General Electric is studying the pipe rupture problem itself in order to determine the likelihood of failure and its extent under given sets of reactor operating conditions, as well as sets of operating conditions which can be relied upon to prevent gross failure. The

significance of crack initiation due to loading or to material imperfections is assessed and the likely nature of any crack propagation (such as brittle failure, ductile failure, and low energy shear failure) is determined for given conditions (such as temperature above or below the NDT and hydraulic or pneumatic loading). In addition, the significance of local stress state upon propagation of cracks is assessed and the history of past pipe failure is examined to determine the state of knowledge that would permit failure - free piping designs and better analysis of the postulated maximum accident. Based upon the current state of piping design and possible improvements, a program of pipe testing and analysis will be outlined.

It might be noted that the IIT research on decompression previously mentioned is partly devoted to examination of the decompression process as it might proceed in the neighborhood of cracks which have just started to propagate rapidly. Knowledge of pressure reductions and possible consequent reduction in stress levels around crack tips would be significant in reducing the possibility of the gross pipe rupture assumed in the maximum accident, particularly if the decompression front outruns the crack propagation.

The consequences of rupture in the primary system are being studied in part at the University of Minnesota in relatively small scale simulations of reactor primary vessels blowing down through orifices of various shapes and sizes. The existence of two-phase critical flow as a limiter of the loss-of-coolant rate is determined, and the pressure and flow regimes in the vicinity of orifices are measured so as to permit prediction of flows likely to occur through ruptures in systems at other scales. It has been found that actual flows from orifices are significantly less than those which would be predicted by use of conventional formulas. The significance of reduced flow rate in rate-sensitive systems, such as pressure suppression containment or emergency cooling systems, is obvious. The effect of various flow rates upon the pressure suppression concept has also been investigated by means of highly idealized model studies at the University of Minnesota, and the relative importance of system variables such as pool or dry-well capacity or vent line diameter has been assessed as an aid to optimized design of suppression systems.

Hanford Laboratories is beginning the investigation of blowdown phenomena at a large scale in the Containment Systems Experiment (CSE). Shown in Figures 2-10 and 2-11. The blowdown of a 150 cubic foot vessel by pipes up to 12" in diameter will be assessed to determine the flow rates generated and resultant pressures in containment systems (such as simple shells and vapor suppression pools). The blowdown process will be examined for the natural duration of resultant pressure, the required efficiencies and mode of operation of pressure suppression devices, and for the effect upon the transport of real and simulated fission products released in various time relations to blowdowns. The consequences of accidents, such as transport within and leakage of radioactivity from various containments, and the effectiveness of containment and other

CONTAINMENT SYSTEMS EXPERIMENT

II-28

Figure 2-10

CONTAINMENT SYSTEMS EXPERIMENT

ARRANGEMENT IN T-PLANT

Figure 2-11

engineering safeguards (such as sprays and filters) will be studied so as to define the actual course of accident phenomena. These studies would also include mechanical effects such as reaction loads upon pipe systems, jet impact on reactor components, as well as optimization of existing and new containment schemes to account for mechanical, thermodynamic and radioactive loads.

In the Nuclear Safety Pilot Plant (NSPP), Oak Ridge National Laboratory is studying the transport of wide ranges of aerosols and other chemical and physical forms of activity that might be produced in an actual meltdown so as to bracket the accident phenomena which might occur. The NSPP will not attempt to simulate all the characteristics of a real fuel meltdown since it uses non-nuclear heating. However, by varying the rate of melting, temperature of melt, and its surrounding atmosphere (including atmospheric type and impurities), and by using fuel of various types with various claddings and levels of burnup, it is expected that the types of activity produced will give results applicable to accidents in many types of reactors. The natural transport and plate-out of the activity will be assessed in a model containment shell, as well as the effectiveness of sprays and filters in reducing the mobility of fission products. To permit extrapolation of results to actual accidents in larger systems, extensive characterization of the ambient activity distribution in time and space will be made for various pressures, temperatures, air-steam atmospheres, and activity concentrations typical of real accidents.

A further example of studies to improve currently used containment systems is work done at Atomics International to measure leakage of conventional building components (such as metal or concrete panels). The leakage of individual components has been determined for the conditions in which they are installed according to normal practice and as they might be improved to reduce leakage (by use of appropriate sealers, paints, etc.). Leakage coefficients are determined for each component, together with the probable statistical error in the measured coefficients and the pressure ranges within which they are valid. The validity of combining the coefficients for individual leak paths to predict the total leakage of large containment buildings at various pressures has been investigated by means of small model buildings containing typical leaks. It has been found that (with proper design) one could probably build a large concrete building which would leak no more than 1% per day at up to 0.5 psi, and a metal panel building which would leak about 25% per day at up to one inch of water pressure.

To compile and assess the foregoing containment research (as well as pertinent material from other sources on design, construction, maintenance, and economics, etc.) and provide the nuclear industry with a summary of available data, ORNL is compiling a handbook of available material. The handbook is not intended to be a standard of design but to present a guide to whatever data and alternative analytical techniques can be brought to bear upon the problems of accident analysis, containment design, reactor siting, applicable regulations, and public safety. It is expected that a

first version of the handbook will be available in the fall of 1964, and that periodic revisions may be made as new material is developed by the various research programs.

F. Fast Reactor Safety

The major emphasis in fast reactor safety studies is on fuel meltdown investigations since fast reactors typically contain much more fuel than is required for a minimum critical mass. This characteristic, in conjunction with the short prompt neutron lifetime of fast reactors, can produce a severe accident if a meltdown is followed by rapid re-assembly into a more compact configuration. The Transient Reactor Test (TREAT) facility is being used for the experimental investigation of the complex factors influencing fuel failure and movement. The TREAT experiment, in which fast reactor fuel samples are subjected to transient nuclear heating, are conducted by Argonne National Laboratory and are supplemented by an intensive theoretical effort. Also included in the ANL program are supporting out-of-pile experiments and the development of special experimental and analytical tools.

TREAT experiments conducted with pre-irradiated EBR-II Mark I fuel specimens indicate that the amount of burnup (up to 1 atom percent) does not have a great effect on the threshold of failure, and that the meltdown residue of 1 atom percent burnup samples has more numerous and larger voids and about half the density of the unirradiated elements. Irradiated Enrico Fermi type fuel elements were found to fail somewhat sooner than was the case with unirradiated samples.

A number of experiments have been performed on gas-bonded, UO_2 -fueled, EBR-II sized elements in the absence of sodium. In these tests, stainless steel clad elements failed in a comparatively non-violent fashion near the melting point of the clad due to the internal pressure buildup of the inert gas. The high temperature failure of tantalum-clad elements (clad temperature of about 2400°C) was accompanied by extensive fuel movement and the production of small fragments.

In order to permit meltdowns under more complicated experimental conditions such as high temperature in the presence of sodium, the development of specialized equipment has been undertaken. Several package sodium loops, designed to hold 1 to 7 EBR-II size fuel samples and still fit into a 10 cm square by 240 cm long dummy TREAT element, have been fabricated and the initial experiments with flowing sodium have been performed. The large TREAT loop, a semi-permanent installation for meltdown studies on large clusters of elements in flowing sodium, is nearing completion. These loops will permit experiments which more closely simulate the conditions in a real reactor.

It is anticipated that the TREAT meltdown experiments will continue with increased emphasis on advanced fuels (e.g., oxides, carbides) and experiments in flowing sodium. Additional out-of-pile experiments to study liquid metal behavior will be initiated.

Analytical studies of fast reactor safety are carried out by United Nuclear Corporation. These studies include an independent review and evaluation of significant work in this and related areas, critical evaluation of analytical techniques, and recommendation of additional necessary work for initiation.

Figure 2-12

SPERT IV Facility

PART III

ENGINEERING FIELD TESTS

III. ENGINEERING FIELD TESTS

The programs in nuclear safety engineering field testing were initiated in 1961 to provide engineering field tests on terrestrial and aerospace nuclear systems to demonstrate their safety and reliability and to provide information required for the design of more economical systems without compromising safety. The terrestrial program is complementary to the reactor safety research and development experimental work described above. Laboratory and small scale safety tests initiated under research and development programs are continued and expanded into field prototype and full scale safety tests under conditions simulating those of actual service operation as closely as possible. The purpose of such field tests will be to:

- (1) Determine total system behavior under abnormal conditions with each of the parameters performing its normal role.
- (2) Determine the consequences of severe nuclear accidents and identify related phenomenology.
- (3) Define the characteristics of terrestrial and aerospace systems which are of importance in evaluating the nuclear safety hazards.
- (4) Obtain engineering data which would assist in designing engineering safeguards for incorporation in future systems.

The terrestrial test program has had no long historical development similar to the SPERT tests, but is a logical extension of earlier reactor safety programs. The basic planning for the aerospace test program was carried out by the Aerospace Nuclear Safety Board established in August 1959.

A. Terrestrial Systems

In the fall of 1961, the Phillips Petroleum Company was assigned responsibility as the prime contractor for performing safety tests of potential nuclear accident situations on an engineering scale. These tests are intended to extend the Nuclear Safety R&D information available from small scale and laboratory type tests into regions more realistic in size involving reactor decay heat, complete primary systems, and large containment volume. In many respects, these tests are similar to those initiated by the Federal Aviation Agency (FAA), in that they both involve complete systems. Because of the cost involved in using large scale power reactors, some compromises are necessary with respect to size. The predominance of water moderated power reactors in the anticipated nuclear power growth led AEC to first investigate accident tests on these systems.

The importance of obtaining accident consequence information in a continuing AEC-sponsored program may be indicated by the predictions of installed nuclear electric capacity used in the Report to the President (Civilian Nuclear Power) 1962^{1/}.

<u>Year</u>	<u>Nuclear Capacity (EMW)</u>
1965	1,350
1970	5,000
1975	16,000
1980	40,000
2000	734,000

It seems clear that water moderated converter systems will provide the bulk of the nuclear capacity through 1980. At that time, at least \$4 billion will have been invested in capital costs alone, assuming a minimum of \$100 per installed kw. A section called "Reactor Safety: Siting Problems" also received attention on page 66 of the Report to the President:

"Until experience is gained and adequate safeguards are proved out, prudence dictates that large reactor installations be fairly far removed from population centers. This adds both to transmission system costs and to expensive power losses in the lines. It also reduces the availability of sites, already low for large plants because of the need for ample supplies of cooling water."

In January 1963, Phillips issued a feasibility and conceptual design report^{2/} for a loss-of-fluid accident test (LOFT). This is the maximum credible accident most frequently postulated for water moderated systems. A rail-mounted dolly containing a 50 MWT reactor would be moved into a containment facility through a large door which is designed to be hermetically sealed. After the test is completed, the reactor will be moved out of the facility for examination in a large hot cell. By this method, the major part of the facility could be reused for subsequent tests on reactors of different design. The first dolly test would be performed with a pressurized water reactor system.

In June 1963, the Commission approved the selection of Kaiser Engineers as the architect-engineering design contractor, and of the Babcock and Wilcox Company as subcontractor for review of certain aspects of the Phillips reactor design. As of May 1964, Title I (preliminary design) was complete and Title II (detailed design) had begun. The Preliminary Safety Analysis Report was transmitted to the Division of Reactor Licensing and in August 1964, pre-construction review by the ACRS was completed. Construction of the LOFT facility was authorized in October 1964.

^{1/} Excerpts from Appendices to ... "A Report to the President - 1962" page 66, Table 16.

^{2/} "Feasibility and Conceptual Design for LOFT Facility", IDO-16833.

The following preliminary target dates indicate the current estimated schedule:

- (a) Fall 1964. Initiate construction. (This facility was approved as a capital budget item in the FY 1964 Congressional Appropriations.)
- (b) Fall 1967. Finish construction.
- (c) Fall 1968. Complete the non-nuclear operations.
- (d) Spring 1969. Complete the initial loss-of-coolant test.

During the non-nuclear test phase, the containment facility will be subjected to extensive tests never before performed on power reactor containment shells. In addition to the air pressure leak tests usually performed, repeated tests will be run in an air-steam environment created by rupturing the primary system piping at various pressure and temperature levels (blowdown) with various rupture sizes, and at both inlet and outlet rupture locations. During this phase, the pressurized water system will be artificially heated and will contain a dummy reactor core. From these tests, the pressure-temperature-time and leak-rate history will be developed to demonstrate the integrity of the containment facility. The structural integrity of the dummy core designed in accordance with current practice under the blowdown loads will provide guidance to industry. During this test phase, the design of special instrumentation will be validated.

The nuclear operations phase includes the usual start-up and power operations typical of pressurized water systems and additional operations unique to LOFT. After approximately 400 to 800 hours of operation at 50 thermal MW, the inventory of I^{131} (which has been established as potentially the most dangerous isotope to the general public in a nuclear accident situation) will have reached essentially an equilibrium value. When appropriate weather conditions are available at the NRTS, the primary system piping will be deliberately ruptured to permit expulsion of the coolant, uncovering of the nuclear core, meltdown of the fuel, and release of fission products to the primary system and containment facility volume. Over a period of about five days, data will be obtained on the transport of I^{131} (and other fission products) into the containment. Leakage to the atmosphere, if any, will be checked. The natural pressure decay history will be recorded and the effectiveness of spray-washdown of airborne fission products will be measured after the five day test. The effectiveness of decontaminating equipment and of industrial type cleanup filters under post-accident conditions will also be measured. During this test, and in the post-test examination, the intent is to measure the fraction of fission products trapped by the primary and containment system, and the fission product fraction escaping to the environment, both through irreducible leakage paths and perhaps through a controlled leakage device.

This information will provide the first comparison of measured release fraction to those currently postulated^{3/}. At the present time, it is accurate to say that nuclear power reactors are not being sited on the basis of demonstrated accident consequences.

The LOFT facility as designed will be capable of accommodating a series of dolly-mounted tests. The decontamination techniques to permit reuse are being formulated by Phillips. These techniques will minimize the dependence on remotely operated equipment and will utilize the experience of Phillips in direct (rather than remote) maintenance of the Chemical Processing Plant.

Although the philosophy of the first test is directed toward measuring the consequences of an accident with only the containment shell as a barrier to fission product release, other tests could be performed as follow-on experiments aimed at evaluating various engineered safeguards which have been proposed as accident limiting devices.

It is also probable that the LOFT facility can be used to accommodate tests of other thermal reactor systems including organic and gas-cooled should the need develop. It also appears that engineering scale tests should be performed on fast breeder systems. Although the time scale and accident mode have not yet been determined, it is possible that a facility such as the LOFT containment shell might be appropriate for such tests.

Schematics of the containment facility, dolly-mounted test assembly, and existing locomotive to move the dolly are shown in Figures III-1, III-2, and III-3, respectively.

B. Aerospace Systems

Engineering scale safety tests of aerospace nuclear systems have been accelerated and expanded to meet the needs of a variety of applications. Resolution of the technological problems involved in the design, construction, and operation of compact, high performance systems, which also incorporate adequate safety features under launch, controlled flight, and re-entry conditions, requires a major effort. During 1964, this effort was directed primarily toward the development programs of the Systems for Nuclear Auxiliary Power (SNAP). Emphasis was placed on the more imminent operational flights of each category -- the SNAP-10A for reactor systems and the SNAP-19 (the Pu²³⁸ fueled thermoelectric generator designed to supply electric power to the NIMBUS B satellite) for isotopic systems.

To test the safety design features of a generic system of the SNAP-19 type, a re-entry flight demonstration (RFD-2) was conducted on October 9, 1964, by the Sandia Corporation, the principal AEC contractor for aerospace nuclear safety. This was done in collaboration with the Martin Company, development contractor for the SNAP-19. Figure III-4 shows the Martin built payload used in the test.

^{3/} TID-14844 "Calculation of Distance Factors for Power and Test Reactor Sites" by J. J. DiNunno, et al, dated March 23, 1962.

Figure 3-1

LOFT - REACTOR EQUIPMENT LAYOUT

Figure 3-2

Figure 3-3

Figure 3-4
RFD-2 Re-entry System

The basic safety approach for the SNAP-19 is predicated on achieving an orbit of sufficient lifetime to allow the radioactive fuel to decay to non-hazardous levels before the generator re-enters the atmosphere. However, in the case where a less than nominal orbit is achieved, the generator is designed to (1) disassemble early in the re-entry regime and (2) expose the fuel capsules to a sufficient amount of the heat of re-entry to allow for (3) complete burn-up of the radioactive materials. It has been determined from theoretical analysis that, in the process, any hazardous materials would be ablated* into a fine particulate which would become suspended and dispersed safely in the upper atmosphere.

The basic objective of the RFD-2 flight test was to demonstrate with an inert system that the generator would disassemble and release its fuel capsules as designed. A second objective was to determine the burn-through response of the encapsulating materials used to contain the nuclear fuel (particulate formation is the subject of laboratory analysis described later). Specifically, RFD-2 was to:

1. Investigate the disassembly of an inert isotopic generator during re-entry.
2. Determine the time history of fuel capsule exposure to re-entry heating.
3. Study aerodynamic heat input to the generator during re-entry.
4. Correlate the results of the test with analytical predictions and with experimentally measured heating rates.
5. Study thermal-battery activation by the re-entry heat pulse. The purpose is to space check the operation of a device which can automatically initiate a satellite destruct charge during re-entry.

RFD-2, composed of the isotopic generator mounted on the front of a re-entry vehicle, was launched from the NASA Wallops Island Range on a Scout vehicle. The re-entry vehicle, which was designed to survive burn-up, contained the telemetering equipment used to radio back events as they occurred during the re-entry regime. The design included movement switches which would indicate generator disassembly during re-entry (thus releasing the inert fuel capsules). Preliminary data from the flight indicates that the generator disassembled as predicted. (The re-entry trajectory, event sequence, and stagnation heating rate pattern are shown in Figure III-5.)

Fuel capsules containing flare materials instead of Pu^{238} were included as part of the payload in order to determine the time required to burn through the fuel encapsulating material. This phase of the test was important because the plutonium cannot ablate until the encapsulating material burns off. Since the flare colors from the burning capsules were visible with the naked eye from Bermuda, we expect that the data (yet to be reduced) obtained with spectral cameras at Bermuda and on the downrange aircraft will give positive identification of burn-through altitudes and provide gross ablation rates.

* Ablation: Sublimation, vaporization, or melting of a surface material due to aerodynamic heating.

Figure 3-5

RFD-2 Re-Entry Trajectory

During one portion of the re-entry, it is impossible to transmit radio signals from the re-entry vehicle due to burn-up plasma which causes telemetry blackout. A tape recorder was therefore included in the re-entry vehicle for the purpose of recording events during blackout and playing back the information after blackout but prior to impact into the ocean. Playback was successful on RFD-2, probably due to the special antenna design. (Playback was not obtained after blackout on RFD-1.)

In the event playback had failed, it would have been advantageous to recover the re-entry vehicle so that the tape recorder could be retrieved. An actual examination could then also have been made of the re-entry vehicle. In order to achieve recovery, a flotation gear was incorporated into the re-entry vehicle along with a radio transmitter which would aid in locating it floating on the ocean. Although the parachute and its flotation gear and transmitter were recovered, the re-entry vehicle was never found. However, since playback was achieved, no significant test data were lost.

Although it appears that the flight was highly successful, based upon the excellent results described above, it will take several months to analyze the hundreds of photographs that were taken and to translate the telemetry tapes into actual temperatures and events that occurred on the generator during re-entry burn-up.

A similar re-entry flight demonstration (RFD-1) was carried out in May 1963 on a prototype of the SNAP-10A reactor system (see p. 144 of the 1963 AEC Annual Report). Figure III-6 shows the RFD-1 re-entry vehicle with mounted prototype reactors. It was determined from that test that an aerospace reactor could be designed to disassemble as a first step in the re-entry burn-up process associated with the ultimate disposal mechanism of any residual radioactive fission products. Ablation rates of clad uranium-zirconium hydride fuel elements were demonstrated by the flare technique described above. The predicted and actual events as they occurred during the re-entry flight are shown in Figure III-7.

The emphasis during 1964 on the SNAP-10A reactor system also included a test to determine the consequences of an accidental nuclear excursion. Such an accident could be induced by the inadvertent water immersion of a SNAP-10A reactor. This could take place if a 10A mission should abort on launch in such a way that the reactor were thrown into the waters off-shore of the range or into the deluge water flume or catch basin. In the experiment, designated SNAPTRAN-3, the reactor, surrounded by a boron sleeve, was immersed in a tank of water. The boron sleeve was propelled up from around the reactor core very rapidly by a squib-initiated black powder charge, thus providing for the insertion of the maximum amount of reactivity in one large step. The resulting energy release -- determined to be between 35 to 45 megawatt-seconds -- was essentially as predicted by analytical techniques. The amount of radioactivity resulting from the fission product cloud and the debris from the test itself was found to be

Figure 3-6
RFD-1 Re-Entry Vehicle

Figure 3-7

RFD-1 TRAJECTORY

very low. Figures III-8, III-9, and III-10 show the reactor before and after the test. The test was conducted by the Phillips Petroleum Company in conjunction with Atomics International at the National Reactor Test Station in Idaho.

Another test, called SNAPTRAN-1, is scheduled for completion in June 1965. The purpose is to examine the kinetic behavior of a SNAP-10A reactor when it is subjected to varying reactivity insertions with the control drums. A nuclear excursion destruct is planned for the final test of the series.

While flight tests have provided relatively gross data on the ablation rates for the encapsulating materials used in SNAP isotopic systems and for the fuel elements used in SNAP reactors, it is necessary to obtain data of a more fundamental nature in order to understand completely the re-entry burn-up phenomenon itself. This greater understanding is a prerequisite to determining the particle size and attendant dispersion pattern of the re-entry debris, an important consideration in identifying the biological hazard associated with radioactive material in the atmosphere. To achieve this objective, the Sandia Corporation is conducting theoretical and experimental studies of the burn-up (ablation phenomena) associated with high-speed re-entry of a body into the earth's atmosphere. Tests have been undertaken on oxidation rates of specific reactor fuels as a function of their shearing stresses. For example, the heat input was supplied to small wire specimens by electrical resistance heating. Flow conditions were simulated by extending the wires across a small wind tunnel. Test results confirm that the oxidation rate of zirconium, for example, varies as a function of the aerodynamic shearing force on the body. A separate theoretical study of thermal effects of metallic oxidation is continuing to establish the possible contribution of oxidation to the ablative destruction of SNAP reactor and isotopic fuel materials. Further, wave superheater tests have been run on various combinations of materials such as zirconium, zirconium hydride, stainless steel, and molybdenum as a part of a study of the reaction of metals in a simulated re-entry environment. Observations of oxidation and ablation rates associated with the destruction of zirconium hydride revealed that rapid ablation results although the heat added by convection alone is insufficient to raise the surface to the melting point. Also, the low values of effective heats of ablation in nitrogen strongly suggest that Zr and ZrH_x react rapidly with the nitrogen of the air under test conditions.

A series of plasma tunnel tests have been conducted to study the oxidation characteristics of tantalum under high-enthalpy, low-pressure flow simulating orbital decay re-entry conditions. No tantalum specimen could be made to melt, decompose, or lose weight in pure nitrogen flow; other specimens, when exposed to the same enthalpy and pressure condition as in the nitrogen tests but with a nitrogen-oxygen mixture, did attain a melting condition. The length of time to attain surface melting increased with decreasing oxygen flow rate. The surface temperature of the specimens, at melt, was observed to be as much as 1000°C below the melting temperature

Figure 3-8

SNAPTRAN-3 Core Before Test

Figure 3-9
SNAPTRAN-3 Core After Test

A

B

C

D

Figure 3-10

SNAPTRAN-3 Core Vessel
Expansion During Excursion

of pure tantalum. Oxidation rates, based on measured length changes in these tests, appear to be faster than the predicted length changes for given oxygen flow rates. These preliminary studies have provided valuable insight toward understanding the thermo-chemical behavior of re-entering tantalum. Similar work is now underway to resolve discrepancies between theory and experiment.

To provide additional information on re-entry burn-up, a novel flash heating technique has been developed for studying the combustion of freely falling droplets of metal. This method involves melting and igniting carefully cut squares of metal foil with an intense pulse of heat from a capacitor discharge lamp. With this device, droplets of zirconium and plutonium have been burned in air containing various amounts of water vapor and in a gas consisting of 20% oxygen-80% argon mixture. A curious explosion phenomena was observed (Figure III-11). These studies are expected to result in an understanding of the mechanisms responsible for the explosive behavior of molten droplets. This is important to aerospace safety because it is postulated that molten droplets will be formed by the intense re-entry heat pulse. These droplets could explode and contribute to formation of burn-up debris. We must ultimately know the debris size and character.

The importance of re-entry burn-up to aerospace nuclear safety has prompted an investigation of the high-temperature thermal, physical, and chemical properties of materials used in SNAP systems involving Sandia Corporation, the Oak Ridge National Laboratory, and Atomics International. The Oak Ridge National Laboratory is producing a data book on the physical and chemical properties of SNAP encapsulating and fuel materials. A tabulation was made of all known properties of isotopic and reactor fuels as reported by fuel production agencies.

In addition to our need to understand re-entry burn-up phenomena (and we also are investigating other means of ultimate disposal), it is necessary to examine pre-flight incidents on both reactors and isotopic systems. For example, prior to orbital flight of a SNAP system it is, of course, possible for accidents to occur due to vehicle aborts at the launch pad in which the entire system is destroyed by fire or explosion. For this reason, tests have been conducted to determine the behavior of a SNAP system in environments simulating a launch abort fire and/or explosion (Figure III-12). In addition, the nuclear device may, in a launch pad abort, fall from the top of the vehicle to the launch pad; or in case of an improper launch, the SNAP system might impact on land or in the ocean immediately off the coast. For this reason, the various systems have been subjected to impact tests at selected velocities simulating land and ocean impacts in order to determine the destruction mode and related information (Figures III-13, III-14, and III-15).

Figure 3-11

Time Exposure of Exploding Droplets

Figure 3-12

Explosive Ground Environmental Testing

Figure 3-13

Earth and Water Ground Impact Experiments

Figure 3-14

Nose-on 100-Foot Drop Test of SNAP Reactor

Figure 3-15

SNAP-9A Fuel Capsule Impact Test

Nuclear Safing and Recovery

A survey has been initiated of approximately thirty AEC reactors, typical of their class, to assess the emergency equipment which would be required in an effort to effect recovery operations in the event of a major nuclear reactor accident. This assessment will be used to determine if the AEC's capability to recover from such an accident can be enhanced by an equipment improvement program or if the development of unique equipment is required. Operations to be considered will be the deactivation of the reactor to preclude a follow-on excursion, implementing the retrieval of evidence relating to the cause of the accident, reducing the radiation hazard to the general public, and expediting the return of the facility to an operational status. The basic objective of this program is (a) to determine if special or unique nuclear safing and recovery equipment is needed, and if so, (b) to develop it.

PART IV

EFFLUENT CONTROL RESEARCH AND DEVELOPMENT

IV. STATUS SUMMARY - EFFLUENT CONTROL RESEARCH AND DEVELOPMENT

The programs in Effluent Control Research and Development are directed toward the safe management and disposal of various types of radioactive wastes, and the development of criteria associated with the environmental aspects of nuclear technology operations. Through these activities, it is sought to provide a basis for defining and controlling the ultimate fate and possible effects of radioactivity in the environment. The work is carried out at Commission laboratories and installations, in cooperation with other Federal agencies, and through contracts with research institutions.

A. Environmental Studies

Programs in this area are related to the controlled discharge and disposal of radioactive solid, liquid, and gaseous wastes to nature. Detailed evaluations of specific environments are made to establish rational technical criteria for the safe dispersal of radioactive effluents. Investigations are made to determine the ultimate fate of specific isotopes in water and land environments and the safe capacity of these environments for radioactivity. The Commission's operations at Hanford, Savannah River, the National Reactor Testing Station, Oak Ridge, and other installations constitute a large part of AEC's environmental research and development effort. The U. S. Geological Survey undertakes studies on hydrogeological aspects of waste disposal, and the U. S. Weather Bureau conducts meteorological studies concerned with the fate and behavior of radioactive materials in the atmosphere. This work is supplemented by specific studies by scientists at universities and research institutions. The estuary and marine studies, stream and river studies, and soil and earth studies included in the environmental research and development are summarized in the following paragraphs.

Estuary and Marine Studies

1. Estuary Studies. Field studies of physical dispersion of radioactive effluents in estuaries have been conducted for the AEC by the Chesapeake Bay Institute (CBI) of the Johns Hopkins University. A comprehensive study of New York Harbor, involving field measurements of currents, temperature, and salinity by the U. S. Coast and Geodetic Survey and data analyses by CBI has provided a means for evaluating the safety of nuclear ship operations within the harbor. The development of a Rhodamine-B fluorescent technique by CBI, which has a field sensitivity of 0.2 parts per billion, has constituted a major technical breakthrough in that direct measurements can now be made of the dispersion of a controlled-introduced containment in natural water bodies, on a significant time and space scale, at reasonable material cost.

2. Coastal Studies. Extensive environmental studies have been carried out in the Atlantic and Pacific Coast sea disposal areas to determine if the discharge of solid packaged low activity waste was causing any adverse effect on the oceanic environment. Seasonal surveys

have included the collection of plankton, bottom sediments, fish and seawater samples to the thousand fathom depth. Based on the results of alpha, beta and gamma low level counting analyses, it was determined that no radioactivity existed in bottom sediment, benthic organisms and bottom fish that exceeded natural background.

Investigations of oceanographic problems related to the safety of operations under the aerospace nuclear program are being conducted by the Chesapeake Bay Institute, in the Cape Canaveral (AMR) area, and by Scripps Institution of Oceanography in the Point Arguello (PMR) area. The objective of this program is to define any environmental hazards associated with these operations by comparing the possible environmental releases and subsequent fate of radioactivity with permissible concentration limits in the marine environment.

3. Water Circulation. Measurements on the rate and extent of total (vertical and horizontal) circulation in a 200-mile wide band of Atlantic coastal waters are being carried out by the Woods Hole Oceanographic Institute. Studies are also being carried out on coastal waters in the vicinity of missile launch sites in connection with aerospace nuclear safety.

Stream and River Studies

1. Clinch and Tennessee Rivers. Although downstream users experienced no problems, a program was initiated in 1961 to determine if radionuclides released to the Clinch River would concentrate in the bottom sediments or in the biological activity and possibly constitute a subsequent potential hazard. Objectives are to determine the ultimate fate of radioactive materials discharged to the system, the total diluent capacity of the system for low-level waste disposal, and the long-term effects of such use. A comprehensive cooperative investigation of the White Oak Creek - Clinch River - Tennessee River system, used by ORNL for the disposal of low-level radioactive liquid waste, is being carried out. Participants include ORNL, TVA, USGS, USPHS, AEC, the Tennessee State Department of Public Health, the Tennessee State Game and Fish Commission, and the Tennessee Stream Pollution Control Board. Chemical treatment in the ORNL plant generally provides sufficient decontamination of the primary offending radionuclides (Sr^{90} and Cs^{137}), thereby reducing these isotope concentrations to a level at which further dilution by the receiving streams produces final concentrations well below the maximum permissible levels. A comprehensive water and suspended and bottom sediment sampling program, established to measure the radioactive material moving through the river system, has indicated that 90-95% of the Sr^{90} , Ru^{106} , and Co^{60} are in the water phase of the sample, while over 80% of the Cs^{137} is associated with suspended sediment and approximately 5% with the bottom sediments. Dye and radioactive tracer tests have been run to determine the time of travel of radioisotopes and to measure the vertical, horizontal, and longitudinal dispersion in the Clinch River. This study also includes determination of the effect of fish movements and other factors (e.g., operation of Melton Hill Dam) on the distribution and transport of radioactivity in the Clinch River system.

2. Studies at Other AEC Sites.

a. The U. S. Public Health Service has conducted studies in various streams to determine the extent to which physical, chemical, and biological factors in a stream tend to remove radionuclides; to determine the effect of these factors on subsequent water treatment processes; and to develop parameters applicable to the disposal of radioactive wastes into streams. The results of completed stream studies (e.g., KAPL effluent into Mohawk River, Shippingport effluent into Ohio River) indicated radioactive isotope concentrations too low for quantitative determination. Preliminary results from observations on the Lower Three Runs River indicate that all levels of radioactivity arising from the discharge of effluent from the Savannah River Plant are well below the prescribed tolerances.

b. The USGS, in cooperation with General Electric-Hanford, Oregon and Washington State agencies, and the University of Washington, has initiated a study in the Columbia River and its estuary to determine the ultimate fate of radionuclides released from the nuclear operations at Hanford in the water-borne phase and those associated with the bottom sediment.

c. The University of Texas has installed a hydraulic facility in order to establish a model ecosystem to determine the relative effect of parametric variables on the fate of radionuclides.

d. Harvard University is developing a mathematical model of the ultimate fate of radioactivity in streams which will also be utilized to determine the relative effects of the many variables on the fate of the activity. Data from various river studies will be utilized to verify the model.

Soil and Earth Studies

1. Geohydrologic Studies. The entire NRTS site in Idaho is covered by a varying layer of alluvial and is underlain by a succession of basaltic lava flows which form layers of dense hard rock varying in thickness, physical characteristics and areal distribution, and interbedded with sediment, cinders and flow breccia. Approximately 3-4 billion gallons of low level radioactive effluents from the ETR-MTR test reactor complex have been discharged to the ground via a leaching pond since 1951, and approximately three billion gallons and 1000 curies (beta-gamma) have been discharged from the CPP via an injection well which penetrates the regional water table 160 feet to a total depth of approximately 600 feet. Drinking water for the entire station is obtained from other wells which draw on an aquifer distant from but penetrated by the injection well. Therefore, it is essential that the movement of any hazardous radionuclides within this domestic water supply be monitored and predictable. Area and regional studies of the ground water are underway including water level contour mapping, chemical quality studies, stratigraphic studies, and borehole geophysical surveys of all existing wells. More than twenty wells have been drilled in the vicinity of the injection well into the

basalt formation to depths of 150 feet below the water table, to trace the movement of waste discharged to the injection well. Through movement of the waste water has been traced, by the use of the inherent tritium, to the Central Facilities Area and the OMRE water supply well, $5\frac{1}{2}$ and $3\frac{1}{2}$ miles from the injection well, respectively, the gross concentration of radioisotopes beyond 1400 feet from the disposal well was less than 10^{-7} $\mu\text{c/ml}$, the detection limit of routine analyses at NRTS.

A similar number of wells that do not penetrate the basalt were drilled around the leaching pond serving the MTR-ETR to follow the flow patterns of both water and radioactivity produced by that disposal operation. The investigations have shown a characteristic perched water table formed by percolating waste water perched on the dense basalt layer which prevents penetration into other aquifers in the basalt. The movement of the perched aquifer in the vicinity of the leaching pond is being monitored and parameters governing its movement are being determined. Current results indicate a lateral movement of water only 2000-2500 feet in 9 years of operation, with no detection of strontium or cesium activity above normal background beyond 400-500 feet from the pond.

B. Low and Intermediate Level Waste Management

These programs involve the development, testing, and application of improved systems for handling, treatment, and disposal of low and intermediate level radioactive waste materials. Classification of radioactive effluents for this purpose is based on the decontamination factor (DF) required to yield final radioisotope concentrations within the prescribed maximum permissible values for public consumption. A low level effluent is one requiring a DF in the range from 10 to 10^3 , and an intermediate level effluent is one requiring a DF in the range from 10^3 to 10^5 . This classification is arbitrary, but there is no universally accepted scheme. In general, low and intermediate activity wastes pose fewer problems than high activity wastes since it is normally possible to achieve sufficient decontamination by relatively simple chemical treatment methods in the former case to permit discharge to the environment. Volume reduction is also important since the largest volumes of waste generated in the various U. S. atomic energy activities (e.g., uranium mining and milling, feed materials preparation, fuel fabrication, etc.) are low or intermediate level effluents.

1. Ion Exchange Removal. ORNL is investigating decontamination processes using both organic and inorganic ion exchange materials. A pilot plant has been constructed and extensively operated to demonstrate the feasibility of using various resins and combinations of head-end treatment for the removal of strontium and cesium radionuclides. Laboratory experiments indicate consistent DF's of 1000 or better for these isotopes, and a volume reduction of about 2000. Studies conducted to determine the feasibility of utilizing a continuous ion exchanger for optimizing the removal of Cs and Sr indicated better results than the fixed bed. Hanford is currently utilizing ion exchange material for the

removal of Sr from the high salt, high level waste supernate from decladding tanks. This resin used in series with a mineral exchange column for Cs removal provides excellent decontamination and volume reduction factors.

2. Clay Mineral Exchange. Earth materials illustrate different degrees of affinity for various radionuclides. Hanford has demonstrated the operational feasibility of both "calcite" and clinoptilolite for the removal of strontium and cesium from process waste streams. ORNL has illustrated the "fixation" of cesium within the crystal lattice of illitic type clays and the reaction of strontium with bauxite. Local natural materials in the vicinity of the NRTS are being studied to determine the utilization of such materials for treatment of wastes from the ATR-MTR-ETR complex. It has also been demonstrated at all Commission sites that the various clay fractions in the earth sorb large percentages of radionuclides present in the liquid wastes discharged to the ground which come into contact with the clays. Basic mineral exchange studies are being conducted at the University of North Carolina, Hanford, and ORNL to develop the various parameters of mineral exchange kinetics which affect the radioisotope sorption characteristics of the minerals.

Waste Disposal

1. Hydraulic Fracturing of Shale. ORNL has been investigating the feasibility of discharging intermediate activity waste into the ground by hydraulic fracturing of shale formations. This work includes the general determination of the type of fracture pattern developed by the hydrofracturing technique in Conasauga shale, the most suitable geological formation in the ORNL area. Fracture patterns parallel to the bedding planes of these formations have been developed at depths from 300-1000 feet in three separate hydrofracking experiments conducted at ORNL. In each test, a grout mixture of diatomaceous earth, cement, and water containing Cs¹³⁷ and Co⁶⁰ as tracers was injected under high pressure into the formation to produce the fracture. The pressure was maintained on the injection well until the grout mixture set. Coring indicated that the grout sheets conformed to a wafer-like pattern and followed the bedding plane showing no tendency to escape the shale by cutting across these planes toward the surface. A hydraulic fracturing plant has been constructed at ORNL and a full-scale engineering demonstration with actual intermediate level waste will be conducted during 1964-1965. A sectional view of the pilot plant is shown in Figure 4-1.

2. Injection in Deep Permeable Formations. The possibility of pumping large volumes of low-level radioactive waste into deep, isolated porous formations offers some promise, in suitable areas, as an alternative to surface discharge. Problems involved in safe industrial waste disposal through deep wells depend very strongly on the geology of the particular formation and on the physical and chemical characteristics of the waste. The thickness, porosity, permeability, and pore size of the geological formation determine its volumetric capacity for liquid waste disposal, and the overlying cover plus the regional hydrology determine

Figure 4-1

its liquid retention capability. Mineral exchange reactions, for attenuation of radionuclide movement, depend upon the mineral types in the geologic formation and the chemical quality of the waste. Required waste pretreatment to permit pumping into the well without plugging depends on waste composition and disposal formation characteristics. A cooperative program is being carried out by several organizations. Laboratory investigations and theoretical studies are being carried out by ORNL, the U. S. Bureau of Mines, the U. S. Geological Survey, and the University of California. Full-scale field investigations will depend on the results of these studies.

In related studies, the U. S. Geological Survey has compiled stratigraphic data on known deep sedimentary basins, and a subcommittee of the American Association of Petroleum Geologists is providing detailed geologic and hydrologic information on these basins. Permeability studies are being carried out by the Bureau of Mines by injecting simulated waste solutions into drill core samples from various geological formations. A theory has been developed at the University of California to permit prediction of the arrival of radioisotopes at any point in the geologic formation following the start of injection operations. The arrival time of a specific radioisotope is estimated by determining the distribution coefficient (K_d) for the isotope in an in-situ core sample and relating this to the rate of water movement and dispersion through the geologic formation determined by using tritium. Calculated and actual dispersion curves from a shallow injection field test using tritium and strontium tracers showed good agreement. A field study at least one order of magnitude larger in scale is planned to verify the concepts developed in the initial field tests.

C. . High Level Waste Management

Problems associated with the treatment and disposal of high level radioactive wastes receive a major part of the over-all effluent control research and development effort. The chemical processing of irradiated reactor fuels to separate plutonium from uranium and the various fission products provides the largest volume of these wastes. Long-term storage has been relied upon at the three generating sites (Savannah River, Hanford, and Idaho) for the discharge of between 65 and 70 million gallons of high level wastes. In general, tanks of carbon steel or stainless steel equipped with decay heat removal systems are employed. At Hanford and Savannah River, high-level liquid wastes are also neutralized with caustic prior to storage. The storage equipment and procedures employed at Idaho (NRTS) are somewhat different due to the nature of the fuel materials processed (enriched U) and various dissolved materials (stainless steel, aluminum, zirconium) in the waste.

Alternatives to long-term tank storage are being investigated because of the problems involved including potential leakage and the need for transferring the contents to new tanks at intervals over a total period of several hundred years due to the limited useful life of any given tank.

A vigorous research and development program is in progress to develop and demonstrate, on an engineering scale, systems for conversion of high level liquid wastes to solids suitable for permanent disposal in a satisfactory environment. Several "conversion-to-solids" methods are under study including the use of fluidized beds, radiant-heated spray columns, and heated pots. The addition of glass-forming materials to the waste for the purpose of providing a relatively nonleachable final product is now receiving intensive study.

Waste Treatment

1. Fluidized Bed Calcination. The largest development effort in this area has been the installation of a 60-gallon per hour hot pilot plant facility at the National Reactor Testing Station in Idaho for the fluidized bed calcining of acid aluminum nitrate waste from MTR-type fuels at temperatures from 400-550°C. Cold engineering development work and successful Na²⁴ tracer runs have been completed in 1962 and early 1963. Following certain plant modifications, hot operation was initiated on December 8, 1963, with a feed stream of five year old waste stored in tanks at NRTS. By the end of August 1964, approximately 445,000 gallons of waste had been calcined, with a volume reduction of about 10 being obtained.

2. Radiant Heat Spray Calcination. A second promising method for calcining waste is the radiant heat spray calcination process which is being investigated at the Hanford Laboratories. Wastes are fed into the top of a heated column (850°C) through a pneumatic nozzle. Water vapor passes down the column through successive zones of evaporation, drying and calcination. The solid product from the column may be stored as a fine powder or may be converted to a glass product by fusing with certain added materials. Several successful hot cell runs with full-scale Purex waste have been completed during 1962-63. The design, development, and fabrication of a Waste Solidification Engineering Prototype Plant for installation in the Fuels Recycle Pilot Plant is now underway. A flow sheet for the prototype is shown in Figure 4-2, and a mockup of one of the racks which will be used for the plant is shown in Figure 4-3. Present plans call for this plant to be operational in early 1965, and the radiant heat spray calcining process is scheduled for hot demonstration during late 1965-66.

3. Pot Calcination. The Oak Ridge National Laboratory has developed a "conversion-to-solids" method known as the pot calcination process. In this system, liquid wastes are evaporated to dryness and the solids then calcined in a portable pot which serves as the final storage container. Extensive laboratory and cold engineering operation have been conducted at ORNL to obtain information required for the design of the hot pilot plant facilities to be installed in the Fuels Recycle Plant at Hanford. Cold engineering studies are continuing in 1963-64 with emphasis on engineering problems associated with the production of glassy solids. Demonstration of the pot calcination process with full-scale high activity waste is planned during FY 1965-66 in the Fuels Recycle Pilot Plant (FRPP) at Hanford.

WASTE SOLIDIFICATION ENGINEERING PROTOTYPES

Figure 4-2

Figure 4-2

WASTE SOLIDIFICATION ENGINEERING PROTOTYPE REMOTEABLE, PROCESS RACK-2A

IV - 10

Figure 4-3

4. Conversion to Glasses. During the past three years, the Brookhaven National Laboratory has been studying the incorporation of high activity waste into phosphate glasses. Glassy solids have the advantage of being relatively nonleachable in comparison with calcined oxides, and also appear to have better heat transfer characteristics, thereby permitting the use of larger disposal containers and less ultimate storage space. The operation of cold pilot plant at Brookhaven to investigate the feasibility of phosphate glass fixation with respect to larger scale operation problems, with particular emphasis on continuous operation, will be carried out during 1963-64. Concurrently, Hanford will perform several hot cell runs with full level waste to demonstrate the feasibility of the BNL continuous molten glass process. If results of this work are successful, it may be possible to have a glass process ready for engineering demonstration in the FRPP on the same time schedule as the pot and radiant spray calcination processes. At the present time, all the above processes show specific promise for engineering application to various reprocessing wastes by 1968.

Waste Disposal

1. Disposal in Salt. After high level liquid wastes are converted to solids, there still exists the requirement for storage or ultimate disposal of these solid wastes. This has led to the investigation of selected geologic formations for this purpose. Salt has been chosen as the most optimum disposal media because of its unique geologic characteristics. Salt formations are dry, impervious to water, and not associated with usable groundwater sources. Because of its plasticity, fractures in salt seal or close rapidly. Deposits of rock salt underlie some 400,000 square miles of the United States and may represent some of the few naturally occurring dry environments in the Eastern part of the country. Extensive laboratory investigations at ORNL and field studies in the Carey Salt Mine, Hutchinson, Kansas, have shown strong promise. A field experiment is now being designed using ETR fuel elements, to simulate the thermal and radiation characteristics of full-scale activity waste which would exist in a pot containing calcined solids. This field demonstration is planned for early 1965. Figure 4-4 is a cutaway drawing of the experimental arrangement in the mine.

2. Disposal in Deep Impermeable Formations. As an alternative to high level liquid waste tank storage, the AEC is investigating at its Savannah River Plant the feasibility of storing long-cooled, non-boiling fuel reprocessing waste in deep impermeable (basement rock) formations, approximately 2000 feet beneath the plant. Several widely spaced exploratory holes have been drilled into the underlying bedrock for detailed geologic correlation. Extensive field hydrologic tests of the basement rock have been performed and continuous core samples have been obtained for correlation of physical and chemical compatibility characteristics of the rock with Savannah River Plant waste. Technical studies have been conducted with respect to waste characteristics, including heat generation rates, age of waste to be stored, and the physical form of waste considered most desirable. An economic comparison

DEMONSTRATION OF RADIOACTIVE SOLIDS DISPOSAL IN SALT

Figure 4-4

with perpetual tank storage indicated this concept could produce lower costs. A preliminary safety analysis has been completed, and indicates that no off-site hazards would be involved in this method.

Engineering Evaluation

ORNL is conducting an engineering evaluation of high-level liquid and solid radioactive waste treatment and disposal systems. The economics and hazards of various combinations including interim tank storage, removal of specific fission products, calcination, interim solids storage, transportation, and ultimate disposal in salt formations, are being studied.

D. Gaseous and Aerosol Effluent Studies

Investigations of treatment and disposal methods and equipment for the control of radioactive gaseous and aerosol effluents are directed toward the development of new systems and the improvement of existing systems. The radioactivity of these wastes may vary by several orders of magnitude from tracer amounts to thousands of curies. Treatment may also be complicated by the presence of airborne particulate matter, sprays and aerosols, and by the evolution of volatile compounds of polonium, ruthenium, and other elements. Sources of airborne activity include off-gases from dissolvers in fuel reprocessing plants, spray from waste-storage tanks, ventilation air from processing plants and research laboratories, coolant gases from gas cooled reactors, flue gases from incinerators, and off-gases from solids fixation processes. Principal areas of effort in gaseous and aerosol effluent studies are decontamination equipment research, development, and testing; and meteorological investigations of the fate of radionuclides discharged into the atmosphere.

An extensive research and development program is being carried out at the Harvard Air Cleaning Laboratory on air and gas cleaning with particular emphasis on the achievement of high efficiencies for submicron particle removal at high temperatures. Research and development is also being carried out on methods of removing radioactive iodine from reactor and chemical processing off-gas systems, and on the application of foam techniques for the suppression and removal of gaseous and particulate effluents in any possible accidental releases. Additional projects include the development of performance evaluation methods for air cleaning systems, and evaluation of air cleaning operations throughout AEC installations. A commercial prototype incinerator built for field use has recently been completed. Decontamination factors and submicron particle removal efficiencies requiring multistage cleaning systems in which the last stage is a high efficiency filter with earlier roughing stages will be evaluated. The Harvard Laboratory has developed an electrostatic precipitator capable of removing most of the tarry particulates which cause rapid plugging and deterioration of filter surfaces. This device can be operated at temperatures as high as 700°F with an average removal efficiency greater than 95 percent.

The U. S. Weather Bureau conducts and supervises through its Environmental Meteorological Research Project investigations of the atmospheric transport and diffusion (dilution) of gaseous effluents. Short range diffusion and deposition studies were carried out at the National Reactor Testing Station, in which the release of uranine dye tracer was used to measure the effects of wind fluctuation on dispersion as compared with estimates based solely on wind velocities. Radar tracking tests of balloons (tetroons) to determine air trajectories and turbulence provide information on complicated wind patterns and wind trajectories for specific effluent releases or radioactive clouds in an emergency (e.g., SL-1 incident). Further studies of the feasibility of longer range (50 mile) tracer runs are being carried out. The Idaho Operations Office is providing general support for the NRTS meteorological studies (e.g., computer service, instrument development and maintenance, etc.). A revision of the handbook "Meteorology and Atomic Energy" is being prepared for publication. As part of a staff function to assist ACRS in evaluating the meteorological aspects of reactor hazard analyses, a diffusion climatology of the Los Angeles Basin was prepared. Also, a wind tunnel study at New York University simulating the wind flow and diffusion around typical reactor buildings, initiated in FY 1962, has been completed.

Nuclear Science and Engineering Corporation is studying the feasibility of utilizing I^{129} as a long range tracer. The I^{129} is reacted with a fluorinated carbon to form CF_3I which is released as a gas. The compound is sorbed onto a solid adsorbent, removed from the sorbent, activated in a reactor, and analyzed for I^{130} . Following completion of the laboratory phase early in calendar year 1964, a field test to determine its application will be conducted at NRTS.

To evaluate the feasibility of using the unique lithologic environment at the NRTS for disposal of large quantities of gaseous effluents, a series of field tests were initiated in FY 1964. Traced air will be pumped into specific geologic formations. The area will be monitored to determine the integrity of the formation, and various studies will be conducted to determine the relative effects of the numerous parameters involved in this method of disposal at the NRTS site.

The University of California has also initiated laboratory studies to develop the various types of geologic formations which can accept and treat radioactive off-gases associated with power reactors and to determine the scope of studies necessary to define the phenomena associated with gaseous disposal in each type of formation.

PART V

ANALYSIS AND EVALUATION

V. ANALYSIS AND EVALUATION

Activities in the area of nuclear safety analysis and evaluation are of recent origin and include responsibilities other than the technical direction of projects. The primary objectives of this work are to provide internal coordination among the various nuclear safety programs under DRD direction and external coordination with other AEC Headquarters organizations having safety responsibilities (e.g., Division of Reactor Licensing, Division of Operational Safety); to develop and apply knowledge, experience and techniques for nuclear safety analysis and evaluation; and to insure the continued improvement of nuclear system design, construction and operation from a safety viewpoint through the dissemination of information gained in the over-all nuclear safety program to governmental and industrial groups engaged in the nuclear energy field. Specialized expert consultation is also provided in specific problem areas of nuclear safety (e.g., pressure vessels and other primary system components, reactor kinetics and control, etc.) and independent safety assessments of reactor systems are made as required. The following discussion is concerned primarily with the technical direction of nuclear safety analysis and evaluation projects. These studies are generated by, and contribute to, the objectives and needs of the over-all Nuclear Safety Program.

A limited program of analytical and survey studies in nuclear safety is being carried out. Studies currently underway include a survey of power reactor operating and safety experience, and an investigation of the human factors affecting the performance of reactor operators. An additional study has been initiated which involves the application of probabilistic methods in reactor safety analysis. Further expansion of the study program to include other relevant technological areas (e.g., control and safety system reliability) is being carried out as required. Analytical and experimental investigations are also being conducted to establish more adequate specifications (including inspection and quality control techniques) for reactor vessels, piping, and other primary system components. A Nuclear Safety Information Center was established at the Oak Ridge National Laboratory in 1963 as a focal point for the collection, evaluation, and dissemination of information in major areas of nuclear safety technology.

Surveys of operating and safety experience have been carried out for AEC by Holmes and Narver, Incorporated, on test reactors and on research reactors. The results of these reactor surveys have been published by the AEC in HN-172 "A Study of Test Reactor Operating and Safety Experience" dated May 10, 1963, and HN-180 "A Study of Research Reactor Operating and Safety Experience" dated June 12, 1964. These surveys developed technical information on actual operating experience at both AEC and industry test reactors, and at ten selected research reactors, with emphasis on the nuclear safety aspects of reactor operation. Attention was also given to administrative organization, procedures, and controls in relation to safety.

Marked differences were found among the test reactor facilities in the specific role of the operator, varying from that of a mere watch tender to a participant in operational planning and execution. Wide variation was also found in the modus operandi of safety surveillance groups, although their use was an established practice at all test reactors. Other common factors include established operator training and retraining programs, detailed safety procedures and guides, close attention to control and safety instrumentation, and the keeping of basic operating and maintenance records. Control rod malfunctions provided the major source of unscheduled power reductions, with control rod magnets as the principal offender.

The research reactor survey showed inadequate formal summary documentation of operating and safety experience at many facilities, although record keeping usually improved with accumulated operating experience at a specific facility. Local safety surveillance groups were an established common practice but with widely varying modus operandi, as in the test reactor survey. Difficulty in obtaining accurate research reactor power level measurements was observed. Control rod magnet failures were frequently encountered as in the case of test reactors. Radiation monitoring practices varied widely among research reactor facilities, but difficulty was generally experienced in assuring compliance with argon-41 release limits due to monitor sensitivity limitations in the range of the 10 CFR Part 20 limit (4×10^{-8} microcuries per cc of air). Atmospheric dilution and steps to minimize neutron irradiation of air are considered adequate to avoid A⁴¹ overexposure. Site surveys have detected no significant radioactivity levels that can be directly related to reactor operations.

An investigation was initiated in 1963 under the joint sponsorship of the AEC and the Department of the Army (DOD) to develop and validate an improved basis for the selection and qualification of reactor operators. This study is being carried out by the American Institute for Research of Pittsburgh, Pennsylvania. It is anticipated that the application of human factors research techniques will yield quantitative information which will permit the correlation of measurable personal characteristics, technical knowledge, and biographical records of individuals with the probability that these individuals will or will not perform safely and competently as reactor operators. This is the major problem in the first phase of this study which is scheduled for completion in January 1965, including a final Phase 1 report. Many site visits have been and are being made for data collection purposes to selected facilities, including both government-owned and licensed research, test, training, power generation, and production reactors. Upon the successful completion of this initial phase, a second phase will be initiated to develop and demonstrate the adequacy of selection and testing predictors, instruments, and methods which can be employed with confidence to select and qualify safe and competent reactor operators at both government-owned and licensed facilities. Completion of the second phase will require about nine additional months.

In an effort to improve the basic analytical techniques for reactor safety determinations, an attempt is being made to apply the probabilistic techniques of operations research to these techniques. It is expected that a probabilistic approach will provide a more quantitative and less subjective basis for reactor safety determinations. A limited investigation is being carried out by the Planning Research Corporation of Los Angeles, California, to develop and apply an analytical model based on probabilistic damage analysis to the safety analysis of a PWR civilian nuclear power plant. The scope of work also includes the incorporation of engineering considerations (e.g., potential accidents, fission product release factors) into the basic analytical model, and the development of a general release model which will account for the major fission product release, transport, and removal phenomena involved in normal and accidental releases. The procedure is being adapted to automatic computation and applied to the Yankee pressurized water reactor plant at the Rowe, Massachusetts, site. This work is scheduled for completion in January 1965. A recent progress report indicates that conceptual study of the probabilistic model has been completed and that substantial progress has been made in implementing the model, including computer programming of the various major sequential steps in the analysis. The availability of reactor design information appears to be adequate, but deficiencies in component and operator failure experience data present a problem requiring the use of "best estimate" parameter values.

An important part of the analysis and evaluation program in nuclear safety is a research effort aimed toward establishing more adequate specifications for reactor vessels and primary system components, including piping. The specific areas of investigation are included in those deemed necessary incident to the development of an ASME Code Section covering reactor vessels and other vessels in the primary system. The first edition of such a section was issued as ASME Code Section III - Nuclear Vessels, dated 1963, but actually issued in February 1964. The areas currently under investigation are as follows:

- (1) Methods for the calculation of stresses in openings and connections in vessels and piping systems, and development of better rules for the reinforcement of such openings, being carried out at ORNL and BMI.
- (2) The development of test data to substantiate proposed methods of fatigue analysis at Southwest Research Institute.
- (3) Tests also at Southwest Research Institute to determine the significance of weld defects in cladding and base metal welds, as needed to establish realistic fabrication and inspection requirements.

The reinforced openings study is being accomplished as part of a cooperative effort sponsored by the Navy and NASA, in addition to AEC. Recent work indicates that certain commonly used design methods for both vessels and piping are inadequate. Work on spherical vessels is essentially completed and that on cylindrical shells and piping is continuing.

The fatigue investigation has included tests of full size pressure vessels. A representative test vessel is illustrated by Figure 5-1. Although all failures to date have fallen on the "safe" side of the design curve presently used, the results indicate that on the basis of statistical scatter of fatigue data some results might possibly fall on the "unsafe" side if enough specimens were tested. In two cases, terminal failure was catastrophic in character. One of these failures has provided a critical point in the development of the NRL brittle fracture analysis diagram. The second failure involved a quenched-and-tempered material which has been used in at least one containment structure. The results of this test on PVRC Vessel No. 5 are shown in Figures 5-2 through 5-6. The failure was primarily of the low energy type, in the heat affected zone of a main seam weld, and indicates the desirability of caution in the use of these materials until such time as problems associated with their welding, etc., are fully understood. Future work will include fatigue test of a model with 304 stainless cladding having defects of types sometimes encountered in current equipment. Tests will also be made on half-scale models to obtain some information on the significance of "size effect".

The third item covers fatigue test of specimens having certain types of defined defects to establish their significance from the standpoint of service life. The primary effort will be on common weld defects.

An additional area of investigation which has been undertaken is the effect of irradiation on the materials performance of fabricated structures (simulating an actual vessel). The first step in this investigation is a cooperative effort to establish the fracture characteristics of the PM-2A reactor vessel. Subsequently, it is contemplated that testing will be undertaken on model "vessels" of sufficient size and in sufficient number to establish the fracture characteristics of full scale irradiated vessels.

An investigation of reactor vessel material creep-fatigue performance at high temperatures has been initiated to provide information applicable to high temperature (e.g., gas cooled, liquid metal cooled) reactor systems. This work, which is being carried out under the Nuclear Fuels and Materials Development Program, supplements and extends the scope of the over-all Division of Reactor Development effort on the improvement of primary coolant system integrity and safety to a wider class of reactor systems.

Specialized information centers have become an established mechanism for the processing and evaluation of information in specific areas of science and technology. The Nuclear Safety Information Center (NSIC), which recently completed its first year of operation, utilizes the part-time services of recognized specialists who remain active in their specialties while contributing to the work of NSIC. Areas of nuclear safety technology initially assigned to NSIC include containment of nuclear facilities; fission product release, transport, and removal; meteorological considerations; nuclear instrumentation, control, and safety systems; radioactive effluent control, monitoring, movement, and dosage; and reactor transients, kinetics, and stability.

Figure 5-1

Full Size Pressure Vessel used in PVRC Fatigue Tests

Figure 5-2
Catastrophic Failure of PVRC Vessel No. 5

Initiation region of catastrophic fracture in PVRC Vessel No. 5.

Figure 5-3

Initiation Region of Catastrophic Fracture in PVRC Vessel No. 5

Illustrating development of fatigue cracks and terminal fracture of PVRC Vessel No. 5. Stage 1, location of initial microcracks in HAZs; Stage 2, growth of cracks by fatigue; Stage 3, initiation of fast fracture after 3603 load cycles.

Note: The numbered regions correspond to those in Fig. 5-3.

Figure 5-4

Development of Fatigue Cracks and Terminal Fracture
in PVRC Vessel No. 5 Stages 1, 2, and 3

Figure 5-5

Fatigue Striations in Zone of Fracture Initiation

V-9

Figure 5-6

Rupture Dimples in Zone of Fast-Running Ductile Fracture

The NSIC is taking an increasingly active part in the generation and exchange of information on nuclear safety technology. A smaller but fully operative replica of NSIC, designated as the Technical Information Center, was included in the United States exhibit at the Third United Nations International Conference on the Peaceful Uses of Atomic Energy. NSIC produces the quarterly journal, Nuclear Safety, which was initiated in 1959 to provide current coverage of nuclear safety technology in summary form. Mr. William B. Cottrell, Director of the NSIC, is also editor of Nuclear Safety. Responses to specific inquiries are prepared such as the recent survey of methods for handling argon-41 radioactivity requested by the AEC Chalk River (Canada) Office. Reports are also prepared in response to special study assignments by AEC Headquarters, industry, or on the initiative of NSIC personnel. An example is ORNL-NSIC-3 "Current Practices in the Release and Monitoring of I^{131} at NRTS, Hanford, Savannah River, and ORNL" by K. E. Cowser. This report summarizes information on principal sources, methods of treatment prior to release, quantities released, criteria for release, sampling methods, and other pertinent data on the control of I^{131} hazards at these AEC sites.

A survey report is being prepared on the behavior of iodine in reactor containment systems. The importance of radioactive iodine in reactor accidents, demonstrated in the Windscale and SL-1 releases, is reflected in the regulatory site criteria guide (10 CFR 100) and the related technical analysis (TID-14844). Major topics covered in the NSIC report are fission product iodine (1) physical and chemical properties; (2) characterization, including generation, particle size determination, sampling techniques, compounds and their identification; (3) transport and natural removal processes (e.g., deposition); and (4) trapping and induced removal processes (e.g., filters, scrubbers, core cooling systems, foam-suppression). A comparative summary of actual reactor fuel failure incidents and their associated radioactivity releases is also included.

Other special assignments on which investigations are in process at the Center are reactivity effects of fuel element bowing, effectiveness of safety injection systems for emergency reactivity control, kinetics of large reactor cores, review of safety analysis procedures and methods, and fast neutron gas cooled reactor safety. Reports will be issued on these studies by NSIC as they are completed.