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REACTOR CONTAINMENT
Including a
Technical Progress Review

by

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FOREWORD

This document, and future supplements are to serve as a guide and information source to those who may have containment problems, and to the staff of the Atomic Energy Commission who must analyze, judge, license, and inspect reactor installations.

To achieve this status, an attempt has been made to present available information on or pertinent to the subject of reactor containment. This is done directly, by summary and reference, or by reference alone.

To provide a reference framework, the first review document must necessarily be handled differently from supplemental periodic reviews. Therefore, the plan followed herein is to: (1) provide a detailed account of the problem and suggestions for work needed to yield adequate solutions; (2) present the accumulated knowledge and accomplishments to the effective date of this initial review; (3) give an account of experience in applying the containment concept; and (4) append extensive bibliographical material. Under (2) an attempt has been made in each case to indicate the significance of the information and its relation to the problems outlined in (1).

If one specifies the types of systems which must be considered in an omnibus on containment, one arrives at a list of all systems in which radioactive materials are involved which could constitute a danger to the public. To examine the problem reasonably and suggest work which treats the most urgent questions first, the only systems considered here are those which provide containment as the major measure for public protection rather than isolation. Because of the particular urgency of dealing with containment problems of central station reactors emphasis is placed on these, although other nonproduction reactors are not ignored. Coupling of processing plants with the stationary power reactors and in-port containment for marine reactors may soon need special attention. It is possible that in the future the doctrine of isolation may be abandoned for other systems so that they should have a niche in the program, allowing activation at a future date. To cope with new problems which will arise as these other systems require consideration, special supplements will be added in future reviews.

Future periodic supplements will present only additional material. These supplements will refer to this initial effort so that it will be unnecessary to provide background material each time one is issued.

In many respects then, it is hoped that this review series will help to resolve an important part of the general problem; the collection and dissemination of information on existing technology and that currently being produced.

SYMBOLIC DESIGNATIONS

I. Companies and Installations

AECL	Atomic Energy of Canada Limited
AERE	Atomic Energy Research Establishment (United Kingdom)
AGC	Aerojet-General Corporation
ANL	Argonne National Laboratory
APDA	Atomic Power Development Associates
ARF	Armour Research Foundation
BMI	Battelle Memorial Institute
CU	Columbia University
GE-ANP	General Electric Aircraft Nuclear Propulsion (Evandale, Ohio)
GE-AP	General Electric Atomic Products (San Jose, California)
GE-HAPO	General Electric Hanford Atomic Products Operation
HW	Hanford Works (Richland, Washington)
KAPL	Knolls Atomic Power Laboratory (Schenectady, N. Y.)
LASL	Los Alamos Scientific Laboratory
NAA-AI	North American Aviation - Atomics International (Canoga Park, California)
NBS	National Bureau of Standards
NOL	Naval Ordnance Laboratory
MSA	Mine Safety Appliances
ORNL	Oak Ridge National Laboratory
PRDC	Power Reactor Development Company
S & L	Sargent & Lundy
SRI	Stanford Research Institute
UCRL	University of California Radiation Laboratory (Livermore, Berkeley)
WAPD	Westinghouse Electric Corporation, Atomic Power Division
USNRDL	United States Naval Radiological Defense Laboratory

II. Reactors

AFNETR	Air Force Nuclear Engineering Test Reactor
APPR-I	Army Package Power Reactor No. 1
BORAX	Boiling Reactor Experiment
CP-5	Chicago Pile No. 5
Dresden	Dresden Nuclear Power Station
EBR-II	Experimental Breeder Reactor No. 2
EBWR	Experimental Boiling Water Reactor
Enrico Fermi	Enrico Fermi Atomic Power Plant
GETR	General Electric Test Reactor
HRT	Homogeneous Reactor Test
Indian Point	Consolidated Edison Thorium Reactor
KEWB	Kinetic Experiment Water Boiler
LPTR	Livermore Pool Type Reactor
MITR	Massachusetts Institute of Technology Reactor
MTR	Materials Testing Reactor
NASA-TR	National Aeronautics Space Administration - Test Reactor
PRTR	Plutonium Recycle Test Reactor
PWR	Pressurized Water Reactor
SPERT	Special Power Excursion Reactor Test
TREAT	Transient Reactor Test Facility
West Milton	Submarine Intermediate Reactor
Yankee	Yankee Atomic Electric Company
VBWR	Vallicitos Boiling Water Reactor
ZPR	Zero Power Reactor

REACTOR CONTAINMENT

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1. THE PROBLEM

1.1 General

Containment, as applied here, arises from a change in circumstances influencing the basic philosophy regarding protection of the public from the effects of uncontrolled release of the radioactive material inventory of a reactor. Studies show, and it is generally conceded, that uncontrolled release to the atmosphere of the radioactive contents of a reactor system could result in public disaster.^(1,2)

Control has been maintained in most past instances through acquisition of an adequate exclusion area surrounding the installations. The uncontrolled geophysical properties of the environment determine the transport and deposition of the radioactive material after release from the reactor system. Except for unknown genetic effects, if enough distance exists between the system and the controlled perimeter, no jeopardy is involved. The size of controlled areas generally dictated locations in wilderness areas. Several things have conspired to change this picture. Most prominent of these is the fact that location in wilderness areas is not compatible with today's need of providing electric power for our industrial complexes since this need implies location in heavily populated areas. The use of reactors in marine propulsion systems also brings problems to the public's front door. In addition, there is an increasing desire and need to use reactors for experiment and research. This need implies location at research centers, not in the "sticks." In fact, the first so-called containment vessels, the West Milton facility of KAPL, was conceived because of a combination of these desires to conduct large scale experiments on power reactors near home base rather than at the National Reactor Testing Station.

The most prominent approach to satisfying the implications of the changed picture is to provide restricted control of the radioactive materials under abnormal conditions. This new facet of the philosophy of public protection requires, first a thorough examination of the circumstances and conditions under which initial loss of control of the radioactive material inventory arises. If it is found impossible to prevent release of the radioactive materials from the multiple series of containers normally provided (fuel elements, cladding, reactor pressure vessel, and piping, for example) an additional special container usually must be provided around the reactor

complex. It must necessarily be "tight" because the radioactive sources, on release from the system, can have the form of a gas, or easily transportable micron-size solids. This is the substance of the new approach used in the process of complying with the basic philosophy in the face of the changed picture.

The government now requires that, in lieu of isolation, controlled containment be provided which is adequate to ensure the public welfare and safety under all conditions. This requirement will undoubtedly remain as long as fission is used as a power source, and it becomes more significant as the installations using the fission process for power and research mushroom in number throughout the country.

It is necessary here to point out that this does not mean providing a pressure vessel for every reactor. Use of pressure vessels is necessary in some cases. However, "Adequate Containment" may also be realized at the first line (fuel element cladding) or at the second line (the reactor vessel and piping). Underground locations or even conventional housing can be used if the "Adequate Containment" requirement is met. The requirement is solely that the public safety and welfare be maintained.

A federal statute⁽³⁾ requires that the radioactive materials be so controlled during normal periods of operation that in uncontrolled areas no individual receives doses in excess of those specified, and that the specified permissible concentrations in air or water are not exceeded. There is as yet no requirement as to the maximum permissible doses and concentrations under abnormal conditions. It has been a tacit assumption in examining the efficacy of a given installation that about half the lifetime dose suggested by the National Bureau of Standards⁽⁴⁾ can constitute the criteria for "short" exposures.

Use of such criteria, coupled with the characteristics of the site, is the starting point for determining the degree of containment control required. Conversely, a particular scheme of containment control can determine the site requirements. In any event, the reactor owner is faced with the problem of erecting a barrier between the radioactive contents of the reactor complex and the public at a cost not out of line with reasonable economy. The government must assure that an adequate barrier has been erected, and is being continually maintained.⁽⁵⁾

This approach results in a number of problems to which solutions are being sought on an ad hoc basis by the reactor builders and from a more general and continuing standpoint under sponsorship of the Atomic Energy Commission.

For the purposes of this summary, discussion of the problem is generally limited to cases in which the primary or secondary containment (fuel cladding or coolant system, for example) prove on analysis to be inadequate, requiring consideration of another line of defense. It is not intended to limit only to consideration of pressurized barriers, but to include any barrier required for adequate control if the radioactive inventory gets outside of the primary reactor system.

These general statements serve to show that the containment problem is that of providing a barrier between the radioactive materials of the reactor system and the public adequate to provide for its safety and welfare.

In examining the general problem it seems that it is composed of two classes of problems.⁽⁵⁾ One class of problems arises from the nature of the energy releases which provide the forces to move the radioactive materials and assault the barriers, and from the need to reduce the energy damage potential. The solutions to this class are required to arrange for choice and design of the supplemental containment structure. A second class of problems stems from the attempt to establish, maintain, and assure integrity of the barrier within economical bounds. Solutions to this class are needed to provide and prove containment, both initially and on a continuing basis and to allow determination of compliance with codes and regulations.

This classification of the general problem permits convenient division of the discussion to follow into two parts: one dealing with the barrier and everything within; the other with the barrier and everything without.

1.2 Part One - Within the Barrier

To serve as a guide, the block diagram shown in Fig. 1-1 is intended to cover all aspects of the assault on the barrier from within.⁽⁵⁾ By the use of the bypass leads around and among the blocks all the reactor types as well as all significant forms of accidents can be examined. Penetration from without by external causes is not shown, but any action within the barrier as a result of such penetration is considered. With this diagram it is possible to follow the course of events from accident inception through energy release to the production of mechanical and radioactive loads on the barrier. The conditions leading to an assault on the barriers are assumed to initiate from accidents of three types; viz., excess radioactivity, loss of coolant, and coolant flow stoppage. Penetration or rupture of the barrier requires an energy source. The basic sources considered are nuclear, chemical, and stored energy. Energy from one or more of these may be accidentally transformed to do work on the barrier and to transport radioactive material to a position of penetration.

Fig. 1-1

Assault on the Barrier from Within (Ref. 5)

1.2.1 Energy Sources

First let us consider theoretical maximums for each source type. The maximum value for nuclear energy is obtained if every atom of fuel undergoes fission. It has a value of about 2×10^{10} calories per gram of fuel. In the case of chemical energy the maximum value would be represented by the total heat released by all possible exothermic reactions whether they occur simultaneously or in sequence. Each such reaction may contribute between 10^3 and 10^4 calories per gram of reactant. It does not seem unreasonable to put an upper limit, then, of 5×10^3 calories per gram of reactant per reaction. The stored energy is principally the heat content of the fluid system so that it can have maximum values of the order of 10^3 calories per gram of fluid. The stored energy in graphite constitutes a special case and can be of the same magnitude.

These are generalized upper limits and are big numbers. If these upper limits could be realized, containment would of course be impossible. Careful consideration of the kinetics of each reactor system could result in the reduction of perhaps a factor of 10^6 or 10^7 in realizable nuclear energy. Most specific cases analyzed indicate the maximum values are unlikely to be greater than 5×10^8 calories for any reactor system and probably considerably less in most cases. The chemical energy which is quickly available, as from a metal-coolant reaction, can probably be shown to be lower by a factor of 10 or 100. However, when one considers stored energy of a fluid, it is evident that nearly all the stored energy is available if the fluid is contained in the reactor system under pressure. These energies do not become available at the same time. Except for the nuclear energy, the time histories of the energy releases are not well known. Furthermore, only a portion of this energy is available for doing damage, since much of it is stored as heat to be released slowly to the available energy sinks.

Let us consider the nature and progress of the energy releases in more detail. The most important consideration is that until the system is placed in a condition to produce pressures, none of the energy is available for doing the work necessary to breach the primary system. Of the nuclear energy released, a portion goes to heating and expansion of the fuel elements, and to heating and production of radiolytic gas in the coolant. The rest escapes the reactor system with neutrons, gammas, and neutrinos giving up energy on the way to the reflector and to structural and shielding material. Part of the energy which goes to heating of the fuel is transferred to the coolant with a lag imposed by heat transfer characteristics, thus, further heating and expanding the coolant. The inertial and mechanical restrictions on this expansion give rise to a pressure which can do damage. Even under the most pessimistic considerations, the expansion of the system leads eventually to a reduction in density. This in turn leads to stopping the basic nuclear energy released by terminating the chain reaction. If conditions are suitable, at some time during this initial pressure buildup due to nuclear heating, a chemical reaction will occur between the coolant and the fuel elements. This will release heat which leads to further expansions and gives rise to a new pressure pulse with a different characteristic time behavior than the nuclear source. At some time during development of these pressure waves, the container in which the thermal energy is stored may be ruptured, and the third energy source, stored energy, becomes available to do damage. This is true only if the coolant is under pressure.

In some reactors additional releasable energy is stored in graphite through dislocation of atoms under irradiation. It is probable that this rate of release is so slow as to make it unnecessary to consider this source in determining the possibility of breaching the primary system. However, the possibility of explosive release has not been eliminated. If it were shown that such an explosive release of the stored energy in the graphite were possible, such a release could be handled in the same manner as the chemical energy source.

To effect loading of the containment shell, the integrity of the primary structure or the pressurized system must be destroyed, permitting the contained fluids and gases to expand into the region bounded by the shell and to create missiles. This is made possible by the basic energy releases discussed above. One can now proceed to examine the loading of the barrier, both mechanical and radioactive.

1 2 2 Mechanical Loading of the Barrier

The mechanical loadings are classified in the following manner:

(1) Internal pressures arising from expansion of the reactor materials in vapor or gaseous form into the environment surrounding the primary reactor structure and bounded by the barrier shell.

(2) Internal pressures arising from chemical reaction between the material of the reactor and the environment within the barrier after contact.

(3) Internal pressures arising from the addition of fission product heat to the gases within the shell after shutdown of the nuclear chain reaction.

(4) Concentrated loads which may arise from spalling of the primary structure.

(5) Concentrated loads arising from conversion of ejectable parts of the system to missiles.

(6) Concentrated loads applied at contact points through items normally attached to or in contact with the barrier such as structure or cranes.

(7) Other mechanical loads.

Consider these in more detail to permit further elaboration and clarification as follows:

(1) Pressures due to expansion of core and coolant.

The rupture of the reactor pressure vessel or components of the cooling system would find liquid or pressurized gas being expanded into the gas space enclosed by the outer containment vessel. Subcooled liquid in the case of a pressurized water reactor, or two-phase liquid vapor in a case of a boiling water reactor, will dilate to a mixture of liquid and vapor, or vapor only, if the constraints are removed; a compressed vapor or pressurized gas will expand. During this expansion process, heat will be transferred to the gas normally present, such as the air, resulting quickly in a thermodynamically stable mixture except for heat losses to the solids present and to the sink outside the barrier and except for heat addition from the fission product source. These expected processes usually proceed more slowly and affect the pressure time histories only in the case of relatively slow escape. The analysis of the heat balance is made difficult because the rate of release of the fluids can vary over a large range, from slow heat storage with a few degrees temperature rise and no appreciable pressure increase to creation of a shock wave.

Two simple time independent calculations are frequently made in an attempt to establish the upper and lower limits of this range. The first, which is applied to slow releases, assumes heat storage in the system at uniform temperature and no heat transported outside the system.

The second, applied to fast release, assumes all the heat remains in the gas or vapor.⁽⁶⁾ In the first case, an overestimate of the lower bounds can result. In the second case, an overestimate of the upper bounds is obtained from the thermodynamic standpoint, but an underestimate is possible if a pressure wave results during the expansion. For conservative results, the loading must include information on such a wave, but critical partition of energy is not easy to specify. The pressure time history is strongly dependent on: (a) the degree of constraint removal as damage is done; (b) heat dissipation rates for the case where access to the gas space is restricted; and (c) the heat production rate.

(2) Pressure loads due to chemical reaction.

Breaching of the system can result in expansion or expulsion of the coolant and of core materials in solid, liquid, or vapor states allowing combustion upon contact with the air. This process applies to the case of liquid metal and some organic coolants and to pyrophoric core materials such as uranium metal. The time histories of the resulting pressure and temperature rises will depend on the quantities of reactants in contact and on rates of combustion.

Access to the air is also afforded any gaseous combustibles which may have been previously released in the initial chemical reaction (hydrogen from metal-water reactions, for example). These may mix with the air and steam in percentages favoring further combustion, or even possibly a detonation. Subsequent cooling could result in a partial vacuum, with differential pressure leading to collapse.

The pressures considered can be quite high, but fortunately there are at least two recourses if a barrier cannot be provided to withstand the loading. The obvious one is to provide an inert atmosphere within the containment vessel, which entails only a minor inconvenience. Alternatively, the more expensive venting scheme is available to take care of continuous burning. This provides for an inertia block to seal a vent during a transient pressure wave phase and later allows venting to a heat sink, filters, and gas storage space. Heat removal by spraying within the barrier is also proposed to reduce pressures and quench, although in some cases such a procedure would conceivably add to the hazard.

(3) Pressure loads from fission product heat after a shutdown

By radioactive decay, the fission products continue to release energy for an indefinite period after the chain reaction has subsided. They thus provide a continuing heat source. Immediately after shutdown of the chain reaction, this heat source will be 6-8% of the normal operating power level of the reactor. This heat source diminishes very gradually, having a time dependence like $t^{-0.2}$. This heat source, after cooling has been

interrupted, may be sufficient to melt and even vaporize some of the fuel. This leads to a side effect resulting in additional spreading of the fission products within the barrier. The heat from the fission products will be given up to the surroundings, serving for a period of time to gradually elevate the temperature of all the solids, liquids, and gases in the system. If this were the only source, then the temperature of the gas in the containment shell would continue to elevate until the heat transfer rate through the shell to the outside heat sink matched the strength of the fission product heat source. This contribution to the loading generally does not manifest itself until several hours after the earlier pressure rise, and hence is frequently neglected in determining the loading of the shell. In many cases it may contribute only a slight perturbation to the pressure time history. In some cases, however, insulation provided on the outside of the containment shell may be a sufficient heat transfer block to slow down the rate of transfer and allow the pressure to build up to a peak higher than that initially experienced. These heat balance factors cannot be ignored.

(4) Concentrated loads from spalls.

If the pressure wave which may emanate from the source region has sufficient amplitude when it reaches the outer surface of the concrete biological shield structure, spalling of the concrete may occur. If the material velocities are sufficient, and geometries and masses proper, these spalls could penetrate the barrier shell. The problem is to determine the pressure wave conditions required to cause penetrating spalls and compare this requirement with the specified wave source. If such a condition exists, the pressure wave must be attenuated by providing a special means of energy absorption before the outer surface is reached. The nature of the last line of defense, the outer gas barrier, is not conducive to economic provision of a non-penetrable barrier. Rather, the materials and cost should go into preventing penetrating spalls.

To determine the pressure wave characteristics at the outer surface, its attenuation must be followed from the source. This gives rise to a whole series of difficult problems in both the experimental and analytical approaches.^(7,8) Gaps exist in information on the equations of state for the various materials through which the pressure wave travels. There is little damage experience for sources at the center of the system having the characteristic release rates associated with the nuclear and chemical reactions in the core. Most damage experience has been realized for a wave source outside of the system. One of the few exceptions is that of the case of pressure wave behavior in water accumulated during depth charge experiments. These tests, however, do not generally have media boundary conditions and restraints found in reactor systems, nor similar source behavior. If this problem of studying wave form behavior is pursued, the damage experience for pressure wave sources simulating the real

source must replace that available for a high explosive source. The relative damages and pressures close-in and far-out for the existing experimental sources (propellants and high explosives) and the actual sources associated with a reactor, remain to be determined before applicable experimental results and analytical methods are acceptable. Even earlier attempts to obtain TNT equivalents on a damage basis for the destructive Borax test led to much smaller values than estimated on the basis of total nuclear energy released.⁽⁹⁾

Another aspect of examining the attenuation of the pressure wave energy must not be neglected. This involves consideration of the earlier lines of defense, the materials of the core, blanket or reflector, coolant, and primary containers. Although these are sometimes considered part of the source and could have been brought up when the problems of establishing the source pressure waves were examined, it is more convenient to discuss them here. The blanket or reflector, coolant, large liquid regions between source and shield, and the reactor pressure vessel can really be considered close-in lines of defense which must first be breached. Breaching the massive blanket, and stopping its pieces in a surrounding bath of liquid coolant leads to energy removal. The core pressure wave can be strongly attenuated by surrounding liquid. This energy absorption is accomplished without serious change of state of the liquid. This attenuation is especially large for large pressure wave amplitudes. The reactor containers such as the pressure vessel absorb energy from the pressure wave by conversion to strain energy during breaching. Permanent strains in the pressure vessel may even be sufficient to stop the damage process short of breaching so that integrity is maintained, especially in the less violent accidents. Such an analysis resolves into the problem of determining the energy absorption potential of reactor pressure vessels for strains above the elastic limit, up to rupture. If the pressure vessel proves to be the containment vessel, special care in its design may considerably ease the containment problem. To do this, the problems arising with respect to effects of welds, irradiation damage, penetrations, and methods of closure on the energy absorption potential become doubly important.

(5) Concentrated loads from internal missiles.

It is possible that the pressures in the primary reactor system could break loose and eject specific missiles such as control rods and drives, sections of pipe, fuel handling mechanisms, top plugs, and ancillary pressure vessels to impinge on and penetrate the outer shell. Pressure waves have access through piping to heat exchangers located outside the biological shield. If this results in disconnections at piping attachments, rocketing becomes possible on the release of stored energy through the breach.

At the outset, as for spalls, the effort should go into eliminating missile formation by design rather than trying to protect against them after they are formed. If this is not reasonably possible, the principal problem lies in determining first whether penetration can be expected; then if so, what can be done about it. The first determination has been impeded by the fact that the backlog of experience and analysis on missile penetration developed by ordnance groups does not cover the lower velocity range (below 500 ft/sec). Provision of specific protection is probably more economical than provision of general protection to the gas shell. The restriction of missiles can be realized by application of specific barriers based either on mechanical strength (armor plate) or on variable mass impedance (rope mats for example). The problems arising in considering motions of more massive pieces, such as top plugs, are sticky only because it is difficult to determine the forces acting on the body. If the forces are known, the motions, kinetic energy, and momenta of the pieces can be obtained in a straightforward calculation. It may be practical and has proved possible to provide a special structure to hold down top plugs, and protect the hold-down structure itself from inordinate damage by the pressure waves.⁽¹⁰⁾

An additional worry is the vulnerability of overhead emergency coolant storage. The emergency coolant storage tank is frequently a part of the gas barrier or outer containment shell. Although it would be useful for stopping missiles, its value in providing emergency cooling would be destroyed if penetrated. It is conceivable that missiles may be generated without complete disruption of the core. Furthermore, if the coolant is allowed to fall into the mess created in the region of the primary reactor structure after an accident, the possibility of effecting further chemical energy release must be considered.

- (6) Concentrated loads from structure or equipment in contact with the barrier.

Expansions and pressure waves can result in large displacements of material through which they act. Consequently, motion of the primary structure of the reactor and of the building structure directly connected to it may be fairly large. In order that the outer barrier may not be loaded as a result of restraint of this motion, the structural tie between the outer barrier and the rest of the system has to be elasticized to the point of isolation. The tie-in members, if any, will have an extremely small spring constant and low mass. In many cases it is possible to properly design to eliminate this form of loading. If not, this loading must be considered in the barrier design.

As a result of release of stored energy during expansion of core and coolant, or as a result of missiles, loads may be imposed on crane bridges and other items in contact with the outer container. These

loads will be transmitted to the outer shell as reaction loads. These concentrated loads can be expected to act at the same time as pressure loads, increasing the difficulty of analyzing the shell. The resulting problem here is that of: (a) determining magnitude and timing for the loads if they exist; (b) developing methods of analysis of the shell with combined loading; (c) giving design consideration to the possibility of relieving the load magnitude or protecting the items in contact from such loading.

(7) Other loads.

Other loads can, in general, be considered to have their sources outside the containment shell, and are usually of a type normally considered in modern structural design. They are treated here for the sake of completeness, along with the other mechanical loads rather than in the discussion of the problem from without the shell which is presented later.

Certain geophysical loads on the outer barrier to be considered are those arising during an earthquake or as a result of non-uniform earth settling. The latter loads, although not bringing about actual rupture of the containment vessel, are probably the more dangerous since they are not easily detected. In the case of earthquake loads, the loads cease after the earthquake and, even though the containment shell may be ruptured, no radioactive material escapes necessarily unless the earthquake brings about simultaneously a rupture or other accident in the primary system. In areas where earthquake loads must be considered in normal structural design practice, the possibility of damage to the primary system is explored and, if necessary, the system is isolated structurally. However, in the case of the settling loads, which normally come on over a period of time, the containment shell can be put in a state of prestress. At the time of an accident, the possibility of such building stresses is not generally considered in predicting the failure of the containment shell.

It is necessary to determine loadings and distortion resulting from settling to the extent of specifying the loads, their alleviation, and methods for detection on a continuing basis. Proper foundation arrangement can probably be considered to eliminate uneven settling which may produce distortions or prestressing. Primarily such loads are mentioned here so that they will not be overlooked in determining factors important to design.

Further geophysical loadings are pressure variations arising from changing meteorological conditions. This type of loading is independent of the problem in a sense, since it is regularly considered in structural design. In some containment structures it may provide the governing design criteria. Although such loadings are not generally large compared with possible internal pressures, a shell structure has very low

stability against buckling under external loads. Hence, when providing a design to take an external positive pressure as from wind loads, it will frequently be found that the resulting structure can withstand high internal pressures without failing. Principal reason for lack of concern is that buckling under external loads would not occur simultaneously with a reactor accident. Hence, no radioactive material would be released. There could be concern that items attached to the shell might be precipitated in such a manner that piping might be broken. However, normal design practice now employed for such loads would ensure against failure. The only problem which might arise in connection with such loading is lack of consideration.

There is one aspect of external pressure loading, however, which arises after a severe accident and must be considered seriously. If emergency spray cooling is used, for example, it may be possible to induce a net external pressure. Consideration of this hazard might lead to a one-way pressure relief system which would permit outside air to enter. Another example of how a large net external pressure might be realized lies in the case where the oxygen in the vessel is removed by chemical combination during burning. Here subsequent cooling could conceivably produce a net external pressure of about one-fifth of an atmosphere.

Penetration of the gas barrier by external missiles, such as aircraft and debris carried by high velocity winds, is also considered. Such a problem is not of primary concern because, although the outer barrier would obviously be penetrated with ease, release of radioactive materials would not necessarily follow unless the missile also precipitated an internal accident, including rupture of coolant containment system. It seems unlikely that the problem can be studied in general, and one must decide in each specific case whether such an internal incident could be precipitated. Certainly penetration cannot be economically avoided. The only recourse would be internal design changes to prevent radioactive release.

One further external loading merits discussion. This is pressure arising from external blast. It is unlikely that economical above-ground containment can be designed which will withstand the blast from a bombing attack, nuclear or otherwise. If such a requirement is set up, there is no recourse but to go underground. If this is done as a national defense measure, as it could be primarily, containment of radioactive materials could be easily ensured.⁽¹¹⁾ This would be in the nature of a bonus in addition to not losing the productive features of the facility. Studies of such an underground system have been made, and economic penalties have been found to be not too severe, adding a few per cent to overall plant costs. The additional cost could reasonably be charged against national defense budgets.

1.2.3 Radioactive Loading of the Barrier

In addition to the problems arising from attempts to establish the mechanical loads on the barrier, a discussion of problems encountered in determining magnitudes of the radioactive loads is in order. This portion of the discussion is restricted to the radioactive load within the outer barrier, and outside the reactor proper. That is, only that radioactive material which becomes available to leak out of the containment system to the atmosphere is considered. If realistic allowable leak rates are to be specified, it is necessary to know which radioactive materials and how much of each becomes available to leak out. To become available, the material must get outside of the primary reactor structure into the region encompassed by the leaking container. In addition, it must be and remain in the form of a gas, vapor, or of non-precipitating micron or sub-micron size solid particles. There is a notable lack of information needed to allow reasonable estimates of the amount of radioactive material which gets out in the proper form. This has forced the assumption that essentially all the radioactive materials are available and of the right form. Large reductions in radioactive loadings can be realized if such information can be provided. As time goes on, and more information is acquired, it will probably be found that radioactive loading is smaller by several orders of magnitude than it is now necessary to assume.

The amount of each kind of radioactive material present in the system can be estimated very closely, if one knows the time-dependent power level history and the relative intensities of the radiation fields in and surrounding the reactor. The problem is clearly to determine what and how much becomes available. The materials which might participate in the loading are fission products, fuel, and coolant (including adulterants) and structural material which have been transformed by neutron capture or other nuclear reactions. There are a multiplicity of ways whereby these radioactive materials can be transported to positions where they constitute a load in terms of reference significant to the containment problem, and these ways vary markedly from system to system. Most of these are outlined in Fig. 1-1. For a particular accident and a particular system it should be understood what percentage of each of the critical materials moves from the source to the barrier and of this what fraction remains in the air as a function of time. In accidents, the materials are first dispersed in the coolant through melting or rupture of fuel elements (liquid homogeneous systems already contain fission products) along with erosion and corrosion products, then find passage to the barrier through breaches in the coolant system, or the materials are forcibly ejected through a major core vessel and shielding rupture at the time of the explosion.

FIG. 1-2
RADIATION EXPOSURE AND CONTAMINATION (REF. 5)

On breaching the further volatilization to a lower pressure results in additions to the normal gaseous products and production of aerosols as well as miscellaneous particle sizes up to chunks of fuel and structure. The ejected materials, even the large sizes, may burn on contact with air, increasing the volatiles and fractions of smaller particles. It is possible, then, to have an initial source which is added to as burning continue, and subtracted from by agglomeration and fallout, or by condensation on contact with cool surfaces. The fallout is complicated by conversion of normally gaseous fission products by decay, and of volatiles by cooling. The heat transfer is complicated by continued after-glow heat contributions from the fission products. The fallout by diffusion and settling processes under gravity is complicated by agitations of turbulence and convection. Superimposed on these factors is the radioactive decay resulting in reduction of source strength with time by conversion to more stable isotopes. This latter factor is inexorable and governed only by the number of radioactive nuclei present. Unfortunately, the general decay scheme is slow, proceeding like $t^{-0.2}$, and the half-lives of the more dangerous materials which must be considered are very long, often of the order of years or centuries.

All these factors conspire to pose a very difficult problem if one really wishes to estimate the radioactive load available to leak out through the barrier at any time. In spite of the large difficulties involved, an attack on the problem and its solution provides by far the most fruitful way of alleviating the overall containment problem and hence the situation cannot be ignored.

1.2.4 Summary

The principal lines of attack with regard to the problem from within are then: (1) increasingly accurate determination of the maximum possible initiating energy release (nuclear and chemical), where one can hope to pick up reduction factors of 10 or 100 for the current pessimistic maximums; (2) determination of a reasonable division of energy into that soaked up and that which goes to physically damage the system; (3) accumulation of damage experience for systems having energy releases of the form encountered, and comparison with currently used assumptions of release as from propellants and high explosives; (4) a more accurate appraisal of the release of stored energy with respect to the effect of its time history on the pressures and temperatures and removal of energy to existing sinks which may introduce reductions in loading by factors of 10; (5) a better determination of pressure loads from materials burning after ejection into the space proximate to the barrier, which may also lead to loading reductions by a factor of 10; (6) an attack on the problem of reasonably estimating the motion of radioactive materials to a position of availability for discharge from the system, and determination of the time rate of change of this loading as a result of removal through fallout, precipitation, condensation, and radioactive decay to stabler states, all of which could yield reductions in radioactive loading assumptions of 100 or 1,000 initially, and perhaps an additional 10 or 100 subsequently.

It may not be unreasonable to hope to nearly eliminate or greatly reduce the overall problem if the various lines of attack yield even partial success.

The foregoing discussion has dealt with the problem of obtaining the design loads, both mechanical and radioactive for the outer barrier. Because many of these problems are very difficult to handle, and because reduction in loading has appeared to be the most fruitful line of attack to follow in attempting to improve the situation, most of the effort has been concentrated in this area until recently.

Currently, however, the increasing activity in actual construction and operation of research and power reactors has produced more pressing problems. This leads to consideration of the barrier itself and the region beyond.

1.3 Part Two - The Barrier and Beyond

Currently, the more pressing problems revolve about design of the barrier, its construction, testing, and continuing maintenance. Certain obstacles hamper successful realization of these activities. The obstacles consist of inapplicability of existing codes to this type of structure, lack of designation of what constitutes a maximum allowable damage to the public

from leakage after an accident, accuracy limitations on initial leak rate tests, non-uniform applicability of specification of allowable leak rates, and lack of means of proving that the leak rates do not increase in the future.

Consider first an examination of problems of design and construction of the outer barrier, having determined the loading. The loading may require a vessel to withstand a considerable overpressure. The ideal shape for resisting pressure loads is spherical, but other requirements of the facility may lead to choice of a cylindrical shape with hemispherical or hemi-elliptical ends. This condition often arises when it is considered that frequently a greater proportion of space is unusable in a sphere than in a cylinder, and that an operational design of the reactor itself may prove to extend vertically much more than horizontally, principally because of vertical travel of control rods and fuel handling systems.

1.3.1 Code Application

Certain features set the containment vessel aside from the usual pressure vessel, leading to difficulty in applying pressure vessel codes.

The reactor containment vessel normally operates with little or no pressure differential across the shell. It is designed to contain the radioactive materials which may be released from the reactor system under accidental conditions. Many accidents resulting in release of radioactive material may not contribute to the loading of the reactor containment vessel. The type of accident resulting in a significant pressure rise would occur only once during the lifetime of the reactor, if at all. In power reactor systems this accidental pressure rise could result in overpressures of one atmosphere or more. Many of the reactors, especially in the category of research reactors, would experience a pressure rise of only a tenth of this during the worst possible accident. Collapsing pressures might govern for such system designs.

As written at the present time none of the available codes cover very closely the present conceptions for containment vessels with their unique requirements. Principally, two codes have been used as guides or have been complied with. For the cases where design loads are greater than one atmosphere, the ASME code for unfired pressure vessels⁽¹²⁾ has been followed. In the case of low power research reactors where one expects the vessel to have design pressures of a few psi, and shell stresses considerably lower than the allowable stresses for the material, the most used guide is that of the American Petroleum Institute for low pressure storage tanks.⁽¹³⁾ Since the design here may be principally governed by questions of stability under geophysical loads, there are few problems raised in application to containment vessels. On the other

hand, because of the nature of the loading and because of unique features of construction and operational requirements, application of the ASME code raises problems of compliance and undue penalties.

The most important of these revolve about the fact that the ASME code assumes continuous operation at the design pressure which in reality occurs only once for a short time. Coupling the maximum possible loads with the allowable stresses cited in the code at present is unrealistic. The allowable stresses now cited assume that the structure is continuously under the design load. They take into account such items as weathering, fatigue failure, stress corrosion, and some fairly sizable factors of safety (perhaps of 4 or 5) to give the structure a long life at the design load. The problem would be lessened if, in design, one were to use the maximum possible loads in such a way that the stress encountered in the maximum over-pressure would be less than the yield stress of the material. This would allow normal operation of the facility with safety and, in effect, would not impose the continuous loading condition on the one-shot maximum possible loading case. Furthermore, the dynamic nature of the maximum loading (since it involves an analysis of the time-motion response) prevents one from using a static loading analysis. Lack of designation of analytical procedures to be used in design of shells for arbitrary pressure-time loading diagrams increases the problem. This unique dynamic loading raises the further question as to whether the materials specified for use by the code are the best suited for dynamic loading application.

There are other principal difficulties encountered in attempting to apply the existing unfired pressure vessel code to containment vessel design, construction and testing.⁽¹⁴⁾ Reactor systems normally call for constant large volume transfer of the air within the vessel (ventilating air), except during emergencies. During these emergencies a leak-tightness requirement is necessary. The code deals primarily with the strength of the structure and not with leakage. In addition, it specifies provision of pressure relief valves, which is incompatible with the desired behavior. The code does not cover use of novel structures which might be proposed for design in some drastic cases (e.g., the built-up double wall structure, the sandwich construction, and use of reinforced or pre-stressed concrete structures). Other questions arising because of the special nature of the containment vessel such as corrosion allowance, supports, and treatment of openings and penetrations further typify the problems encountered in attempted application of existing codes in design to meet insurance and local statutory requirements.

Further problems exist in attempting to comply with code test specifications. The unfired pressure vessel code is normally for a shell in which a liquid or gas is stored. In the case of the reactor containment vessel an operating plant is stored. The test requirements do not take this into account. To comply with code, the vessel would have to be

initially constructed and supported free of contact with full foundations so that complete inspection can be maintained during the tests, and with separate supports for the internals during the test. Furthermore, all the openings must be closed. A large temporary opening later must be made to allow installation of the balance of the equipment and concrete. A further test might be necessary after the closure is made, and equipment inside the structure might be unable to withstand such a test.

The remarks on testing lead to examination of the question of leak-tightness and its relation to damage to the public which form the basis of a whole additional set of problems of current concern.

1.3.2 Damage Criteria

If the maximum allowable damage to the public could be specified and the fraction of the radioactive load required to do this damage estimated, the allowable rate of penetration of the barrier by the radioactive load in the form of an allowable maximum leak rate could be determined for inclusion in the testing requirements.

If maximum allowable leak rates are to be specified, there is the problem of treating the specifications on a uniform basis for all reactors, taking into account the radioactive load and damage target for each case. One would expect, for example, that an allowable leak rate for a research reactor barrier would be several orders of magnitude larger than for a large-scale power reactor to result in the same damage. Currently, there is no uniform set of ground rules defining what containment is to accomplish. Ostensibly, the purpose is to protect the public. It is impossible to guarantee and prove that zero leakage exists, hence, in the event of an accident, some material probably escapes. Because of the remote probability of accidents of a type which will result in escape, it would seem unreasonable to specify acceptable limits of damage to the public in accord with established maximum permissible levels based on a continuing dose such as contained in the Federal Code.⁽³⁾ The problem is insurmountable unless acceptable emergency doses and contamination loads are defined and applied. Without uniform definition, the authoritative bases⁽³⁾ are difficult to interpret,^(4,15) resulting in specifications which vary over a wide range from project to project.

Numerous schemes are available for relating radiation damage to source strength. The many facets and complex nature of the relation are illustrated by the "flow" diagram of Fig. 1-2. Unfortunately, in estimating damage following reactor accidents most schemes are based on having the radioactive material in a cloud formed at one time, and not on the more realistic case associated with continuing leakage from a container which could be termed the "plume" case. This latter has been widely studied in the case of stack gas dispersion, but the problem of adopting

these studies to the reactor case has seldom been applied. If the continuous nature of the source can be substituted, along with radiation from the shell itself as a continually weakening source, in the currently available schemes for estimating doses and depositions the problem of determining allowable leak rates is highly resolved. The unified approaches already suggested can form the framework for establishing the proper relations between radioactive load, leak rates, and damage. Application through use of automatic digital computers can lead directly to establishing the maximum allowable leak rate requirement for a specific installation.

In lieu of an integrated scheme yielding a closed solution to the problem of determining a maximum permissible leak rate, one can consider the overall problem of site selection and relation of the bad accident to it. This can be done on the basis of the following broad procedural outline to yield a leak rate specification.

Select the site on the basis of normal operating requirements, including future expansion. Then select a reasonable leak rate, emergency dose and contamination limits, and check to see whether adequate land has been acquired, remembering that the vessel also radiates as a gamma source. If land is inadequate, acquire more, reduce leak rate, shield source, go underground (wholly or partly), or apply any desired combination of these. The decision is made on economic grounds or on the basis of a lower bound to the specified leak rate. Note that these factors will have a strong influence on plant design and, hence, are among those which must be considered at an early stage.

1.3.3 Leak-tightness

The leak rate requirement established by closed solution or by the procedure above is the one which must be met in tests demonstrating that a satisfactory barrier has been provided and is being maintained. It may prove to be that the allowable leak rates for some reactor systems are so low that currently available methods of leak testing are not accurate enough to determine whether the criteria are being met or not. The problem revolves around the difficulty of determining pressure changes in a system which is non-isothermal with respect to space and time. In a constant temperature system one could presumably pressurize, check the pressure decay over a period of time, and determine the leak rate on a constant volume basis. The energy interchange with the system, however, has gross variations from day to night with minor variations of smaller amplitude and frequency superimposed. In addition, with the typical large structures there can be a vertical and a radial temperature gradient.

The accuracy of the test is a function of the test pressure. Although the power reactor systems may be pressurized in test to one or more atmospheres, many of the systems such as research reactor containment vessels are normally tested at much lower pressures (a few psi). In the latter cases the smallest leak rate that can be detected, using the same measuring techniques, will be an order of magnitude larger than in the former case. Another factor in the problem is that leaks which occur at high pressures do not always show up at more modest pressures.

It does not seem possible to ensure that leak rates will be smaller than about 0.01%/day/atmosphere overpressure. Continuing entry is required for men and materials, and continual maintenance of very low leak rates at necessary points of penetration seems not practically realizable. Also, it appears from the experience to date that for the pressure measuring schemes now conceived, one might hope for an accuracy of the order of 0.01%/day/atmosphere gage. If it develops that the allowable leak rates must be substantially smaller than this, then one must look for other schemes for testing and means of improving maintenance of leak tightness. It may be that methods found satisfactory for subsequent testing will prove more sensitive.

The problems posed by requirements which may be set for continuing maintenance of leak tightness offer even more difficulty than do those of initial acceptance tests. The two basic systems now in use require pressurization and measurement of changes in pressure with suitable correction for internal variation of temperature and humidity with respect to the region outside the system.^(10,16) This is true even in the second type method, the reference system,⁽¹⁷⁾ where some measure of temperature compensation is realized automatically. The fact that all the components of the reactor system are included inside the containment structure after the initial test makes future periodic checks by the pressurizing method nearly impossible. Much of the equipment and components required for normal plant operation are too sensitive and fragile to allow large pressure increases to be used in checking containment after the plant is in operation. Furthermore, to be consistent with the safety requirements of the plant it is necessary to shut down the plant in order to apply the pressure tests. Such tests would certainly take many hours or days and the lost time usually would be intolerable. Fortunately, after initial testing, proof of maintenance of mechanical strength is not of particular concern. Hence, methods not requiring significant pressurizing might be employed. The problem really is to find and check a suitable test scheme.

Except for one, the several test schemes which have been suggested or tried are based in one way or another on including an adulterant in the building air and detecting its escape. Typical of adulterants are helium, freon, and radioactive gas. The adulterant method has an

advantage, in that it may be possible to run the tests at low pressures without inhibiting operation. It must be remembered in this connection, as noted above, that leakage which would show up at design pressures may not show up at the lower test pressures, and conversely (e.g., in the case of some gaskets of the pressure sealing types). In the case of helium, there is the advantage that people can be in the building during the test. Proposals for detection of adulterants escaping are either regional methods, installation of an overall envelope, patch tests, or specific checks at penetrations. It should be noted that, generally, soap bubble testing can be applied only in the initial test, since many of the structures are covered with insulating materials.

The exception noted is not an absolute method but is based on a very reasonable assumption. It is assumed that the general tightness of the vessel is secure on the basis of successful initial tests such as X-ray of welds and soap bubble tests. If this assumption is accepted, it follows that all leaks initially and in the future occur at penetrations. Each penetration is checked by building a "tight" box around it, external to the vessel proper, pulling a vacuum on the box and observing the pressure rise in it over a short period of time. For really tight boxes, the leakage of each may be estimated closely and all leakages summed up to yield a total. Such a test if sound, would solve all the testing problems of subsequently assuming maintenance of a tight system and may even prove superior for initial tests. The accuracy over present tests would be improved by the ratio of test box sizes to vessel size. Further improvement on this system could be effected by use of tracer gases in the containment vessel at normal pressures. The obvious advantages are of course that operation need not be curtailed and the containment vessel need not be pressurized. Of course, breaks in the vessel proper, away from penetrations (e.g., at shell welds) would never become known. It may even be feasible to design permanent "boxes" into the system. There is no difficulty in assuring the tightness of a box itself.

A particular difficulty arises because of the operational requirement that the structure be ventilated and even air conditioned, establishing inlets to and outlets from the shell requiring closures to go into action in case of an accident. Most other penetrations can be reasonably well checked as closures are made, but reliance in the case of ventilation closures is placed on a mechanical system. This is true also with vacuum relief valves. Even in the case of other penetrations, however, the problem of behavior of seals under pressure transients still exists.

1.3 4 Summary

In looking back on the problems encountered in Part Two, it should be noted that development of codes tailored to cover the case of power reactor containment vessels, and success in having such codes

written into local government statutes, would lead to more realistic design, construction, and testing. Savings in material and labor would result in sizeable cost reductions over present practice. Establishment of allowable damage to the public based on emergency doses rather than those currently prescribed by statute can elevate allowable leak rates to a point where they can be reasonably proved and observed. The uniform basis for establishing allowable leak rates which would result from this would ensure equitable treatment, in accordance with the magnitude of the radioactive loads among the different reactor systems. A more realistic treatment of the dissemination of radioactive material after leakage, will result in a reasonable relation between radioactive load, leakage, and damage thus further ensuring more reasonable specifications for allowable leak rates. A strenuous attack must be made on the problem of devising and applying suitable methods for proving compliance with leak rate requirements on a continuing basis or the aim of containment cannot be accomplished.

1.4 References

1. "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants," WASH-740, Appendix H, (U. S. Atomic Energy Commission, Mar. 1957).
2. S. McLain and R. Brittan, "Safety Features of Nuclear Reactors," Problems in Nuclear Engineering - Selected Papers from 1st Nuclear Engineering and Science Congress, Vol. I, p. 1, Edited by D. J. Hughes, S. McLain and C. Williams, (Pergamon Press, N. Y., 1957). Also ASME P/57-A-265, (The American Soc. of Mech. Engineers, N. Y.)
3. "Title 10 - Atomic Energy," Federal Register, 22, No. 19, Jan. 29, 1957.
4. "Permissible Dose from External Sources of Ionizing Radiation," National Bureau of Standards Handbook 59, Issued Sept. 24, 1954 (Supt. of Documents, Washington, D. C.)
5. R. Brittan and J. Heap, "Reactor Containment," P/437, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - Sept. 1958, (United Nations, Geneva, to be published).
6. J. Heap, "Equilibrium P-V-T, Relations for Expanding Liquid-Vapor Systems in a Containment Shell," ANL-5828, (Dec. 1957).
7. F. Porzel, "Hydrodynamic Problems of Reactor Containment," P/434, Proc. of the Second United Nations International Conf. on the Peaceful Uses of Atomic Energy - Sept. 1958, (United Nations, Geneva, to be published).

8. J. Bohannon and W. Baker, "Simulating Nuclear Blast Effects," *Nucleonics*, 16, No. 3, 74, (Mar. 1958).
9. P. M. Frye and E. C. Noonan, "Reactor Safety Program," NAVORD 4286, (Apr. 1956).
10. A. Heineman and L. Fromm, "Containment for the EBWR," P/1891, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - Sept. 1958, (United Nations, Geneva, to be published).
11. D. Inglis and G. Ringo, "Underground Construction of Power Reactors," ANL-5652, (Jan. 1957).
12. "ASME Boiler and Pressure Vessel Code," The American Society of Mechanical Engineers, 29 W. 39th St., N. Y. 18, N. Y., 1956 to date (supplements).
13. "API Recommended Rules for the Design and Construction of Large, Welded, Low-Pressure Storage Tanks (Tentative)," API Standard 620, First Edition, Feb. 1956, (American Petroleum Institute, Div. of Refining, 50 W. 50th St., N. Y. 20, N. Y.)
14. W. Siler and L. Zick, "Design Considerations for an Atomic Power Reactor Containment Structure," ASME P/57-NESC-116, (The American Society of Mechanical Engineers, N. Y.)
15. "The Biological Effects of Atomic Radiation," National Academy of Sciences, National Research Council, Washington, D. C., (1956).
16. J. Sissi and E. Diehl, "Leak Rate Test - Power Plant Building-West Milton Site," KAPL report M-6065, (Sept. 1953).
17. S. Untermeyer and D. Layton, "Leakage Testing on the Boiling Water Reactor Enclosure," VAL-33, (Sept. 1957).

2. WORK PROGRAM

General

Historically, following examination of all the problems set forth in the preceding sections, a program of work was suggested to be used as a guide for the AEC in its efforts to contribute to the solution of the problem. This suggested program has and can serve equally well as a guide to others (than the Federal Government) in fostering their desires to contribute their efforts toward obtaining practical and economic solutions to the problem of protecting the public as well as protecting their own facilities.

The program document prepared for the AEC is not generally available. Hence, for the purposes outlined above, and to serve as a reference guide to future examination and evaluations aimed at determining how far we have progressed with the solutions, this suggested program is included in the initial review. One must refer back to the discussion of the problem to determine the reasoning which establishes the need for the work, and to establish the proper inter-relations. However, some background and tie-in material is duplicated here for continuity and for purposes of describing the general class of work proposed.

A share of the work has already been started under AEC contract and by private organizations. Results of this work will be detailed in the part of this review to follow the work program outline.

It should be noted, further, that work on two subjects related to energy sources, chemical (metal-water reactions) and nuclear (kinetic reactor experiments) is separated from and given parallel program status to containment work by the AEC. It has been assumed, in preparing these work suggestions that such work by the AEC will remain separated, although the results are of interest only to the containment program. However, knowledge of the behavior of all the energy sources and their inter-relation in time, space, and magnitude is essential to the solution of containment problems, so that it is proposed that the task of gaining this knowledge be a real part of this program. The same applies to the program leading to better information on the mode and fraction of fission product release from fuel elements and to the program with regard to earthquake effects. The work required on these subjects is covered here summarily, and the reader must refer to the AEC to determine what is needed and what is being done in any detail.

Work is required on a number of facets of the problem which are general in nature, not necessarily aimed at a specific project, but leading to an advance in the technology. If such work is done on a continuing basis, each future project can improve its lot over that of earlier projects by application of the increasing store of knowledge.

One of the first things which must be tackled is to work toward establishment of realistic criteria and a set of ground rules which may be used to guide containment design on a uniform basis. Studies are to be undertaken to establish permissible leak rates on the basis of maximum of operating power level. (It seems clear that a 20-megawatt reactor plant can leak the air safely at greater rates than a 200-megawatt reactor plant if any leakage is allowable.) Then, a new and separate code and a set of specifications for design and construction of an unfired vessel with a one-in-a-lifetime loading should be prepared. Use of codes for continuously loaded vessels imposes unjustified penalties on design. Some work must be done also on the problem of proving containment (i. e., leak rates and strength), both initially and on a continuing basis for parts which may change with time.

All such work leads to establishing guides and requirements and to exercising regulation.

In line with establishing loadings, there are many total and partial voids which must be filled. Expansion of the ordnance experience on missile penetration and protection to take care of the low-velocity range is needed. Development of information which can be used to place upper limits on peaks of pressure waves near the outside of the shield must be undertaken, so that examination of a particular conventional primary structure will disclose whether special efforts must be made to design further for attenuation of the energy density to prevent spalling. In connection with this, there is a considerable gap in information on the equations of state for materials through which the pressure wave may be traveling. To calculate the pressure-time-space problem this information must be bolstered.

Some of the biggest ignorance factors can be materially reduced by fostering further attacks on the problem of mode, rates, and efficiency of energy release, including nuclear, chemical, and stored types. Along with this, it is necessary to acquire a better understanding of the physical changes taking place in the core. All these relate the three types of release on a time scale. Intermingled with this can be studies leading to a more realistic assessment of the energy division to determine the portion available to do work.

Another task is that of increasing the experience on damage evaluation for the rates of energy release encountered. It is necessary to replace the damage energy experience of high explosives with a new background on damage potential for more realistic energy-time patterns. To do this, equivalent sources for experimental simulation in both time and magnitude should be found. Coupled with this must be a real attempt to prove lack of or determine extent of the effect of scale, so that fractional scale model tests can be used.

The general problems of proving various newly proposed containment concepts also should be worked on. For example, methods of utilizing energy sinks to cut down on loads due to flashing, means of quenching fires, and venting schemes can be studied.

To summarize: the work of this program should lead to establishing the ground rules to be used as a uniform guide, the criteria to be met, the regulatory measures to be adhered to in the reactor design, construction, and operation; provision of means of compliance and control; and exploring means of minimizing the loadings.

To do this, the work has been divided into the following five Areas:

AREA I. CONTAINMENT REQUIREMENTS

Establish containment requirements based on allowable emergency doses and deposition, on the quantity and nature of radioactive poison, and on site characteristics.

AREA II. CODES AND CRITERIA

Establish or modify codes and criteria for outer barrier design construction and maintenance on the basis of unique nature of loading and utilization.

AREA III. PROOF OF CONTAINMENT

Determine acceptable method of proving initial and continuing containment from the standpoints of design loads and allowable leak rates.

AREA IV. LOADS ON THE BARRIER

Accumulate basic information necessary to establish the mechanical and radioactive loads on the barrier under emergency conditions.

AREA V. LOAD REDUCTION, DESIGN INNOVATIONS,
NEW APPLICATIONS

Devise and test proposed new methods and concepts of reducing the loading or improving the economics.

In the following discussion of each area, a general description of work precedes detailed outlines of tasks. Note that these areas are not necessarily in the same order as the associated problem discussions of the first part of this report. They conform instead to the outline of the program proposed to the AEC to avoid future reference confusion. This

should cause no great difficulty. Areas I, II, and III cover Part II of the problem outline and Areas IV and V refer to Part I.

Tasks are numbered consecutively throughout each area.

2.1 AREA I - Containment Requirements

The work here first consists of establishing the rules as to what containment is to accomplish. Ostensibly, the purpose is to protect the public. It is impossible to guarantee and prove that zero leakage exists; in the event of an accident, some material probably escapes. Because of the remote probability of accidents of a type which will result in such escape, it would seem unreasonable to specify acceptable limits of damage in accord with established maximum permissible levels of normal operation. A more realistic approach is to define acceptable emergency doses and contamination loads, with some damage and temporary or permanent property loss expected.

The necessary tasks are as follows:

(1) A working group or committee of a nationally recognized organization should make an effort to specify allowable emergency limits and recommend that these be accepted and encoded to provide a uniform design basis. The specifications would be based on limiting damage to the public to acceptable levels in the case of industrial catastrophes. All containment problems could then be reduced to a common denominator based, for example, on limits of y-roentgen dose and z-roentgen dose-acres contamination. If studies show that application of the y-roentgen dose criteria always results in an acceptable loss of use value for the land, the z-roentgen-acre requirement may be eliminated. The primary aim is physiological protection of the public in any event, and dose includes subsequent radiation damage arising from assimilation of contaminating deposits by all means before control and/or evacuation.

(It may not be possible to arrive at emergency doses which can be encoded. If such is the case, the contributions from the rest of the tasks in this Area I will be to determine the effects of various leak rates, so that recommended procedures for individual evaluation of risk can be made for each case on a nonstatutory but uniform basis.)

(2) It is necessary to arrive at a uniform scheme for interpreting radiation damage to allowable leakage at the containment site. As a first step, it is necessary to establish a uniform procedure for considering population distribution, industrial potential, and agricultural potential as background for overlay radiation patterns established on the basis of local geophysical characteristics. This procedure could be applied, not only by

government agencies regulating an installation, but by plant owners at an earlier stage in site selection. Since the damage is not alone a function of what gets out, but must be considered in combination with radiation from the containment shell as a gradually weakening source, both must be specified in determining the radiation pattern overlays.

The general procedure may make use of the military UTM grid system scheme,^(1,2) such as outlined by FCDA,⁽³⁾ and automatic digital computing methods.

Although much detailed work has been done under the FCDA program, insofar as population, industrial, and agricultural data processing is concerned, it will be necessary to apply the methods and detail available, deciding what change or augmentation is necessary to provide the data to be overlayed by radiation information.

(3) A computational scheme should be established for handling the leakage from the meteorology standpoint. (Numerous approaches have been used to perform these computations, at least two of which have been set up for machine calculations.) This is to provide information on the dose and contamination for each potential poison taken separately, which is suitable for automatic data processing. Since each installation may have its own unique poisons, the general procedure should account for space-time quantity properties of all poisons, some of which will be deleted from consideration in a specific case if appropriate. In addition, the direct radiation doses from the containment shell with varying source strength are defined by data already accumulated by agencies faced with problems of fall-out and decontamination. Provisions will be made for applying effectiveness factors which describe relative physical and biological considerations. The procedure for processing data is to be developed to give the poison overlay results on cards or tape. The procedure will involve a space-time integration, and the units for the results should correspond to those used in specifying the limits.

(4) The results of (2) and (3) above must be coordinated to provide a statistical picture of exposure and contamination, modified by evacuation procedures. Auxiliary data on potential loss of use for agricultural and industrial properties of the neighborhood will form a part of the information. The data then furnish a basis for judgment in site selection and designation of allowable leakage rates. With all necessary data for deciding an allowable leak rate encoded and stored, the specifications for a given case can be arrived at by injecting applicable values of pertinent parameters into the digital machine. The prescription code must be so designed that the question as to the maximum rate of leakage can be asked and answered if the coordinates of site location, operating power level, refueling cycle, and original material types and quantities are input data. Arrangement will

allow for insertion of modifying information such as upper estimates of fractions of material available for distribution, shielding, and special site features not stored under the general plan.

(5) An auxiliary plan should be worked out to categorize acceptable leak rates on the basis of reactor type, operating power, and fissionable material for preliminary estimating purposes. The results of this latter work can best be reported in the form of curves or nomograms.

2.2 AREA II - Codes and Criteria

The primary job and aims for this area are to come up with codes acceptable throughout the country, which better fit the conditions associated with containment vessels. These codes can be design, construction, and testing codes. They can also be procedural codes aimed at unifying the approaches used employing existing codes and methods. They can be revisions or supplements to existing codes, or they may be entirely independent. This will obviously be committee work primarily sponsored by national associations like the American Standards Association and by the various professional societies.

(1) The primary job of these committees should be to review present practice, review existing codes to determine where they fail or break down when applied to containment, and to determine revisions and supplements necessary.

The principal responsibility is to support the work of these committees and to supply them with pertinent information in some detail.

(2) Some of the work to undertake which would help cover items needed by the committees is set down here:

Organization of the code would be enhanced by including a subsection defining in detail the analytical procedures to be used in designs of the shells for arbitrary pressure-time diagrams. It is probably a basic fallacy that an elasto-plastic and plastic theory be applied to the analysis of behavior of an outer containment shell. It is probably not agreeable to count plastic deformation in the design of the shell to meet maximum loads. The probable method of design would be based entirely on elastic theory with the requirement that the stresses encountered shall not exceed the yield strength.

There is an alternative to designing to the yield strength, however, which should be explored. This would be for systems where one would determine the maximum design loads, then design the structure not to fail at one and one-half times these maximum design loads. If such a criteria is set, the problem of applying elasto-plastic theory and plastic

theory becomes real. This would require a decision on the part of those preparing the recommended codes. To cover either eventuality, a manual should be prepared outlining the analytical procedures to be used for both the elastic theory and the elasto-plastic theory. Four or five typical containment shell cases would be treated in such a manual. Initially, it is recommended that the following be included as cases: (a) a sphere supported from a skirt, above ground; (b) an imbedded sphere; (c) vertical and horizontal cylinders with elliptical or hemispherical ends supported above ground or imbedded; (d) nondevelopable partial shells of metal attached to concrete foundations; (e) concrete cylindrical shells.

(3) The committees will, in the future, determine certain additional work to be done to allow them to function properly. Such a work list should be transferred to this general program as additional tasks, and carried out.

2.3 AREA III - Proof of Containment

It may prove that the allowable leak rates for some reactor systems may be so low that currently available methods of leak testing would not be accurate enough to determine whether or not the criteria were being met. If this proves true, then new schemes must be sought and developed.

These comments refer not only to the problem of initially testing the containment but also to checking continual maintenance of containment over long periods of time. It would seem that, finally, regulations must be spelled out which will call for some continuing system for insuring compliance with the leak rate requirements. Progress in proof testing has been confined entirely to initial testing, so that the principal future efforts in this area must emphasize subsequent proof of maintenance of an adequate barrier.

It must be remembered that accuracy of leak testing may establish a limit on the allowable leakage rates specified, since nothing is gained by specifying a maximum leak rate lower than can be detected.

(1) Consider existing and proposed schemes for determining leak rates. This is to be done from an analytical standpoint, to determine feasibility and the accuracy one may finally hope to realize, using all practical refinements in set-up, and instrumentation, and all possible care in making measurements.

(2) The various schemes should be tried out after they have been reasonably evaluated from an analytical standpoint. Two approaches are suggested to carry out the second task. One is to build a special containment shell which will allow a number of these methods to be tried in sequence and also allow some work on studying openings and other penetrations. The other approach is to make use of already existing containment structures. This would require asking those in charge of such structures to try

out various schemes for checking the leak rates of their respective structures. Most of the containment structures that exist today are built around experimental or research reactors, and it would not seem too great a loss in time and effort to shut down the reactor plants when necessary. Since those operating reactor plants will be the ones faced with meeting performance criteria on a continuing basis, it would not seem to be difficult to obtain their cooperation.

Part of the investigation would include pressurizing the container and, after checking for leak tightness, arranging for orificed openings which will allow determination of the relation between leak rate and size and type of opening or leak. The investigation will also include an evaluation of the accuracy of the method as a function of test pressures. Checks can be made on the thermal lag between the conditions outside and inside structure. Since most of the buildings are normally ventilated and air conditioned, close attention should be given to the ventilation closures which come into action in case of an accident. Some work can also be done on examining the behavior of various seals under pressure transients.

2.4 AREA IV - Loads on the Barrier

2.4.1 Energy Sources

To realize the pressure loadings originating from within the barrier, breaching of the primary structure must occur first. To experience the concentrated loads, the work energy must be transmitted to external portions of the primary structure or to ejectable objects. To predict breaching, spalling or ejection, the space-time dependence of pressures beyond the source of energy release must be determined. This can be carried out for arbitrary pressure-time source forms. Then when the source forms can be specified, the specific problem can be solved.

Task 1. In order to specify eventually the source pressure-time form so that the analyses based on arbitrary forms can be used, it is necessary to interrelate basic information on energy release to initial formation of the pressure wave and subsequent behavior of the pressure source. Energy available for development of pressure is the portion of the total released that is significant in determining the damage done subsequently on the primary structure. The information being generated elsewhere on basic metal-water reactions and nuclear energy releases is to be brought together as it becomes available. Generalized means of estimating the portion of energy which can be available for producing pressures and doing damage, and of determining the probable pressure-time relation for the source will be worked out. Principal task here will be to gradually narrow down the range of possible wave forms and amplitudes to feed to those parts of the program which deal with space-time behaviors of the pressure waves away from the source, with the accumulation of damage experience, and with the simulation of the source for experiments.

Task 2. An additional energy source in some reactors is energy stored in graphite. This energy, which has been stored through dislocation of atoms under irradiation, may be released on heating. The general opinion is that the rate of energy release is so slow as to make it unnecessary to consider this source in determining the loads on the gas barrier. However, it is not certain that this rate of release is relatively slow under all conditions. It has been postulated that an explosive release may be possible. An evaluation of this energy source should be made experimentally to determine whether or not an explosive or extremely rapid release of stored energy in the graphite is possible. If it is, it will be handled as for a basic chemical energy source before breaching. If only a slow release is possible, it will be handled as for the type of mechanical loading arising from burning of core materials after ejection. In the latter case, it will augment the pressure rise as a heat source.

2.4.2 Mechanical Loading

For ease of reference, the mechanical loads on the barrier itemized under the problem discussion are repeated here.

(A) Internal pressures arising from expansion of the materials of the reactor in vapor or gaseous form into the environment surrounding the primary reactor structure and bounded by the barrier shell.

(B) Internal pressures arising from chemical reaction between the material of the reactor and the environment within the barrier after contact.

(C) Internal pressures arising from the addition of fission product heat to the gases within the shell after shutdown of the nuclear chain reaction.

(D) Concentrated loads which may arise from spalling of the primary structure.

(E) Concentrated loads arising from conversion of rejectable parts of the system to missiles.

(F) Concentrated loads applied at contact points through items normally attached to or in contact with the barrier such as structure or cranes.

(G) Other mechanical loads.

(A) Internal Pressure Loads Due to Expansion of Core and Coolant

The rupture of certain components of reactor plants would find liquid, liquid-vapor, or pressurized gas being expanded into the gas space enclosed by the outer containment vessel. It is necessary to be able to

follow the expansion of materials and heat transfer to the environmental gas if this type pressure load is to be determined.

Task 3. To provide information on these internal pressure loads it is necessary to carry out theoretical and experimental studies of a subcooled liquid, or compressed liquid, which changes to a vapor through a two-phase condition upon release of the initial constraining force. The cases of two-phase and vapor release are to be included. Studies are to yield methods of determining time-dependent pressures and temperatures and, hence, a portion of the loading of the containment vessel.

The rate of release can vary between that for a small leak and that for instantaneous removal of constraint.

The following studies seem necessary:

- a. Determine what the rates of release can really be.
- b. Show the conditions under which a shock wave can occur.
- c. Determine the degree of access to the space between primary structure and containment shell required to allow expansion at a rate just sufficient to load the shell structure in a manner that makes consideration of time dependence necessary.
- d. Determine expansion rates and loadings for better access up to 100%.
- e. Determine the relation of the thermodynamic variables as a function of time for a variety of initial conditions and opening sizes.
- f. Develop methods of analysis necessary to determine attenuation of energy available for producing and maintaining expansion pressure loads.

For the initial studies, it is recommended that the expanding material be assumed to be water and the environment to be air. Subsequently this would be extended to other materials. The analytical studies should be applied to an actual case and judged experimentally.

(B) Internal Pressure Loads Due to Chemical Reaction

To deal with the pressure loads resulting from chemical reactions, it seems reasonable to do the work listed here:

Task 4. Collect the available analytical and experimental information on burning of sodium in air in a confined space. Prepare the information in such a way as to enable pressure-time histories to be

estimated for various rates of supply and degrees of dispersion. Determine conditions under which pressures that are greater than those which can be calculated for the time-independent case are obtained. Determine what further experimental work, if any, is necessary to provide the above information. Carry out the experiments.

Task 5. Collect the available analytical and experimental information on the burning of uranium (in solid, liquid and vapor forms) in air in a confined space. Prepare the information in such a way as to enable pressure-time histories to be estimated for various rates of supply and degrees of dispersion. Determine conditions under which pressures are obtained which are greater than those which can be calculated for the time-independent case. Determine what further experimental work, if any, is necessary to provide the above information. Carry out the experiments.

Task 6. One of the special systems for producing pressures which is of some concern is the hydrogen-oxygen explosion. In many processes, such as a metal-water reaction, hydrogen is evolved. This hydrogen can accumulate in the region between the primary reactor structure and the gas barrier and can then combine explosively with the oxygen in the air if the concentrations are proper and if a means of ignition exists. Such a mechanism is one of the only ones available for producing shock loading on the gas barrier. In addition there is the possibility of induced reactions of materials which are normally outside the primary structure. For example, it may be necessary to locate reactive materials, such as gasoline, in the area between the gas barrier and the primary structure. The additional reaction which might be induced must be considered to be of the same nature as a hydrogen-oxygen explosion.

The task is to determine the conditions necessary for the hydrogen-oxygen reaction to occur in terms of concentration, to determine the reaction rate during the explosive recombination with the oxygen, to determine from this the pressure wave form which loads the outer containment shell, and then to determine the means of analysis of the motion and stresses in the shell. Finally, confirm the analysis by tests.

(C) Internal Pressures from Addition of Fission Product Heating

The work on this is listed under "Radioactive Loads."

(D) Concentrated Loads from Spalls and Missiles from Within

The nature of the last line of defense, the outer gas barrier, is not conducive to the economic provision of a non-penetrable barrier. Hence one must establish a primary design criteria or requirement of no missiles. To do this a designer must obtain information regarding the time-limited pressures or energy densities which will just produce spalling or fragmentation or rupture pipe fittings. It is necessary that no missiles be allowed.

The uncertainties in estimates and calculations require that adequate margins exist between postulated conditions and the conditions calculated to produce missiles. In order to set the limiting conditions for spall production, considerable detailed work on a variety of materials is necessary. Studies on the effect of missiles on the outer barrier need not be undertaken if no missiles can exist in the first place.

Spalls

The problem here is to provide a means of estimating whether or not spalling will occur from the outside of the conventional primary structure under the specified loading. If spalling will not occur, a means should be provided for estimating what increase in the specified loading is necessary before spalling occurs. If it will, determine (under the Area V program) the special changes necessary in the structure to prevent spalling.

Since the time dependence of the energy release which produces pressure waves in the primary structure cannot be specified in general at this time, the problem must be looked at from the standpoint of several initial arbitrary pressure wave forms. Accordingly, specific tasks for spalls have been set:

Task 7. Produce methods of determining the changes in wave form and pressure with time and space in materials likely to be encountered in the primary structure for initial pressure-time functions of arbitrary shape. Progress from one to two to three spatial dimensions.

The relation of the behavior of pressure waves in sodium, NaK, graphite, uranium, iron, and mixtures to the behavior in water or concrete should be established. This will enable work already performed on water and concrete to be useful in analysis of the other materials of interest. Similarly, produce methods of determining the space-time course of pressure waves arising from a series of initial pressure-time functions applied at intervals.

Task 8. Determine a means of experimentally reproducing initial pressure-time forms of arbitrary shape.

Task 9. In determining the behavior of the pressure waves and the material of the primary structure surrounding the energy source, a means of assessing the permanent damage is necessary. Damage experience for pressure wave shapes other than those arising from a high explosive source will be accumulated to replace existing experience with high explosives. This will be done primarily on an experimental basis, but analyses for the behavior of structural members and materials for arbitrary pressure wave load forms will also be made available and correlated with the experiments.

Task 10. Apply the methods of Task 7 to a specific case and check the method experimentally on the basis of Tasks 8 and 9. Check Task 8 against types of release associated with high explosives and propellant charges on the basis of wave forms produced.

Task 11. Determine the relation of spalling to initial pressure wave form and intensity. Determine size, shape and velocity of concrete spalls necessary to penetrate barrier shells. If the required spalls are of a nature likely to arise as a result of the above condition, prevention is necessary; otherwise, no problem exists.

Task 12. In the process of determining the energy releases and making damage assessments, some earlier lines of defense must be considered when means of attenuating the energy released in the accident is by permanent strain of the containers around the energy source. This energy would then not be available for further damage. Permanent strain of the containers, such as the reactor pressure vessels, may be sufficient to stop the accident in the case of minor, less violent accidents. In fact, the reactor pressure vessel may prove to be the containment vessel. Special care in its design, with this in mind, may considerably ease the over-all containment problem. At any rate, as a component in a study of the general attenuation of energy, the energy-absorption potential of such containers is to be determined. In addition to the study of the basic energy-absorption potential of the ideal container, the effects of welds, irradiation damage, penetrations and methods of closure on this potential are to be investigated under this task.

(E) Specific Missiles from Within

Specific missiles, such as control rods, sections of pipe, fuel handling mechanism or other penetrating operating equipment, top plugs, and auxiliary equipment, such as pressure vessels, could be broken loose and/or ejected from the primary structure to impinge on the outer shell in such a way as to penetrate and produce a leak.

The problem is to determine, first, whether this is possible and, secondly, if it is, what defense can be provided. The problem is not readily amenable to theoretical treatments and the empirical approach is recommended in general. An additional difficulty is the fact that the mode and magnitude of energy input to the missile cannot be specified.

Task 13. Fill the gap in information on penetration below about 500 ft/sec, checking by correlation with information already available at higher velocities. For the region in which penetration is possible, generate data on specific protection.

The economic demands are such that the experimental program should be carried out on a scaled-down basis, if feasible. The work, then, has to include checks on the effects of scale. The data obtained would give information on penetration limits for parameters describing the missiles and for various materials. Since certain materials are specified in codes used for shell design, these materials should be used. This would establish limits above which specific protection is necessary. Side effects, such as relation of types of support for this shell, and location of support with respect to the point of impingement, will be studied as the progressing work indicates. The restriction of missiles by specific application of barriers based on mechanical strength (armor) and on variable mass impedance (mat) is to be studied, and data are to be obtained for that region not presently covered.

Task 14. Especially useful in preliminary analysis of the design would be a procedure for determining theoretical maximum loadings on top plugs. To begin, it will be assumed that the plug is loaded either by a pressure wave traveling through a liquid or gaseous medium, by a liquid impinging on the plug, or both. A method of treatment of the problem of loading of a top plug should be suggested rather than a series of specific analyses dealing with a number of cases. The time history of the loading would depend, to some extent, on the method of tie-down of the top plug if it is not fixed in position except by its own weight.

Task 15. Piping provides access for pressure waves to ancillary equipment, such as heat exchangers or turbines. Damage or destruction to such equipment during the course of an accident might lead to making further energy available to do damage by increasing pressure and production of missiles. It is a part of this task to provide the means of analyzing piping for efficiency of transmission of damage energy to such equipment. Principles of arrangement which will allow attenuation of the pressure waves are to be determined. In addition, the possibilities of creating missiles arising when equipment is propelled by release of its stored energy after being broken loose should be investigated under this task.

Task 16. Special studies are to be made on the vulnerability of overhead emergency coolant storage. The emergency coolant storage tank is frequently a part of the gas barrier or outer containment shell. Although it would be useful for stopping missiles, its value in providing emergency cooling would be destroyed if penetrated. It is conceivable that missiles may be generated without complete disruption of the core. Furthermore, if the coolant is allowed to fall into the mess created in the region of the primary reactor structure after an accident, the possibility of effecting further chemical energy release must be considered.

(F) Loads from Structure or Equipment in Contact

Task 17. Motion of the reactor primary structure and of the building structure directly connected to it may be fairly large. In order that the outer barrier may not be loaded as a result of this motion or this desire to move, the structural tie between the outer barrier and the rest of the system must be elasticized to the point of isolation. The tie-in members must have an extremely small spring constant and low mass. This problem is to be examined to determine its importance and to recommend methods of alleviating the loads.

Task 18. As a result of release of stored energy during expansion of core and coolant, or as a result of missiles, loads may be imposed on crane bridges and other items in contact with the outer container. An analysis of the possible magnitude of such loading is to be carried out. In addition, the mode of load transmission to the barrier and type of barrier load resulting is to be determined.

(G) Other Mechanical Loadings

Task 19. The geophysical loads on the outer barrier to be considered are those arising during an earthquake or as a result of non-uniform earth settling. The latter loads, although not bringing about actual rupture of the containment vessel, are probably the more dangerous since they are not easily detected. In the case of earthquake loads the loads are of short duration and the active material does not necessarily escape unless the earthquake brings about a simultaneous rupture in the primary system and of the barrier. In areas where earthquake loads must be considered, the possibility of damage to the primary system should be explored and, if necessary, the system should be isolated structurally. In the case of settling loads, however, occurrence sometimes spreads out over a period of time and the containment shell can be put in a state of pre-stress. At the time of an accident, the possibility of such building stresses is not generally considered in predicting the failure of the containment structure. Sufficient investigation of these settling loads would be necessary to determine whether they are important. The work of earthquake study groups is to be related to the containment problem to the extent of specifying the loading and possible methods of alleviation. Some work is necessary to determine loadings and distortion resulting from settling, to the extent of specifying the loads, the alleviation, and methods of detection on a continuing basis.

Task 20. Penetration of the gas barrier by external missiles, such as aircraft and debris carried by high velocity winds, must be studied to a limited extent. Such a problem is not of primary concern because, although the outer barrier could be penetrated with ease, release of radioactive materials would not necessarily follow unless the missile also precipitated an internal accident. It seems unlikely that the problem can be studied in general. Each specific case would have to be considered to

decide whether such an internal incident could be precipitated. However, the problem should be studied sufficiently to enable specifying consideration of it in a code or set of design regulations.

Task 21. Pressures arising from meteorological variations are normally considered in design of tanks and shells. The loading occurs as a result of barometric and thermal variations. In some containment structure cases it may provide the governing design criteria. Although such loadings are not large compared with possible internal pressures, a shell structure has low stability against buckling under external loads. Hence, when providing a design to take an external pressure, it will frequently be found that the resulting structure can withstand high internal pressures without failing. With one exception, buckling would not occur simultaneously with a reactor accident, so that this type of loading is generally not of importance when considering release of radioactive material. Normal treatment of such loads in design should ensure against failure. The exception where there is a relation to fission product release lies in the post-accident behavior of the structure. Studies should be made to determine possible circumstances where, following an accident, burning and cooling may reduce the internal pressure below its pre-accident level by oxygen removal, inducing a net external pressure. These studies should include determining means of alleviating this loading (such as using one way vents, for example).

Task 22. It is unlikely that an economical above-ground containment shell can be designed which will withstand the blast from a bombing attack. If such a requirement must be met, it is mandatory to go underground. In addition to blast, post-accident radiation can be shielded by going underground. Primarily, this poses a question of economic balance between acquisition of more land, above-ground shielding, and underground installation. The study of underground structures is probably proceeding independently as a national defense measure. However, studies should be made, including economic factors, of the means of placing underground reactor systems already being considered, and of the effect of the peculiar characteristics of reactors on the general problem of underground installations.

2.4.3 Radioactive Loading

In determining allowable leak rates, it is necessary to know what specific radioactive materials and how much of each becomes available to leak out. To become available, the material must get outside of the primary reactor structure into the region encompassed by the leaking container. One of the biggest ignorance factors in the whole problem arises because of lack of information which will allow reasonable estimates of the amount of radioactive material of the proper form to leak out through the barrier. Lack of such information forces the assumption that all the radioactive materials are available and are of the right form.

Large gains can be realized toward reducing the burden if such information can be provided. It may be possible to realize radioactive loads smaller by several orders of magnitude. At present, AEC sponsored work on getting the information is proceeding as a separate program under the reactor safety program, but the information generated is not aimed specifically at the containment problem. It seems necessary to do the following under the containment program:

Task 23. Maintain liaison with those currently working on problems associated with fission product release, accumulating available information, and indicating additional useful areas of work. To determine what additional data are necessary, an attempt should be made to study the motion of radioactive materials after release from fuel elements for the following types of accidents:

- a. An accident in which the radioactive materials normally in the coolant are transported through a piping break to the air within the container.
- b. One where fission products are released to the coolant from fuel elements and are transported through a piping break by the coolant.
- c. An accident where the coolant is lost from the core region so that the fuel elements melt and vaporize.
- d. Same as (c), but reassembly to criticality occurs with a strong energy release which ruptures the primary system as the core contents vaporize and expand.
- e. Excursion occurs with accompanying release of chemical energy, part of the core vaporizing and stored energy in coolant released by major rupture of the primary system.

This approach will probably indicate that certain information and methods are lacking, so that completion of the studies will be blocked. On this basis the work necessary to fill the gaps should then be outlined. It can then be requested that this work be performed under the AEC fission product release program. Failing this, the work should be performed as part of this task.

Task 24. Obtain information on fallout and precipitation within the containment shell as a function of time. Examine, and experimentally evaluate the effectiveness of methods of accelerating this action. This would include heat removal and induced agglomeration. Aggregate this information with the already properly available data on reduction of activity with time through the natural radioactive decay schemes.

Task 25. Perform actual tests on radioactive material release from a scaled-down containment vessel to correlate radioactive loads, leak rates, leak sizes, and distribution functions within the shell.

Task 26. Although the fission products in the post-accident period continue to release energy, only a fraction of this is absorbed by the materials of the shell, and by the gases and other substances within the shell. The rest escapes. Of that absorbed, most all is converted to heat energy.

Since the total energy power level of the fission products initially is greater than 6% of the operating power level and decreases but slowly, the contribution of even part of this to heating the gas in the shell can be considerable. It is necessary to determine the procedures and methods for estimating the fission product contribution to the heat sources, and the influence on pressure-time history. This should include a parametric study relating source strength to pressure rises through temperature increase of the free volume of gases in the container, based on rates of conversion of the beta and gamma energy to heat.

2.5 AREA V - Load Reduction, Design Innovations, New Applications

Other than obtaining a better understanding of the fraction of the source energy release available for producing damage, other conditions of the problem should be considered in deciding what energies are available for damage to the barrier and how they may be alleviated. These are time and mass and strength of materials necessarily present as an adjunct to power reactors because of considerations other than containment, such as biological shielding. The necessary presence of mass (to provide shielding) affords a means of working with the other condition, time, to allow conversion of the work energy to harmless stored heat energy. One considers ways and means of most usefully employing what must be there anyhow to reduce the damage energy available. The importance of the time condition can be shown best by assuming that the energy is released very slowly and has time to spread nearly uniformly throughout the available mass of material of the reactor primary structure, including the shielding. When this is accomplished slowly the energy released has no manifestation other than a rise in temperature of a few degrees. In the case of mechanical strength of the materials in the primary structure, one considers the potential energy of strain required to cause such containers to fail. Although it does not seem possible at the present time to slow down the energy transport to a point where one would get simple uniform heating of the primary structure, this can be accomplished to some degree.

At the present time it appears that satisfactory energy attenuation can be attained for total releases of the order of 10^9 calories without employing any special means. The picture is currently being made more

satisfactory by examining means of attenuating the energy density within the confines of the shield. The attempt is to reduce the rate of energy transport and to stretch out the time scale to allow time for bringing absorption of energy by the various sinks which may be provided. It is necessary to study various choices and arrangements of materials around the reactor to optimize this attenuation and time delay. As these special means of delay are brought into play, the pressure wave intensity can be reduced at the outer boundary of the shield, so that spalling is obviated for increasingly larger initial energy releases. Loads on vital close-in structural members are also alleviated.

In addition, the actual expansion of the core materials, whether it be fuel elements or coolant, can be arrested by energy removal through conduction of heat to whatever energy sink may naturally exist or may be provided. The vapors of flashing pressurized water, for example, can be condensed in a large water reservoir if enough time is available.

(1) One field under this area involves examination of energy absorption from the pressure wave by means of phase, state, or structural changes in the materials of the primary reactor structure. By deliberately including materials in the shield region which change phase or state on passage of the pressure wave, the energy necessary to bring about the change is absorbed from the pressure wave, to be stored (usually as heat) and released later at a slower rate to the surroundings.

An exploration should be undertaken which will examine the possibility of utilizing alteration of crystal form or heats of fusion and vaporization as means of energy absorption by conversion from the energy of passing pressure waves having maximum amplitudes below 100,000 bars. The initial task approach is to suggest specific materials and mechanisms and propose confirming experiments. The task scope can then be increased to include the experimental work.

(2) A second generalization can be made for the case of utilizing permanent structure changes in a material to absorb energy from the pressure wave. It should be recognized that, in order to absorb energy, the structural changes must be permanent, since elastic recovery will not allow significant time lags to be realized. Work has been started on this concept, which has been labeled variously as "Blast Shield" and "Energy Absorption Potential." This work should be carried on past the basic stages to large-scale tests of concepts which have been investigated. Use of large-scale tests is proposed to correlate basic tests with explosives and shock-tube techniques, so that the latter can confidently be employed in "weeding-out" examinations. The proposed tests will be on simulated shield structures, with and without the particular special arrangements incorporated, and with the criterion for efficiency being based on pressure measurements and on extent of damage assessments. Maximum utilization of side information on spalling limits and on comparison of wave forms for

different initial release mechanism and different interior conditions (such as having a large liquid region between the energy source and the shield as opposed to a close-in shield) will be considered in planning and instrumenting the experiments. The tests will be designed and analyzed on the basis of existing theory and data, and actual behavior will be compared with predictions.

(3) Loading resulting from the expanding materials of the reactor can be alleviated if the energy content can be reduced before the gas space between outer shell and reactor primary structure is reached.

One system for reducing certain types of loading is incorporation of an energy dump normally isolated from the system by diaphragms or blow-out plugs. It is necessary to determine under what conditions such a scheme will be useful, so that it may be specified or recommended where necessary. Part of this will be a study of the relation of the strength and inertial mass of the diaphragm and blow-out plugs to the time dependence and magnitude of the expected loads. In addition to this, a study will be made of the relation of the size and nature of the energy dump to the expected reduction in loading. The energy dump could be either an additional available volume or a heat sink, such as a large water tank, to which the expanding products of the source could have access.

(4) Consideration should be given to means of eliminating or curtailing pressure rises resulting from a continuing burning of materials, such as uranium or sodium, after ejection to the space between the primary reactor structure and the gas barrier. It is possible that continuous burning may continue the pressure and temperature buildup to the point where venting through a heat sink and filters is the only practical solution. Such a system has been partially evaluated. The evaluation will be continued by examining a venting system in which the access to vent has a mass block available during time of peak over-pressures from a blast, a filter system, and an energy sink, followed finally by a storage cell to contain gases still present after removal of the heat. In addition, the possibility of quenching and energy transformation within the barrier is to be studied. The evaluation implies production of new design concepts, including extension of existing work on filters. Experimental evaluation should be used as well as analytical evaluation.

(5) It is necessary to give some consideration to heat rate balances between the various heat sources and sinks. A procedure should be developed which will detail the methods of obtaining the component contributions to the heat balance on a time dependent basis. A descriptive history of heat added, heat stored, and heat transferred is necessary. This procedure, if applied, will prevent overlooking pressure and temperature peaks which may occur sometime after a peak established during a blow-down or flashing operation. In addition consideration of proper heat rate balance can result in lower, more realistic loadings.

2.6 References

1. "National Location Code," Stanford Res. Institute, (1956).
2. "The Universal Grid System," TM-241, Dept. of the Army and Air Force, (Aug. 1951).
3. "Summary Report for Civil Defense Damage Assessment Program," (Fed. Civ. Def. Adm., Battle Creek, Mich., 1956).

3. WORK PROGRESS

In this section the information available to the effective date of the initial review is presented. This information has been accumulated through extensive review of sources throughout the United States, including those from work privately sponsored as well as from work performed under government sponsorship. In some cases, where the source is not readily available to the reader, or when the work is particularly significant, a rather complete presentation is made. Most of the work has been summarized; however, in some cases the information is presented by reference only.

In the summaries, references of the original author are included, but have not been checked by the reviewers. A list of references is appended at the end of each section of this collection of work progress as a convenience to the reader. Since some references are used in more than one section, some duplication in listing is evident.

The format for the presentation in this section is to group the work under the five areas shown in the Work Program (Part 2).

3.1 Area I - Containment Requirements

3.1.1 Damage Criteria

There exists only one set of statutory regulations pertaining to personal injury from radiation.⁽¹⁾ These regulate permissible exposures and concentration which may exist when handling radioactive materials. This regulation applies then to reactor operation, but only to normal operation. It sets the values to be used in providing protection against exposure to the radioactive materials handled in normal operation, normal radioactive effluents, and waste disposal. It could apply to return to use of land, crop control, and re-entry at times after an accident. It is not meant to control abnormal or accidental incidents or emergency situations.

The standards established by this regulation will be found to agree substantially with those published by the National Committee on Radiation Protection in National Bureau of Standards Handbooks.^(2,3) The National Committee on Radiation Protection has under review recommendations to limit cumulative exposures over periods of years. The Commission is giving consideration to appropriate amendments to its regulation to deal with this cumulative exposure problem.

Limitations upon levels of radiation and concentrations of radioactive material in areas affected by but not controlled by the licensee are contained principally in Section 20.102 ("Permissible Levels of Radiation in Unrestricted Areas"), Section 20.103 ("Concentrations in Effluents to Unrestricted Areas"), and the sections on waste disposal.⁽¹⁾ These sections

are designed to assure that individuals in "unrestricted areas" do not receive exposure in excess of 10 per cent of the limits established for persons exposed in restricted areas. For this purpose, the levels of radiation and concentrations of radioactive material which may be created in unrestricted areas by licensees, without special authorization from the AEC, are limited to extremely low levels. These levels are believed to be sufficiently low to assure that there is no reasonable probability of individuals, in unrestricted areas, receiving exposures in excess of 10 per cent of the permissible levels for restricted areas. Procedures are incorporated in those sections, however, under which the Commission may authorize licensees in specific cases to create higher levels in unrestricted areas where the circumstances of the particular case are such as to provide reasonable assurance that individuals in the unrestricted areas will not receive exposures in excess of 10 per cent of the limitation established for restricted areas. Exposure levels permissible in restricted areas are given in Table II, Appendix B and correspond to those in Table I, Appendix B. (Appendix B of statute is reproduced below)

It is believed that the standards incorporated in these regulations provide, in accordance with present knowledge, a very substantial margin of safety for exposed individuals. It is believed also that the standards are practical from the standpoint of licensees. It should be emphasized that the standards are subject to change with the development of new knowledge, with significant increase in the average exposure of the whole population to radiation, and with further experience in the administration of the Commission's regulatory program. Certain pertinent portions and data are reproduced here for convenience:

"...20.102 Permissible levels of radiation in unrestricted areas:

a. There may be included in any application for a license or for amendment of a license proposed limits upon levels of radiation in unrestricted areas resulting from the applicant's possession or use of radioactive materials and other sources of radiation. Such applications should include information as to anticipated average radiation levels and anticipated occupancy times for each unrestricted area involved. The Commission will approve the proposed limits if the applicant demonstrates that the proposed limits are not likely to cause any individual to receive a dose in any period of seven consecutive days in excess to 10 per cent of the limits specified in Appendix A of this part. (Appendix A of statute is reproduced below)

b. Except as authorized by the Commission pursuant to paragraph (a) of this section, no licensee shall possess, use, or transfer licensed material in such a manner as to create in any unrestricted area from radioactive material and other sources of radiation in his possession:

1. Radiation levels which if any individual were continuously present in the area, could result in his receiving a dose in excess of two millirems in any one hour, or

2. Radiation levels which if an individual were continuously present in the area, could result in his receiving a dose in excess of 100 millirems in any seven consecutive days. ..."

"...20.103 Concentrations in effluents to unrestricted areas:

a. There may be included in any application for a license or for amendment of a license proposed limits upon concentrations of licensed and other radioactive material released into air or water in unrestricted areas as a result of the applicant's proposed activities. Such applications should include information as to anticipated average concentrations and anticipated occupancy times for each unrestricted area involved. The Commission will approve the proposed limits if the applicant demonstrates that it is not likely that any individual will be exposed to concentrations in excess of the limits specified in Appendix B, Table II, of this part. For purposes of this paragraph, concentrations may be averaged over periods not greater than one year.

b. Except as authorized by the Commission pursuant to 20.302 or paragraph (a) of this section, no licensee shall possess, use, or transfer licensed material in such a manner as to release into air or water in any unrestricted area any concentration of radioactive materials in excess of the limits specified in Appendix B, Table II of this part. For purposes of this paragraph, concentrations may be averaged over periods not greater than one year.

c. For purposes of this section, determinations as to the concentration of radioactive material shall be made with respect to the point where such material leaves the restricted area. Where the radioactive material leaves the restricted area in a stack, tube, pipe, or similar conduit, the determination may be made with respect to the point where the material leaves such conduit. ..."

"...20.4 Units of radiation dose.

a. "Dose," as used in this part, is the quantity of radiation absorbed, per unit of mass, by the body or by any portion of the body. When the regulations in this part specify a dose during a period of time, the dose means the total quantity of radiation absorbed, per unit of mass, by the body or by any portion of the body during such period of time. Several different units of dose are in current use. Definitions of units as used in this part are set forth in paragraphs (b) and (c) of this section.

b. The rad, as used in this part, is a measure of the dose of any ionizing radiation to body tissues in terms of the energy absorbed per unit mass of the tissue. One rad is the dose corresponding to the absorption of 100 ergs per gram of tissue. (One millirad (mrad) = 0.001 rad.)

c. The rem, as used in this part, is a measure of the dose of any ionizing radiation to body tissue in terms of its estimated biological effect relative to a dose of one roentgen (r) of X-rays. (One millirem (mrem) = 0.001 rem.) The relation of the rem to other dose units depends upon the biological effect under consideration and upon the conditions of irradiation. For the purpose of the regulations in this part, any of the following is considered to be equivalent to a dose of one rem:

1. A dose of 1 r due to X- or gamma radiation;
2. A dose of 1 rad due to X-, gamma, or beta radiation;
3. A dose of 0.1 rad due to neutrons or high energy protons;
4. A dose of 0.05 rad due to particles heavier than protons and with sufficient energy to reach the lens of the eye;

If it is more convenient to measure the neutron flux, or equivalent, than to determine the neutron dose in rads, as provided in subparagraph (3) of this paragraph, one rem of neutron radiation may, for purposes of the regulations in this part, be assumed to be equivalent to 14 million neutrons per square centimeter incident upon the body; or, if there exists sufficient information to estimate with reasonable accuracy the approximate distribution in energy of the neutrons, the incident number of neutrons per square centimeter equivalent to one rem may be estimated from the following table:

Neutron Energy	Number of neutrons per square centimeter equivalent to a dose of 1 rem	Neutron Energy	Number of neutrons per square centimeter equivalent to a dose of 1 rem
Thermal	960×10^6	0.5 Mev	38×10^6
0.0001 Mev	480×10^6	1 Mev	29×10^6
0.01 Mev	480×10^6	2 Mev	19×10^6
0.1 Mev	96×10^6	3 Mev and higher	14×10^6

"...20.5 Units of radioactivity.

a. Radioactivity is commonly, and for purposes of the regulations in this part shall be measured in terms of disintegrations per unit or in curies. One curie (c) = 3.7×10^{10} disintegrations per second (dps) = 2.2×10^{12} disintegrations per minute (dpm). A commonly used submultiple of the curie is the microcurie (μc). One μc = $0.000001 c$ = 3.7×10^4 dps = 2.2×10^6 dpm. ..."

APPENDIX A(1)

Permissible Weekly Dose

Conditions of Exposure		Doses in critical organs (mrem)			
Parts of body	Radiation	Skin, at basal layer of epidermis	Blood forming organs	Gonads	Lens of eye
Whole body	Any radiation with half-value-layer greater than 1 mm of soft tissue	600 ^a	300 ^a	300 ^a	300 ^a
Whole body	Any radiation with half-value-layer less than 1 mm of soft tissue	1,500	300	300	300
Hands and forearms or feet and ankles or head and neck	Any irradiation	1,500 ^b	-	-	-

^aFor exposures of the whole body of X or gamma rays up to 3 Mev, this condition may be assumed to be met if the "air dose" does not exceed 300 mr, provided the dose to the gonads does not exceed 300 mrem. "Air dose" means that the dose is measured by an appropriate instrument in air in the region of highest dosage rate to be occupied by an individual, without the presence of the human body or other absorbing and scattering material.

^bExposure of these limited portions of the body under these conditions does not alter the total weekly dose of 300 mrem permitted to the bloodforming organs in the main portion of the body, to the gonads, or to the lens of the eye. . ."

APPENDIX B(1)

Permissible Concentration in Air and Water Above Natural Background

Table I

Restricted Areas

	Column 1 ^a Air	Column 2 ^b Water
A ⁴¹	1.6 x 10 ⁻⁶	1.4 x 10 ⁻³
Ag ¹⁰⁵	3.6 x 10 ⁻⁵	5
Ag ¹¹¹	1 x 10 ⁻⁴	13
Am ²⁴¹	8 x 10 ⁻¹¹	4 x 10 ⁻⁴
As ⁷⁶	7 x 10 ⁻⁶	6 x 10 ⁻¹
At ²¹¹	9 x 10 ⁻¹⁰	6 x 10 ⁻⁶
Au ¹⁹⁸	3.4 x 10 ⁻⁷	9 x 10 ⁻³
Au ¹⁹⁹	8 x 10 ⁻⁷	2 x 10 ⁻²
Ba ¹⁴⁰ + La ¹⁴⁰	2 x 10 ⁻⁷	6 x 10 ⁻³
Be ⁷	1.3 x 10 ⁻⁵	3
Cl ¹⁴	1.4 x 10 ⁻⁶	1 x 10 ⁻²
Ca ⁴⁵	9 x 10 ⁻⁸	1.5 x 10 ⁻³
Cd ¹⁰⁹ + Ag ¹⁰⁹	2 x 10 ⁻⁷	2 x 10 ⁻¹
Ca ¹⁴⁴ Pr ¹⁴⁴	2 x 10 ⁻⁸	1 x 10 ⁻¹
Cl ³⁶	1 x 10 ⁻⁶	7 x 10 ⁻³
Cm ²⁴²	5 x 10 ⁻¹⁰	2.7 x 10 ⁻³
Co ⁶⁰	3.4 x 10 ⁻⁶	5 x 10 ⁻²
Cr ⁵¹	2.4 x 10 ⁻⁵	1.4
Ca ¹³⁷ Ba ¹³⁷	6 x 10 ⁻⁷	4.5 x 10 ⁻³
Cu ⁶⁴	2 x 10 ⁻⁵	2.5 x 10 ⁻¹
Eu ¹⁵⁴	2 x 10 ⁻⁸	1 x 10 ⁻¹
F ¹⁸	3.5 x 10 ⁻⁴	2.6
Fe ⁵⁹	1.8 x 10 ⁻⁸	1.3 x 10 ⁻⁴
Fe ⁵⁵	5 x 10 ⁻⁶	3.3 x 10 ⁻²
Ga ⁷²	1 x 10 ⁻⁵	26
Ge ⁷¹	1 x 10 ⁻⁴	27
H ³ (HTO or T ₂ O)	7 x 10 ⁻⁵	5 x 10 ⁻¹
Ho ¹⁶⁶	1 x 10 ⁻⁵	70
I ¹³¹	9 x 10 ⁻⁹	9 x 10 ⁻⁵
Ir ¹⁹⁰	2.2 x 10 ⁻⁶	4 x 10 ⁻²
Ir ¹⁹²	1.5 x 10 ⁻⁷	2.7 x 10 ⁻³
K ⁴²	6 x 10 ⁻⁶	4 x 10 ⁻²
La ¹⁴⁰	4 x 10 ⁻⁶	3.4
Lu ¹⁷⁷	1.5 x 10 ⁻⁵	70
Mn ⁵⁶	8 x 10 ⁻⁶	5 x 10 ⁻¹
Mo ⁹⁹	5 x 10 ⁻³	40
Na ²⁴	5 x 10 ⁻⁶	2.4 x 10 ⁻²
Nb ⁹⁵	1.3 x 10 ⁻⁶	1.2 x 10 ⁻²
Ni ⁵⁹	5 x 10 ⁻⁵	7 x 10 ⁻¹
P ³²	4 x 10 ⁻⁷	6 x 10 ⁻⁴
Pb ²⁰³	2 x 10 ⁻⁵	4 x 10 ⁻¹
Pb ¹⁰³ + Rh ¹⁰³	2 x 10 ⁻⁶	3 x 10 ⁻²
Pm ¹⁴⁷	6 x 10 ⁻⁷	3
Po ²¹⁰ (soluble)	6 x 10 ⁻¹⁰	9 x 10 ⁻⁵
Po ²¹⁰ (insoluble)	2 x 10 ⁻¹⁰	-
Pr ¹⁴³	2.3 x 10 ⁻⁶	1
Pu ²³⁹ (soluble)	6 x 10 ⁻¹²	4.5 x 10 ⁻⁶
Pu ²³⁹ (insoluble)	6 x 10 ⁻¹²	-
Ra ²²⁶ 1/2 dr	2.4 x 10 ⁻¹¹	1.2 x 10 ⁻⁷
Rb ⁸⁶	1.1 x 10 ⁻⁶	9 x 10 ⁻³
Re ¹⁸³	2.4 x 10 ⁻⁵	2.4 x 10 ⁻¹
Rh ¹⁰⁵	3 x 10 ⁻⁶	5 x 10 ⁻²
Rh ²²² dr	1 x 10 ⁻⁷	6 x 10 ⁻⁶
Ru ¹⁰⁶ Rh ¹⁰⁶	8 x 10 ⁻⁸	4 x 10 ⁻¹
S ³⁵	3 x 10 ⁻⁶	1.5 x 10 ⁻²
Sc ⁴⁶	2 x 10 ⁻⁷	1
Sm ¹⁵¹	4 x 10 ⁻⁸	6 x 10 ⁻¹
Sn ¹¹³	1.7 x 10 ⁻⁶	5 x 10 ⁻¹
Sr ⁸⁹	6 x 10 ⁻⁸	2 x 10 ⁻⁴
Sr ⁹⁰ Y ⁹⁰	6 x 10 ⁻¹⁰	2.4 x 10 ⁻⁶
Tc ⁹⁶	8 x 10 ⁻⁶	8 x 10 ⁻²
Te ¹²⁷	3 x 10 ⁻⁷	8 x 10 ⁻²
Te ¹²⁹	1.2 x 10 ⁻⁷	3.3 x 10 ⁻²
Th ²³⁴	2 x 10 ⁻⁶	10
Th-natural (soluble)	5 x 10 ⁻¹¹	1.5 x 10 ⁻⁶
Th-natural (insoluble)	5 x 10 ⁻¹¹	-
Tm ¹⁷⁰	1.5 x 10 ⁻⁷	8 x 10 ⁻¹
U-natural (soluble) ^c	5 x 10 ⁻¹¹	2 x 10 ⁻⁴
U-natural (insoluble) ^c	5 x 10 ⁻¹¹	-
U ²³³ (soluble)	4 x 10 ⁻¹⁰	4.5 x 10 ⁻⁴
U ²³³ (insoluble)	5 x 10 ⁻¹¹	-
V ⁴⁸	3 x 10 ⁻⁶	1.5
Xe ¹³³	1.3 x 10 ⁻⁵	1.3 x 10 ⁻²
Xe ¹³⁵	5 x 10 ⁻⁶	4 x 10 ⁻³
Y ⁹¹	1.2 x 10 ⁻⁷	6 x 10 ⁻¹
Zn ⁶⁶	6 x 10 ⁻⁴	2 x 10 ⁻⁷
Unidentified beta or gamma emitters or any undetermined mixtures of beta or gamma emitters	-	-
Unidentified alpha emitters or any undetermined mixtures of alpha emitters	-	-

Table II

Unrestricted Areas

	Column 1 ^a Air	Column 2 ^b Water
A ⁴¹	5 x 10 ⁻⁸	5 x 10 ⁻⁵
Ag ¹⁰⁵	1.2 x 10 ⁻⁶	1.6 x 10 ⁻¹
Ag ¹¹¹	3 x 10 ⁻⁶	4 x 10 ⁻¹
Am ²⁴¹	3 x 10 ⁻¹²	1.3 x 10 ⁻⁵
As ⁷⁶	2 x 10 ⁻⁷	2 x 10 ⁻²
At ²¹¹	3 x 10 ⁻¹¹	2 x 10 ⁻⁷
Au ¹⁹⁸	1.1 x 10 ⁻⁸	3 x 10 ⁻⁴
Au ¹⁹⁹	2.5 x 10 ⁻⁸	7 x 10 ⁻⁴
Ba ¹⁴⁰ + La ¹⁴⁰	6 x 10 ⁻⁹	2 x 10 ⁻⁴
Be ⁷	4 x 10 ⁻⁷	1 x 10 ⁻¹
Cl ¹⁴	5 x 10 ⁻⁸	3.6 x 10 ⁻⁴
Ca ⁴⁵	3 x 10 ⁻⁹	5 x 10 ⁻⁵
Cd ¹⁰⁹ + Ag ¹⁰⁹	7 x 10 ⁻⁹	7 x 10 ⁻³
Ca ¹⁴⁴ Pr ¹⁴⁴	7 x 10 ⁻¹⁰	3.6 x 10 ⁻³
Cl ³⁶	4 x 10 ⁻⁸	2.4 x 10 ⁻⁴
Cm ²⁴²	1.8 x 10 ⁻¹¹	1 x 10 ⁻⁴
Co ⁶⁰	1.2 x 10 ⁻⁷	1.8 x 10 ⁻³
Cr ⁵¹	8 x 10 ⁻⁷	5 x 10 ⁻²
Ca ¹³⁷ Ba ¹³⁷	2 x 10 ⁻⁸	1.5 x 10 ⁻⁴
Cu ⁶⁴	6 x 10 ⁻⁷	8 x 10 ⁻³
Eu ¹⁵⁴	6 x 10 ⁻¹⁰	3 x 10 ⁻³
F ¹⁸	1.2 x 10 ⁻⁵	9 x 10 ⁻²
Fe ⁵⁹	6 x 10 ⁻⁹	4 x 10 ⁻⁵
Fe ⁵⁵	1.5 x 10 ⁻⁸	1.1 x 10 ⁻⁴
Ga ⁷²	3.4 x 10 ⁻⁷	9 x 10 ⁻¹
Ge ⁷¹	3.6 x 10 ⁻⁶	9 x 10 ⁻¹
H ³ (HTO or T ₂ O)	2.5 x 10 ⁻⁶	1.6 x 10 ⁻²
Ho ¹⁶⁶	3 x 10 ⁻⁷	2.3
I ¹³¹	3 x 10 ⁻¹⁰	3 x 10 ⁻⁶
Ir ¹⁹⁰	7 x 10 ⁻⁸	1.3 x 10 ⁻³
Ir ¹⁹²	5 x 10 ⁻⁹	9 x 10 ⁻⁵
K ⁴²	2 x 10 ⁻⁷	1.4 x 10 ⁻³
La ¹⁴⁰	1.4 x 10 ⁻⁷	1.1 x 10 ⁻¹
Lu ¹⁷⁷	5 x 10 ⁻⁷	2.4
Mn ⁵⁶	3.3 x 10 ⁻⁷	1.5 x 10 ⁻²
Mo ⁹⁹	1.8 x 10 ⁻⁴	1.4
Na ²⁴	1.6 x 10 ⁻⁷	8 x 10 ⁻⁴
Nb ⁹⁵	4 x 10 ⁻⁸	4 x 10 ⁻⁴
Ni ⁵⁹	1.6 x 10 ⁻⁶	2.5 x 10 ⁻²
P ³²	1.4 x 10 ⁻⁸	2 x 10 ⁻⁵
Pb ²⁰³	6 x 10 ⁻⁷	1.4 x 10 ⁻²
Pb ¹⁰³ + Rh ¹⁰³	7 x 10 ⁻⁸	1 x 10 ⁻³
Pm ¹⁴⁷	2 x 10 ⁻⁸	1 x 10 ⁻¹
Po ²¹⁰ (soluble)	2 x 10 ⁻¹¹	3 x 10 ⁻⁶
Po ²¹⁰ (insoluble)	7 x 10 ⁻¹²	-
Pr ¹⁴³	7 x 10 ⁻⁸	3.6 x 10 ⁻²
Pu ²³⁹ (soluble)	2 x 10 ⁻¹³	1.5 x 10 ⁻⁷
Pu ²³⁹ (insoluble)	2 x 10 ⁻¹³	-
Ra ²²⁶ 1/2 dr	8 x 10 ⁻¹³	4 x 10 ⁻⁹
Rb ⁸⁶	4 x 10 ⁻⁸	3 x 10 ⁻⁴
Re ¹⁸³	8 x 10 ⁻⁷	8 x 10 ⁻³
Rh ¹⁰⁵	1 x 10 ⁻⁷	1.6 x 10 ⁻³
Rh ²²² dr	3.3 x 10 ⁻⁹	2 x 10 ⁻⁷
Ru ¹⁰⁶ Rh ¹⁰⁶	2.6 x 10 ⁻⁹	1.3 x 10 ⁻²
S ³⁵	1 x 10 ⁻⁷	5 x 10 ⁻⁴
Sc ⁴⁶	7 x 10 ⁻⁹	3.6 x 10 ⁻²
Sm ¹⁵¹	1.3 x 10 ⁻⁹	2 x 10 ⁻²
Sn ¹¹³	6 x 10 ⁻⁸	1.6 x 10 ⁻⁶
Sr ⁸⁹	2 x 10 ⁻⁹	7 x 10 ⁻⁶
Sr ⁹⁰ Y ⁹⁰	2 x 10 ⁻¹¹	8 x 10 ⁻⁸
Tc ⁹⁶	3 x 10 ⁻⁷	3 x 10 ⁻³
Te ¹²⁷	1 x 10 ⁻⁸	3 x 10 ⁻³
Te ¹²⁹	4 x 10 ⁻⁹	1.1 x 10 ⁻³
Th ²³⁴	6 x 10 ⁻⁸	3 x 10 ⁻¹
Th-natural (soluble)	1.7 x 10 ⁻¹²	5 x 10 ⁻⁸
Th-natural (insoluble)	1.7 x 10 ⁻¹²	-
Tm ¹⁷⁰	5 x 10 ⁻⁹	2.5 x 10 ⁻³
U-natural (soluble) ^c	1.7 x 10 ⁻¹²	7 x 10 ⁻⁶
U-natural (insoluble) ^c	1.7 x 10 ⁻¹²	-
U ²³³ (soluble)	1 x 10 ⁻¹¹	1.5 x 10 ⁻⁵
U ²³³ (insoluble)	1.6 x 10 ⁻¹²	-
V ⁴⁸	1 x 10 ⁻⁷	5 x 10 ⁻²
Xe ¹³³	4 x 10 ⁻⁷	4 x 10 ⁻⁴
Xe ¹³⁵	1.7 x 10 ⁻⁷	1.4 x 10 ⁻⁴
Y ⁹¹	4 x 10 ⁻⁹	2 x 10 ⁻²
Zn ⁶⁶	2 x 10 ⁻⁷	6 x 10 ⁻³
Unidentified beta or gamma emitters or any undetermined mixtures of beta or gamma emitters	1 x 10 ⁻⁹	1 x 10 ⁻⁷
Unidentified alpha emitters or any undetermined mixtures of alpha emitters	5 x 10 ⁻¹²	1 x 10 ⁻⁷

^aAir concentrations are given in microcuries per milliliter of air.^bWater concentrations are given in microcuries per milliliter of water. These figures also apply to foodstuffs in microcuries per gram (wet-weight).^cFor enriched uranium the same radioactivities per unit volume as those for natural uranium are applicable. It should be noted that the contribution of U²³⁴ to the gross activity of enriched uranium is 20-40 times that of the U²³⁵.

More pertinent to the containment problem is the permissible dose for accidental or emergency exposure to radiation. This is suggested in the NBS Handbook 59:(3)

In principle, an individual can be exposed to radiation at a certain dosage rate for the rest of his life and the dosage rate can be larger the older he is at the time the exposure starts. Taking into consideration the individual's age, previous exposure and current permissible limits for continued exposure, we arrive at the concept of a radiation tolerance status.

Accidental or emergency exposure of the whole body of adults or parts thereof to X-rays with photon energy less than 3 Mev, from external sources, occurring only once in the lifetime of the person, under the condition and in the respective dosages stated below, shall be assumed to have no effect on the radiation tolerance status of that person.

(a) Exposure of the whole body - any adult. Total dose, measured in air: up to 25 r.

(b) Local exposure - any adult. Dose measured in air and additional to whole-body dose: (1) Hands and forearms, up to 100 r; (2) feet and ankles, up to 100 r.

Emergency work involving high-level exposure to X-rays with photon energies less than 3 Mev shall be carried out on the basis that the person will not receive doses higher than one-half the respective doses stipulated above. If the doses actually received in the performance of such work do not exceed the respective maximum doses stipulated, the exposure may be considered to be in the category covered above. Women of reproductive age shall not be subjected to planned emergency exposure.

The foregoing essentially deals with external radiation sources. The air and water concentration used in the statute⁽¹⁾ was derived principally from NBS Handbook 52.⁽²⁾ This handbook, in addition, recommends maximum amounts of radioisotope in the body as well (which lead to the concentration limits for air and water). The handbook also provides a useful procedure and data for estimating and calculating amounts in the body. This is too detailed to reproduce here, and since publication in 1953, some values have been revised.

The recommendations of the National Research Council⁽⁴⁾ are a more recent source of semi-official suggestions for dosages. To provide those faced with selecting emergency doses with a background of current thinking and some quantitative information a considerable portion of this report is summarized here:

Effect on Humans

The inheritance mechanism is by far the most sensitive to radiation of any biological system. Any radiation which reaches the reproductive cells causes mutations that are passed on to succeeding generations. An unavoidable quantity of spontaneous mutations are caused by background radiation. There is no minimum amount of radiation which must be exceeded before mutations occur. From the viewpoint of the total and eventual damage to the entire population, every mutation causes roughly the same amount of harm. For the general population, and in the long run, a little radiation to many people is as harmful as a lot of radiation to a few, since the total number of mutant genes can be the same in both cases. One measure of genetic harm is the amount of radiation, above natural background, which would produce as many mutations again as occur spontaneously. It is estimated that this amount is 30 to 80 roentgens. The 30-year dose to the gonads received by the average person in the U.S. is estimated to be: a) 4.3r from background; b) 3r from X-rays and fluoroscopy and c) 0.02r to 0.5 from weapon tests (based on the rate of testing from 1951 to 1956). Large exposures (say 100r and up) of the whole body or a large part of it are generally harmful when received in a single dose. It is not yet known what minimum dose, if any, would be necessary to produce a statistically noticeable reduction of life span.

Environment and Food Supply

Radiation in the general environment has not yet become a serious problem. In a few decades, however, radioactive waste products from atomic power plants will represent an enormous potential source of contamination. How much of this radioactivity will actually reach the population depends on how successfully it can be kept out of the great network - ocean and air currents, food and water supplies - which connect man with his surroundings. At present, atomic weapons tests are the only significant sources of radiation in the general environment above the natural background. Fallout of radioactive materials subject to meteorological dispersion inevitably contaminates man's food supply. Radioactive elements in the soil are taken up and concentrated by plants. The plants may be eaten by humans, or by animals which in turn serve as human food. Probably the most important food contaminant is strontium 90 - a radioactive element that concentrates in bone tissue. Radioactive material finding its way into coastal and other surface water would directly affect marine life, and within a few years, would contaminate all parts of the world because of the relatively rapid circulation of surface layers. Food sources of marine plants and animals extract and concentrate various radioactive elements that get into sea water. The concentration is cumulative, increasing as it proceeds up the chain from microscopic plankton to edible fish.

In the light of all these considerations, holding radiation to a tolerable worldwide level will require adequate methods for containing radioactive waste from power reactors. Considering all the foregoing factors, it is recommended that radiation to individual persons should be limited to a total accumulated dose from both natural and artificial sources, to the reproductive cells of 50r up to age 30 years and not more than 50r additional up to age 40. Furthermore, the general public of the United States be protected from receiving a total reproductive lifetime dose (conception to age 30) of more than 10r of man-made radiation to the reproductive cells.

In addition to these official sources, many suggestions for emergency dose rates and contamination have been made, some of which are dealt with here:

(1) The acceptable emergency dose used in estimating injury in case of accident has been taken to be 25 roentgens of whole body gamma radiation in one exposure or 50 roentgens in three months. These values were used in an AEC document⁽⁵⁾ presented to the Congress and thus may be construed to have had at least semi-official stature and support as upper limits at that time (March 22, 1957).

More importantly, the document relates concentration in a cloud or on the ground to this acceptable emergency dose (AED).

On the basis of calculations for accidents following 180 days of operation, dose fractions were determined and acceptable emergency doses (AED) recommended for two types of release. One type assumes all species of fission products escaping in proportion to their individual inventories. The other type assumes that only the volatile and noble gas fractions escape.

Exposures are given in terms of concentration as curie-seconds per cubic meter. This unit corresponds to an exposure for one second to a concentration of one curie/cubic meter. In determining these exposures, variations in fission product energy were taken into account in calculating each dose rate. The exposures assume that the fission product cloud is first encountered two hours after shutdown.

The following categories were assigned and dose limits suggested:

	AED, roentgens, any type of release	AED, Curie-second/ cubic meter	
		All species of fission products escaping	Only volatiles and noble gases escaping
Category A (High percentage of fatality)	> 450	> 400	> 350
Category B (High incidence of illness)	100 - 450	90 - 400	80 - 350
Category C (Minor symptoms)	25 - 100	10 - 90	10 - 80
Category D (No injury)	< 25	< 10	< 10

Re-entry permissible if dose rate is less than 0.3r/week.

In addition to suggesting limits for personal injury, some limits were set for evacuation damage to agriculture. From the data and calculational methods given (the same cores are considered as for personal injury) the following limits were selected or deduced by the editor:

Non-personal Injury Limits

Evacuation Limits	Area concentration, curies/sq meter	
	All species released	Volatiles only released
If dose rate gives 25r in first 12 hrs, Urgent evacuation within 12 hrs is called for if	0.18	0.075
If dose rate gives 50r in three months to partially sheltered - partially outdoors people, evacuation within 3 months is necessary if	0.006	0.027
Temporary evacuation (3 months) loss of use of land for at least a year is necessary if	10^{-3}	10^{-2}
Loss of current crop and crop restric- tion for a year is necessary if	10^{-4}	10^{-3}
No losses on expense likely if	10^{-4}	10^{-3}

[These limits imply that for checking adequacy of containment, no concentrations shall exceed the following outside the controlled plant area:

	Curies/cu meter in air	Curies/sq meter on ground
All species assumed released	10	10^{-4}
Only volatiles assumed released	10	10^{-3}

It should be noted that assumption of no exposure in first 2 hours after shut-down is unconservative for close-in considerations, especially in the volatile release case. It should be noted further that in applying the limits given, for the leaking container case, the damage from the container radiating as a gamma source should be added to that from the cloud and ground contamination - Editor]

(2) B. P. Leonard, in a fission product release study⁽⁶⁾ listed maximum permissible dose level criteria which have received some consideration and acceptance by some medical experts. Ostensibly these are applicable to the case of a person immersed in a radioactive cloud.

Maximum Permissible Exposure (MPE) in Rem*

Whole body	25	Kidney	25
Bone	25	Muscle	25
Thyroid	1500	Lung	25

*Roentgen Equivalent Medical

(3) Lauriston S. Taylor of the National Bureau of Standards has provided an examination⁽⁷⁾ of the new permissible radiation exposure levels as recommended by the International Commission on Radiation Protection (ICRP) and their interpretation by the National Committee on Radiological Protection in the United States (NCRP) in April 1956. (Health Physics Vol. 1, No. 1, June 1958). According to this source the two principal recommendations of ICRP on maximum permissible dose are:

(a) For the population as a whole, the average per capita dose should not exceed 10 rems, from conception to age 30 (and by inference one-third of this amount per decade thereafter).

(b) For radiation workers, the individual occupational dose should not exceed 50 rems up to age 30 (and by inference again this same amount per decade thereafter).

The difficulty of reconciling these recommendations with the black-and-white requirements of our legal machinery which deals with

individuals rather than averages has resulted in attempts to interpret ICRP by NCRP. The problem has been overcome to a limited extent by a statement of NCRP relating total permissible accumulated dose to the age of an individual.

The statement is:

"Accumulated Dose: The accumulated maximum permissible dose (MPD) in rems, at any age (N) is equal to 5 times the number of years beyond 18, provided no annual increment exceeds 15 rems. Thus the accumulated MPD = $5(N-18)$ rems. This applies to all control organs except the skin, for which the value is double."

[Note: To derive maximum advantage from the presentation of L. S. Taylor, the above notes should not be studied out of context. It is recommended instead that the reader refer to the whole paper - Editor]

(4) K. Z. Morgan, et al.,⁽⁸⁾ give results of calculations, for 81 radioisotopes, of maximum permissible amounts (MPI) to be inhaled, injected, or ingested in 8 hours to give doses of 0.3 rem in a week, 15.7 rem in a year, and 150 rem in 70 years. Calculations were based on ICRP⁽⁹⁾ and NCRP⁽²⁾ values for continuous exposure. The formula developed for the calculations are also given so that MPI may be obtained for other dosage criteria.

[Note that these values of permissible doses have been superceded by the 1956 ICRP and NCRP recommendation - Editor]

(5) J. J. Fitzgerald and J. L. Flynn⁽¹⁰⁾ suggest that a safety factor of 10 is generally required in going from permissible doses or concentration for the small working population on-site to the larger population off-site. On this basis they conclude that the MPE of NBS Handbook 59⁽³⁾ (25 roentgens) could logically be reduced to 2.5r for an acceptable off-site dose under emergency conditions. Since the background dose varies from 0.1 to 0.5 r per year, such a 2.5 r off-site dose would double the dose to receptors in the immediate area in from 5 to 25 years.

(6) J. J. Fitzgerald⁽¹¹⁾ lists doses to critical organs from exposure to a fission product cloud. It is assumed from NCRP data that a dose of 100 rad over a period of one year is permitted under emergency condition. This assumption is equivalent to permitting an exposure of $38 \mu\text{c/sec/cc}$ from mixed fission products of a reactor inventory after 1 year of operation. The calculation of dosages is for a receptor at one mile from origin and a cloud traveling at 4 meters per second. A biological half-life of 50 days from the lungs was assumed for the mixed fission products.

Dose to Critical Organ, Rads

(Under conditions listed above)

Period following accident	Lung	Thyroid	Bone
0 - 1 day	4	42	1
0 - 10 days	25	105	2
0 - 60 days	79	141	10
0 - 1 year	100	141	28
0 - 7.5 years	100	141	49
0 - ∞	100	141	67

[The results of this calculation are representative of ones made frequently in hazard evaluation studies, and are presented here as such in lieu of a long list of similar studies. Also they illustrate typical doses received when the emergency maximum permissible exposure is applied to one organ - Editor]

3.1.2 Leakage and Shielding Criteria

In a paper prepared for the 1958 International Standards Organization Conference Brittan⁽¹²⁾ surveys methods used to relate containment requirements to permissible damage. Parts of this survey are excerpted and included here to yield a matrix for presenting the work of others on this problem.

In relating damage to containment requirements, one must recognize that in the case of a reactor for which containment is being specifically provided, the principal danger to the public now results from an accident in which the contents of the reactor core are accidentally released from the primary reactor structure to the space between the reactor structure and the outer containment shell. This in effect provides an extremely intense source to which the public must not be over-exposed. Over-exposure can be prevented either by keeping people away at a sufficient distance, by taking advantage of the naturally occurring topographic features which might constitute a shield, by going underground, or by providing additional artificial shielding. In addition, however, one must consider leakage of radioactive material at relatively low rates, since absolute leak tightness cannot be guaranteed. Outside of this it is not proper to consider large scale releases, since if this is done, no economic advantage but rather a loss accrues from using a containment vessel, and land acquisition would necessarily approach that for the case of no containment.

3.1.2.1 Radioactive Material Inventory

A. Fission Products

In considering the plant as a potential radiation source we must look at the source strength. The exact amounts of fission products present in the power reactor at shutdown will depend upon the type of fissionable material used, the power level history, the reloading schedule for the core, and the neutron cross section of the fission products. The reactor will produce slightly more than one gram of fission products per thousand kilowatt days of heat produced so that a 500,000 thermal kilowatt reactor will produce over one pound of these fission products each day, containing many millions of curies of both gaseous and solid products. At the same time, one or more pounds of thorium or U^{238} may be transmuted to U^{233} or plutonium. These fission products and new fuels accumulate in the reactor and by consideration of the decay characteristics one finds that under normal operating conditions there may be present in most power reactors on the order of hundreds of millions of curies of fission products and hundreds of pounds of U^{233} or plutonium.

These quantities and their energy release rates can be calculated precisely for each fission product. The fission product disintegrations result in β and γ emissions. The production rates are proportional to the fission rates in accordance with the fission yields of each of the species. The disintegration rates and transmutation rates are each proportional to the number of fission fragments present.

The expressions for the concentration of each fission product and its activity in terms of disintegrations per second, either while the reactor is operating or at any time after operation, can be written down exactly. No one basic equation will suffice, since any one of the fission products may decay by beta emission or otherwise to another isotope which in turn may decay. Its daughter then again may experience a further disintegration. Many generations may grow up and die before stability is achieved. For any given fission product chain or any given number of daughters, or grand-daughters, an exact expression can be written. However, since there are so many types of fission products, the general analyses become difficult if one does not resort to the facilities offered by large automatic digital computers.

The fission chains and their yields can be obtained from a number of sources. The energy contribution variation with time of each of the fission product nuclides has been calculated by the U.S. Naval Radiological Defense Laboratory and reported by C. F. Miller.⁽¹³⁾ Here the energies, or energy release rates, have been calculated for the basic number of 10,000 fissions occurring all at once. One may take the data for one or any number of these fission products and integrate them to conform to the

power time history of the reactor so that the source strength for the fission products can be determined at any time. This is generally not done in hazard studies, although energy groupings of fission products of interest have been attempted.

In most treatments involving determination of source strength for dispersion problems governed by meteorology, the inventories have been established by considering the total fission product behavior. The energy production rate immediately after shutdown is assumed to be 6 to 8% of the initial power operating level and it is assumed that this rate decreases about like inverse time to the 0.2 power. This general time behavior can also be obtained from direct measurements made on irradiated fuel.

No matter what form or procedure is used or assumptions are made, or what detail is treated, it is an advantage to start with basic data on which much of the labor has been done. Fortunately during the last year an enormous computation and compilation job has been accomplished and reported in the form of several thousand pages of tables and graphs. Because of the tremendous importance of this work, and because of the overconservatism which has been involved in the past by treating fission products as a whole, this work, by J. O. Blomeke and Mary F. Todd,⁽¹⁴⁾ is reviewed here in detail, principally as the authors' summary. [The references are the authors' and have not been checked. - Editor]

Estimates of fission-product levels normally are based either on calculations utilizing the fission yields, half lives, and genetic relations of the various species involved or on physical measurements of fission-product mixtures.

Fortunately, there are in the literature a large number of calculations and experimental determinations which frequently are of great aid in predicting those properties of fission-product mixtures which are of most interest. The curves of Borst and Wheeler,⁽¹⁵⁾ based on a series of cyclotron irradiations of uranyl nitrate salt, were among the first to be used for prediction of the total heat evolved. The mathematical relations of Way and Wigner⁽¹⁶⁾ derived from statistical consideration of the fission process, and the curves of Untermeyer and Weills,⁽¹⁷⁾ based on calorimetric measurements of irradiated uranium, are more recent studies of fission product heat evolution. Very extensive calculations of the individual fission products have been performed by Faller, Chapman, and West;⁽¹⁸⁾ by Lock;⁽¹⁹⁾ and by Hunter and Ballou.⁽²⁰⁾ Fission-product gamma decay levels, divided into seven energy groups suitable for shielding design, have been calculated by Moteff⁽²¹⁾ and by Clark.⁽²²⁾ The thermal neutron poisoning by fission products has been computed by Sampson et al.,⁽²³⁾ and by Walker.⁽²⁴⁾

The work resulting in ORNL 2127 represents an extension of the above calculations, both in detail and in the range of parameters considered. Use of the Oracle computer has permitted calculations for virtually all known fission products, while at the same time taking into account the effect of neutron capture within the fission-product spectrum and the independent yields where known. In so far as possible, the most current and reliable nuclear data have been used to compute the number of atoms of each nuclide present, its activity, gamma power, total power, and thermal neutron poisoning. The gamma disintegrations have been classified into four groups of specified energy ranges. These properties have been totaled for each fission-product chain, for the isotopes of each element appearing as a fission product, for the rare gases, and for the rare earths. Finally, totals of these properties have been made over the entire fission-product spectrum.

The calculations assumed continuous replenishment of U^{235} during fission, constant reactor operating power, and no fission-product removal during irradiation. They were made for six values of thermal neutron flux ranging from 10^{12} to 10^{15} neutrons/cm²/sec, for seven values of irradiation times from 10^5 to 10^8 sec, and for seven values of decay time from 10^3 to 10^9 sec.

The results are presented in two parts. Part I contains curves expressing the atomic levels of the individual nuclides and the gross totals of the properties mentioned above. Also included in Part I are the nuclear data used in the calculations, a discussion of the calculations, and an explanation of how to interpret and use the curves.

Part II contains the results of the summations made on a "per chain" and a "per element" basis and for the inert-gas and rare-earth groups. These results appear in the form of tables from which graphs can be constructed for purposes of interpolation.

The calculations are based on fitting each fission product chain to the generalized decay scheme proposed by Stoughton and Halperin⁽²⁵⁾ (Fig. 3-1). In this diagram the positions of the possible members of chain of mass number A are denoted by the numbers 6 through 11. Although not every decay chain fits this scheme perfectly, the few exceptions can be made to fit with the proper improvisations. Production by neutron capture is taken into account by consideration of the previous chain. To calculate the buildup of members of the A chain (chain of mass number A), there is a total of 11 possible nuclides for which expressions are needed, as follows:

$$\frac{dN_1}{d\tau} = \Sigma_f \phi Y_1 - N_1 \lambda_1 - N_1 \sigma_1 \phi \quad (1)$$

$$\frac{dN_2}{d\tau} = \Sigma_f \phi Y_2 + N_1 \lambda_1 - N_2 \lambda_2 - N_2 \sigma_2 \phi \quad (2)$$

Fig. 3-1. Generalized Fission-Product Decay Scheme.(14)

$$\frac{dN_3}{d\tau} = \Sigma_f \phi Y_3 + N_2 \lambda_2 - N_3 \lambda_3 - N_3 (\sigma_3 + \sigma_{3'}) \phi \quad (3)$$

$$\frac{dN_4}{d\tau} = \Sigma_f \phi Y_4 + N_3 \lambda_3 - N_4 \lambda_4 - N_4 \sigma_4 \phi \quad (4)$$

$$\frac{dN_5}{d\tau} = \Sigma_f \phi Y_5 + N_4 \lambda_4 - N_5 \sigma_5 \phi \quad (5)$$

$$\frac{dN_6}{d\tau} = \Sigma_f \phi Y_6 + N_1 \sigma_1 \phi - N_6 \lambda_6 - N_6 \sigma_6 \phi \quad (6)$$

$$\frac{dN_7}{d\tau} = \Sigma_f \phi Y_7 + N_2 \sigma_2 \phi + N_6 \lambda_6 - N_7 \lambda_7 - N_7 \sigma_7 \phi \quad (7)$$

$$\frac{dN_8}{d\tau} = \Sigma_f \phi Y_8 + N_3 \sigma_{3'} \phi + \alpha N_7 \lambda_7 - N_8 \lambda_8 - N_8 \sigma_8 \phi \quad (8)$$

$$\frac{dN_9}{d\tau} = \Sigma_f \phi Y_9 + N_3 \sigma_3 \phi + (1 - \alpha) N_7 \lambda_7 + (1 - \beta) N_8 \lambda_8 - N_9 \lambda_9 - N_9 \sigma_9 \phi \quad (9)$$

$$\frac{dN_{10}}{d\tau} = \Sigma_f \phi Y_{10} + N_4 \sigma_4 \phi + \beta N_8 \lambda_8 + N_9 \lambda_9 - N_{10} \lambda_{10} - N_{10} \sigma_{10} \phi \quad (10)$$

$$\frac{dN_{11}}{d\tau} = \Sigma_f \phi Y_{11} + N_5 \sigma_5 \phi + N_{10} \lambda_{10} - N_{11} \sigma_{11} \phi \quad (11)$$

In the above equations the numerical subscripts refer to the respective nuclides as denoted in Fig. 3-1. The value N represents the number of atoms present at irradiation time, τ , and $dN/d\tau$ is the differential change in N with respect to time. Σ_f represents the macroscopic fission cross section, or the product of the number of atoms of U^{235} present and the microscopic fission cross section; Y , the independent fission yield; ϕ , the thermal neutron flux; λ , the decay constant, $(\ln 2)/\text{half life}$; σ , the thermal-neutron capture cross section; and α and β , the branching ratios.

Equations 1 through 11 were solved analytically, considering Σ_f and ϕ as constants, to give N_1 through N_{11} as functions of τ .

To obtain solutions of N as functions of decay time, τ , those terms containing ϕ in Eqs. 6 through 11 were dropped, and the resulting expressions were solved analytically for N_6 through N_{11} .

The two groups of solutions were coded for Oracle operations and numerical values of N for every isotope were computed for the following values of parameters:

Thermal Neutron Flux $\phi(\text{neutrons}/\text{cm}^2/\text{sec})$	Irradiation Time, $\tau(\text{sec})$	Cooling Time, $\tau(\text{sec})$
10^{12}	10^5	10^3
10^{13}	3×10^5	10^4
3×10^{13}	10^6	10^5
10^{14}	3×10^6	10^6
3×10^{14}	10^7	10^7
10^{15}	3×10^7	10^8
	10^8	10^9

From the values of N so calculated, the various functions of N (including activity, power, and poisoning) were computed, stored on magnetic tape, and eventually summed in the several groups desired.

A compilation of the nuclear properties of U^{235} fission products made primarily for use in these calculations has been published previously.⁽²⁶⁾ The data were current in January 1954 and represented an accumulation of what was believed to be the best single values available at that time. The data, as Tables, are republished in the report with those changes made prior to starting the calculations.

Since nuclear data are very well organized and abstracted in the literature, the information was collected from a relatively few major sources. The decay chains were taken chiefly from Coryell and Sugarman;⁽²⁷⁾ half-life values and decay schemes were taken from Hollander, Perlman, and Seaborg⁽²⁸⁾ and from the nuclear data collection of W. H. Sullivan of ORNL. Thermal-neutron capture cross sections were taken from the compilation of Hughes and Harvey,⁽²⁹⁾ and the majority of fission yields were values recommended by Glendenin and Steinberg.⁽³⁰⁾

The fission-product decay chains from mass number 72 through mass number 161 are given. Besides containing all nuclides for which evidence of formation in fission exists, this table includes certain products which could be formed indirectly as a result of η - γ reactions.

A summary of nuclear properties, arranged in increasing order of atomic number and mass of the nuclides is also provided. In addition to the cumulative and chain yields, independent yields are given where known. For shielding considerations, the gamma transitions were classified in four groups, depending on the energy of the transitions: 0 to 0.25 Mev; 0.26 to 1.00 Mev; 1.01 to 1.70 Mev; 1.71 Mev and higher.

There is also a separate listing of nuclides which may be expected to be present only as products of neutron capture from the preceding chain. This table is not complete, in that only those nuclides are included which can be calculated from known values of thermal-neutron absorption cross sections.

The results given in Part I are the N values for the individual fission-product nuclides and the grand totals of those functions of N which were computed. The results, all in graphical form, are presented in five groups, (A, B, C, D, and T, in Appendix B)⁽¹⁴⁾ and are preceded by a table which serves as an index to the figures. In this table each isotope for which results are presented is listed under its chain, or mass number. To facilitate and condense the presentation, the N values for the radioactive nuclides have been normalized as described below.

Group A: The curves give the saturation values of each radioactive isotope as a function of thermal neutron flux. These saturation values were used to normalize the N values during their buildup (irradiation time) in the reactor. Saturation values of stable isotopes were not computed, for there is no advantage to be gained from normalizing the buildup of the stable species.

Group B: Values of N_T , the number of atoms of fission-product nuclides present as a function of irradiation time, are presented as curves. By normalizing these numbers with the saturation values, N_S , from the curves of group A, the buildup of an isotope can frequently be represented by a single curve, independent of flux, expressing the fraction of saturation of the isotope, N_T/N_S , at various irradiation times. The product of this number and the saturation value of the isotope at the flux in question, N_S/N_{25}^0 (where N_{25}^0 is the number of U^{235} atoms initially), yields the number of atoms present at irradiation time, τ , per initial atom U^{235} , N_T/N_{25}^0 .

Group C: Curves give N_t , the number of atoms of the individual radioactive nuclides present as a function of decay time, or time following reactor shutdown. These numbers have been normalized and are presented as fraction-of-shutdown values. Normalization in this way made it possible in many cases to represent the values for all fluxes and irradiation times by a single curve. The number of atoms present at decay time, t , per initial atom U^{235} , N_t/N_{25}^0 , is found as the product of N_t/N_T , N_T/N_S , and N_S/N_{25}^0 .

Group D: Compress curves showing the buildup of the stable isotopes during irradiation. These numbers are not normalized.

Group T: The curves in this group present the gross totals of the various properties of the fission product spectrum as a function of ϕ , τ , and t .

Figures are also given for the following: Total fission-product activity, expressed in curies per initial atom of U^{235} ; total fission-product gamma power, expressed in watts per initial atom of U^{235} ; total fission-product power, expressed in watts per initial atom of U^{235} ; total fission-product poisoning, expressed in barns per initial atom of U^{235} ; total fission-product gamma disintegrations per second per initial atom of U^{235} of energy ≤ 0.25 Mev; total fission product gamma disintegrations per second in the

energy range 0.26 to 1.00 Mev; total fission-product gamma disintegrations per second in the energy range 1.01 to 1.70 Mev; and total fission-product gamma disintegrations per second with energy greater than 1.70 Mev.

Further results of the fission-product calculations described in Part 1 are presented in Part 2. Properties of individual fission products were summed for each decay chain (mass number), each element (atomic number), the rare gases, and the rare earths. The results, in tabular form, are given for the same values of thermal neutron flux, reactor operating time, and decay time as in Part 1.

In Section I of Part 2, the following properties, per initial atom of U^{235} , are tabulated for each fission-product chain:

1. activity, curies
2. gamma power, watts,
3. total power, watts,
4. poisoning, cm^2 ,
5. gamma disintegrations per second, with energy equal to or less than 0.25 Mev,
6. gamma disintegrations per second, in the energy range 0.26 to 1.00 Mev,
7. gamma disintegrations per second, in the energy range 1.01 to 1.70 Mev,
8. gamma disintegrations per second, with energy greater than 1.70 Mev.

Section II contains the properties listed above summed for the isotopes of each element addition, the total number of atoms of each element, $\Sigma N/N_{25}^0$, is presented.

Section III contains the same properties as in Section II, summed for the rare gases krypton and xenon.

Section IV contains the same properties as in Sections II and III, summed for the rare-earth elements lanthanum through dysprosium. Because of its chemical similarity to the rare earth, yttrium is included in the totals of this section.

The results as presented are reproductions of the computer output. The individual numbers are eight-digit floating point numbers. For example, the number +469338-08 is equivalent to 0.469338×10^{-8} . The limit of reliability of the numbers is about 1 part in 100 in the best cases.

The large accumulation of data in ORNL-2127 has been outlined above in some detail for the convenience of the reader. In addition to this, a concise treatment has been rendered by J. R. Stehn and E. F. Clancy⁽³¹⁾ which may be used very handily in preliminary estimates (and final ones in some cases) of the decay power and activity of all the fission products following bursts and also after infinite irradiation periods. The gamma activities are shown for six energy groups; 0 - 0.2, 0.2 - 0.4, 0.4 - 0.6, 0.6 - 1.0, 1.0 - 2.0, >2 Mev.

Shutdown time is extended from 0.1 sec to 10^4 years for total decay power and from 1 day to 10^3 days for the beta and gamma powers. In the case of the burst, power values are given in watts/megawatt day. For the infinite irradiation time they are given in watts/megawatt. The activity is in curies/megawatt day for the burst and curies/megawatt for the long irradiation. The curves are reproduced here in Figs. 3-2 through 3-7. These curves are independent of power level and may be used for any length of operating time. This is accomplished on the basis of a theorem stated in the paper that: "The intensity of fission-product radiations following a finite period of operation is simply the difference between two values on the infinite irradiation curve" (Fig. 3-7). The values are those for two particular cooling times:

- (a) the time elapsed from the end of the exposure period to the moment in question (the time normally considered to be the "cooling time" following a run), and
- (b) the time elapsed from the beginning of the exposure period to the moment in question (the sum of the time of exposure and the cooling time).

If the cooling time is longer than the exposure period, the two particular times just defined are relatively close together, and it is difficult to read accurately the difference in the corresponding ordinates. It is just such cases that invite the use of the "burst" curve. Whenever the cooling time is much longer than the exposure period, it is possible to treat the exposure as a "burst" that occurred at the midpoint of the exposure period. Even if the cooling time is twice the exposure period, the error involved in treating the exposure as a burst is 2% if the exposure is to a constant neutron flux, and the error cannot exceed 35% no matter how the flux varies during exposure. (These percentages apply, strictly speaking, only to the $t^{-1.2}$ portions of the curve).

B. Other Materials

After determining the inventory of fission products, one must estimate the activation of materials of the reactor which may participate in providing the total radioactive load on the barrier. The two forms of

FIG. 3-2
BETA AND GAMMA POWERS FOLLOWING BURST
(REF. 31)

FIG. 3-3
BETA AND GAMMA POWERS AFTER INFINITE IRRADIATION
(REF. 31)

FIG. 3-4
ACTIVITIES FOLLOWING BURST (REF. 31)

FIG. 3-5
ACTIVITIES AFTER INFINITE IRRADIATION
(REF. 31)

FIG. 3-6
DECAY POWER FOLLOWING BURST (REF. 31)

FIG. 3-7
DECAY POWER AFTER INFINITE IRRADIATION (REF. 31)

principal interest are coolant activation and conversion of fertile material to fissionable materials. The first includes not only the primary coolant for the core but air or other coolant used for shield cooling. The second includes the plutonium production chain and the uranium-233 production chain. The calculation of these burdens is generally straight-forward, since the reaction cross-sections, fluxes, and decay schemes are well known. R. O. Brittan⁽³²⁾ has treated these calculations and pertinent portions are reproduced here since the original treatment is not generally available:

The high cross-section fertile materials of interest which change to fissionable isotopes by β decay (if fission does not take place) are Th²³², U²³⁸, and Pu²⁴⁰. These, along with the original fuel then become α -emitters. The fuel element structural material, although usually of low cross section, also becomes active as a result of neutron capture. Of the coolants, water itself does not become very active directly because neutron capture in the hydrogen produces deuterium which is stable, and the oxygen has a very low cross section (~ 0.0002 barns) for capture and yields a stable form. The deuterium formed, of course, can now capture neutrons to form tritium but the capture cross section is again small (~ 0.0006 barns). In the case of heavy water, the deuterium forms tritium directly. This tritium gas is normally handled by a recombiner and returned to the coolant. It is always present in a D₂O reactor however, and hence becomes a problem if an accident occurs. Of the liquid metals, Na-K, and Bi must be considered, the former because of the strong association with animal life, and the latter because of production of polonium, the worst of the α -emitters. Organic coolants containing only the very low cross section carbon besides the nuclides present in water can be treated to the same degree as water.

The concentration for any source from activation is given by:

$$C_{Ti} = \bar{\phi} \sum a_i V_i \frac{(1 - e^{-\lambda_i T})}{\lambda_i},$$

where

C_{Ti} = concentration of i th isotope at end of time T

V_i = volume of material of i th isotope

$\bar{\phi}$ = average neutron flux

λ_i = decay constant of i th isotope

T = irradiation time

$\sum a_i$ = macroscopic absorption cross section of i th isotope

(Removal by further neutron capture after initial activation is a secondary effect which can nearly always be neglected. If necessary, the term of the secondary production can be coupled with the primary expression as a removal term.)

t seconds after removal from the irradiation field, the concentration will be

$$C_{t_i} = C_{T_i} e^{-\lambda_i t}$$

The activity in disintegrations per second is λC , or

$$A_{T_i} = \bar{\phi} \Sigma a_i V_i (1 - e^{-\lambda_i T})$$

and

$$A_{t_i} = \lambda_i C_{t_i}$$

These expressions are most conveniently put in terms of reactor operating power by relating them to the U^{235} fission process energy release in the same flux:

$$C_{T_i} = F_i \frac{P}{K_f} \left[\frac{(1 - e^{-\lambda_i T})}{\lambda_i} \right], \quad C_{t_i} = C_{T_i} e^{-\lambda_i t}$$

where

$$F_i = 10^{10} \frac{\sigma_{a_i} f_i}{A_i}$$

σ_{a_i} = microscopic cross section of isotope of material to be irradiated

A_i = atomic mass weight of isotope

f_i = isotopic abundance fraction

P = operating power, kilowatts

K_f = amount of U^{235} in reactor, kilograms

C_{T_i} and C_{t_i} here are nuclei/gram of original natural element

(Care must be taken to include a ratio of fluxes if the material being irradiated does not see the same average flux as the fuel does. This factor is not included in these expressions but would be made a part of F_i .)

The values of λ_i and F_i are given for normal structural material constituents and for coolant and air constituents in Table 3.1. The list is arranged in ascending order of λ_i . For long irradiation times the C_t values are given to 10% accuracy by:

$$C_t = F_i \left(\frac{P}{K_f} \right) T e^{-\lambda_i t} \quad \text{for the first 8 isotopes listed} \quad (I)$$

and

$$C_t = F_i \left(\frac{P}{K_f \lambda_i} \right) e^{-\lambda_i t} \quad \text{for the others} \quad (\text{II})$$

(Note that frequently $e^{-\lambda_i t}$ can be assumed = 1, i.e., for $\lambda_i t < 0.01$)

The limiting values of T to yield 1% accuracy are listed for each approximation. T must be smaller than T_I to use (I) and larger than T_{II} to use (II) above.

Table 3.1

	Original Isotope (and A_i)	f_i Fractional Isotopic Abundance	σ_{a_i} Activation Cross Section	Radio*** Nuclide Formed	λ_i Decay Constant 1/seconds	$\frac{F_i}{A_i}$ $10^{10} \sigma_{a_i} f_i$	T_I	T_{II}
1	Zr ⁹²	0.171	0.2	Zr ⁹³ β S	2.45×10^{-14}	3.72×10^6	25,900 years	
2	Ni ⁵⁸	0.68	4.3	Ni ⁵⁹ KS	2.77×10^{-13}	5.04×10^8	2,300 years	
3	C ¹³	0.0111	0.0009	C ¹⁴ β S	3.94×10^{-12}	4.68×10^3	161 years	
4	A ³⁸	0.00063	0.8	A ³⁹ β S	8.45×10^{-11}	1.33×10^5	7.5 years	
5	Ni ⁶²	0.037	15	Ni ⁶³ β S	2.77×10^{-10}	8.95×10^7	838 days	
6	H ²	0.00015	0.00057	H ³ β S	1.79×10^{-9}	4.28×10^2	130 days	
7*	Co ⁵⁹	0.514	19	Co ⁶⁰ β S	4.22×10^{-9}	1.66×10^9	55 days	
8	Fe ⁵⁴	0.059	2.2	Fe ⁵⁵ KS	7.56×10^{-9}	2.40×10^7	31 days	
9	Zr ⁹⁴	0.174	0.1	Zr ^{95**} Nb ⁹⁵	1.23×10^{-7}	1.85×10^6	45 hours	430 days
10	Fe ⁵⁸	0.0033	0.9	Fe ⁵⁹ β S	1.73×10^{-7}	5.12×10^5	32 hours	300 days
11	A ³⁶	0.00337	6	A ³⁷ KS	2.29×10^{-7}	5.62×10^6	24 hours	230 days
12	Zr ⁹⁶	0.028	0.1	Zr ⁹⁷ Nb ⁹⁷	1.13×10^{-6}	2.92×10^6	30 minutes	110 hours
13	Na ²³	1.000	0.53	Na ²⁴ β S	1.28×10^{-5}	2.30×10^8	24 minutes	97 hours
14	K ⁴¹	0.068	1.1	K ⁴² β S	1.54×10^{-5}	1.83×10^7		81 hours
15	Ni ⁶⁴	0.010	2	Ni ⁶⁵ β S	7.48×10^{-5}	3.12×10^6		17 hours
16	A ⁴⁰	0.996	0.53	A ⁴¹ β S	1.06×10^{-4}	1.32×10^8		12 hours
17*	Co ⁵⁹	0.486	18	Co ⁶⁰ β S	1.10×10^{-3}	1.48×10^4		70 minutes
18	Al ²⁷	1.000	0.23	Al ²⁸ β S	5.02×10^{-3}	8.52×10^7		15 minutes
19	O ¹⁸	0.00204	0.00022	O ¹⁹ β S	2.39×10^{-2}	2.49×10^2		190 seconds
20	N ¹⁵	0.0037	0.00002	N ¹⁶ β S	9.35×10^{-2}	4.93×10^1		49 seconds
21	B ¹¹	0.812	0.05	B ¹² β S	27.7	3.69×10^7		0.17
			0.19	RaE Po ²¹⁰	1.6×10^{-7}	9.10×10^6	35 hours	330 days
22	Bi ²⁰⁹	1.000	< 0.15	Bi ²¹⁰ Po ²¹⁰	8.5×10^{-15}	7.18×10^6	74,600 years	

*Isomers

**Daughter beta or alpha-decay to stable granddaughters

***Indicates beta-decay, S indicates stable nucleus formed directly

Abundance of daughter isotope is C_{tD}

$$C_{tD} = C_t \frac{(1 - e^{-\lambda_D T})}{\lambda_D} e^{-\lambda_D T}$$

As can be seen, most of the λ 's are small and the inventory of radio nuclides is proportional to the integrated power of the reactor or power level times radiation time. Also, after activation the inventory stays nearly constant for a long time.

In the case of large λ (short half-lived material) the inventory saturates, and is proportional to the normal operating power. For fission products, the situation is much the same, except many generations sometimes grow up and die before stability is achieved. This makes the general quantitative analysis difficult. In general, a gram of fission products

results from a gram of fuel fissioned. Another generality is that the energy production rate immediately after shutdown is about 6% to 8% of the initial power operating level, and this rate decreases slowly - approximately as inverse time to the 0.2 power.

For 6 months operation at a 500 megawatt power level, the power level of the fission products, after leaving the reactor, decreases as follows: (The power for the volatile fission products are given also.)

Time	Power, Kilowatts	
	All Fission Products	Volatile Fission Products
t = 0	32,000	-
t = 1 min	15,000	2500
t = 1 hr	7,000	1300
t = 1 day	3,000	200
t = 1 month	750	77
t = 6 months	200	16

It is expected that in reactors containing U^{238} , some plutonium will be produced. The greatest amount of plutonium is produced in a converter, where it is nearly equal to the fuel fissioned. Thus after 1% burn-up, the amount of Pu present will be about 0.01 K_f or 1%

of the amount of fuel. The Pu is an energetic alpha-emitter with a half-life of 24,300 years or $\lambda = 9.03 \times 10^{-13} \text{ sec}^{-1}$, and alpha energy of about 5.2 Mev.

The magnitude of radioactive load to be expected even for minor accidents of the type normally expected during operations is shown by the following: it is normally anticipated that there will be fuel element failures and loss of primary containment during reactor operation. The failure is any loss of integrity which results in the escape of fission products or fuel from the element. A small failure results in rapid release of the gaseous and volatile fission products, with possibility of slow leaching of solid fission product coolant. A gross failure results in disintegration of most of the elements with release of both solids, and gases. The magnitude of release from an element is a function of the fraction (total fission products)/(No. of fuel elements). The number of basic fuel elements varies in reactors from about 200 to about 100,000. Hence for the two types of failure one could conceivably expect the following activity per fuel element failure for the 500 megawatt reactor.

Time After Failure	Gross Failure				Small Failure			
	kw	Curies	Watts	Curies	Watts	Curies	Watts	Curies
	(200 elements)	(200 elements)	(100,000 elements)	(100,000 elements)	(200 elements)	(200 elements)	(100,000 elements)	(100,000 elements)
0	160	5.4×10^7	320	1.1×10^5	12,500	4.2×10^6	25	8400
1 min	75	2.5×10^7	150	5×10^4	6,500	2.2×10^6	13	4400
1 hr	35	1.2×10^7	70	2.4×10^4	1,000	3.4×10^5	2	670
1 day	15	5×10^6	30	10^4	380	1.3×10^5	0.8	270
1 month	4	1.3×10^6	8	2700	80	2.7×10^4	0.2	67
6 months	1	3.4×10^5	2	670				

One megawatt of fission product power is equivalent to 3.4×10^8 curies of activity. This is only approximate, since curies are based on disintegrations per unit time and the energy available in disintegration is ignored. The average energy per disintegration varies with time after shutdown.

3.1.2.2 Radioactive Material Release

Many assumptions have been used to determine the fractional release of fission products from the reactor system to the region bounded by the outer barrier. The best experimental support is given by Parker and Creek.⁽³³⁾ On the bases of these findings, J. W. Healy⁽³⁴⁾ has proposed upper limits for a generalized release pattern. He suggests that from the portion of the reactor affected by the accident one considers that it is possible to effect the release of:

- 100% of noble gases
- 70% of halogens (cesium and tellurium, tentatively)
- 10% of remaining mixture.

In preparing data for the AEC report to Congress,⁽⁵⁾ inventories were calculated for a 500 MW reactor 24 hours after shutdown from 180 days operation as tabulated:

Item	Inventory Activity Curies	Composition of Released Inventories, Per Cent	
		All Species Released	Only Volatiles Released
Total fission products	4.1×10^8		
Strontium-90	3.8×10^5	0.093	0.0045
Strontium-89	1.7×10^7	4.1	
Cerium-144	8×10^6	2.0	
Plutonium-239	3.8×10^3	0.001	
Iodines	5×10^7	12.2	59.5
Noble gases	3.4×10^7	8.3	40.5
Total volatile fission products	8.4×10^7	20.5%	
Total unspecified	30.9×10^7	75	

3.1.2.3 Radiation

After determining the radioactive load, one can divide the radiation problem (see Fig. 1-2) into two parts and deal first with the direct radiation of gammas from the container with appropriate attenuations due to time, mass and distance. The other problem becomes one of studying the leakage of these fission products from the containment vessel. The dosage to people in, and contamination of, the region can be determined for both cases. If the total does not exceed the maximum emergency dose recommendations past the edge of the area under control, the size of the site is proper.

Consider first the doses from direct radiation from the source. Specifically, we are concerned with the radiation dose to human beings in the gamma field at various distances from the reactor container after the accident. The energy release rate for the gammas is known at various times after shutdown so that the source strength is known in terms of watts or ergs per second. The initial rate is attenuated by the time decay characteristic of total fission products. On top of this, a correction is applied for absorption by the air between the container and the person. Further, a geometric attenuation correction is applied so as to finally obtain an energy rate value for the radiation field at the receptor. Thus, a dose rate can be established in roentgens per second. If this dose rate is integrated for a given exposure time, the total dose in roentgens can be obtained.

This procedure is carried out in an unpublished paper by Fitzgerald and Flynn.⁽³⁵⁾ The formulae developed assume that all fission products are uniformly dispersed in the container, and that they decay according to time to the -0.2 power. Hence they cannot be used where only volatile fractions or other selective releases are assumed. The analysis was based on a range of photon energies, since average values would be greatly in error in considering attenuation by the steel shell. Ground scatter contributions were neglected. Past 100 meters, the dose rate 10 seconds after the incident is given as $Dr_{(10)} = k_i X_0^{-10/3}$. Here k_i is a factor accounting for structure size and steel shell thickness and is given approximately for thicknesses 2" by $\log_{10} k_i = 9.1 - 0.46 \theta$, where θ is steel thickness in inches. The value at any time, t , after shutdown is obtained by multiplying by $(0.1t)^{-0.2}$. X_0 is distance from source in meters, and $Dr_{(10)}$ is dose rate in roentgen/hour/megawatt of operating power level 10 seconds after shutdown from infinite operation. The total dose recorded during the time t (D_t) is $5.5 \times 10^{-4} Dr_{(10)} Pt^{0.8}$ roentgens, where P is power level in megawatts.

A more extensive treatment is given by L. Geller and R. Epstein⁽³⁶⁾ who examined containment vessel sources to determine gamma radiation fields with shielding, with partial shielding, and with no shielding (except the shell). Both direct and scattered radiation is accounted for. The solution of the exact problem is quite beyond any analytical or computational technique presently available. Some simplifications or idealizations are called for which lead to a more readily solvable problem, but which do not sacrifice too much as far as the physical validity of the results are concerned. To this end the following assumptions have been made:

1. The gamma emitting complex composed of containment vessel and external shielding may be replaced by a suitably located anisotropic point source. The angular distribution of radiation from this point source is chosen so that it agrees at large distances with that actually emerging from the complex. The radiation from this equivalent source either proceeds directly to the dosage point of interest (the direct component) or else it gets to that point via scattering in the air (sky-shine).

2. No scattering with the air takes place inside of some suitably defined exclusion volume surrounding the point source.

3. The dose rate at a point is composed of the direct portion which has suffered no scattering collisions in the air external to the exclusion volume and the scattered portion obtained by computing the first scattered flux and then increasing it by an appropriate buildup factor.

Without entering into a detailed discussion of the implications and validity of these assumptions it may be said that items 1 and 3 are introduced to properly assess the dosage at large distances from the containment vessel, whereas item 2 should assure reasonable results within its vicinity. In applying the method three steps are involved:

1. the true sources within the containment vessel need to be specified;
2. the equivalent anisotropic point source for each energy must be found;
3. the dose rates at the point of interest for the anisotropic sources have to be determined.

The principal features of the analytical methods developed are: first, using them one can calculate both direct and scattered gamma radiation fields around a non-uniformly shielded containment vessel in terms of exposure; second, they can be applied to any combination of radioactive source energy groups. The methods employed give agreement between calculation and experiment (of those checked) to within 25%.

An example given assumes that a 500 megawatt reactor has been operating for 600 hours when 25% of all the halogens and inert gases, and 1/16 of all other fission products are released to the space bounded by the container. Dose-distance relationships are obtained for a spherical container half-buried, with no other shielding, with cylindrical shielding only, and with complete shielding. The results for four calculations are shown in Figs. 3-8 to 3-11. Comparisons show, for example, that at 1400 ft from the source the dose after one hour will have been:

Concrete Shielding Thickness - Feet		Dose in 1 Hour
Cyl wall	Roof	
0	0	17.0 roentgens
4	0	8.5 roentgens
4	1.55	6.7 milli-roentgens
4	2.15	1.2 milli-roentgens

FIG. 3-8
DOSAGE VS. DISTANCE 1 HOUR AFTER INCIDENT FOR UNSHIELDED SPHERE
(REF. 36)

DOSAGE - ROENTGENS FOR CURVE A: MILLIROENTGENS FOR CURVE B

FIG. 3-9
DOSAGE VS. DISTANCE FOR SHADOW SHIELDED SPHERE

DIRECT AND SCATTERED DOSAGE RECEIVED DURING THE FIRST HOUR AFTER AN INCIDENT, AS A FUNCTION OF DISTANCE FROM THE CONTAINMENT CENTER, WHEN THE FULL HEIGHT SHADOW SHIELD IS EMPLOYED
(REF. 36)

FIG. 3-11

DOSAGE VS. DISTANCE FROM CENTER OF CORE COMPUTED FOR
1 HR. AFTER INCIDENT FOR FULLY SHIELDED SPHERE: CASE 2 (REF. 36)
FOR CASE 2 THE ROOF THICKNESS WAS CHOSEN TO GIVE A SCATTERED DOSAGE OF 500
MILLIROENTGENS IN THE FIRST HOUR

Here the striking relative importance of direct versus sky-shine scattered radiation is made clear as well as the large gain to be realized by complete shielding as opposed to the small gain due to use of shadow shields.

The direct radiation problem has also been treated in an AEC report to Congress.⁽⁵⁾ Here the special case of 500 MW reactor is treated, using both the $t^{-0.2}$ decay law and a more exact seven energy group analysis from NDA-27-39 for source strengths. The attenuation for one inch of steel was assumed, although self-absorption in the shell was neglected. In both cases air and geometry attenuation were accounted for by direct application of the $K = B(\mu r)e^{-\mu r}/4\pi r^2$ where $B(\mu r)$ is the build-up factor, r is distance from source, and μr equals the total number of mean free paths traversed. It is pointed out that use of a single mean energy in the fission products can underestimate the actual levels by as much as a factor of 10 at distances of about 20 mean free paths. Hence energy de-

pendent values of K were applied to each of seven energy groups before determining totals. The results are shown in Fig. 3-12. Values for other power levels can be obtained by multiplying the ordinates by the power level ratio.

FIG. 3-12
INTEGRATED γ DOSE AS A FUNCTION
OF EXPOSURE TIME AND DISTANCE (REF. 5)

The approaches used above can be extended to account for going further down into the ground, and for certain topographical features which may be available. Some caution must be used here, because a reasonable fraction of the radiation dose received by a person at a distance from the source is a result of sky shine which means essentially a reflection of fission product energy from the air above the reactor container.

It should be pointed out again here that most of the information said to be lacking in these treatments is available, and that unnecessary discrepancies can be avoided by individual consideration of fission products and use of automatic digital computers. The assumptions made are affected principally by the inadequacies of grouping in energy when employing build-up factors for determining air attenuation.

Next a case of leakage of fission products from the containment shell must be considered. For this case the radioactivity is assumed to be in a plume extending downwind from the reactor. The strength of the source feeding the plume is determined by the concentration of fission products in the containment shell and the volume leak rate for the containment shell. The concentration at any point in the plume is

generally calculated by considering the numerous influences of the meteorological parameters on meteorological diffusion. This problem originated in attempting to study the dispersal of stack gases and in studies for Chemical Warfare Services. The generalized problem is discussed by White and Pack of the U.S. Weather Bureau.⁽³⁷⁾ Their discussion provides an excellent background for studies of the relation of meteorological diffusion to the problem of transport of radioactive material.

The development and formulation of the mathematical statement of the diffusion problem which must be solved is due to O. G. Sutton of England.⁽³⁸⁾ The basic equation developed therein is for a continuous point source, exactly the model we have for leakage from a container. The validity of Sutton's equation in lapse or zero gradient conditions with appreciable wind is fairly well established experimentally, and its use enables reasonably accurate predictions to be made of the dosage from a given release for ranges up to about 6 miles. Under inversion conditions, however, the diffusion depends on local factors and is not generally predictable. In the latter case, the local factors which are necessary for prediction would have to be filled by experimentation at each reactor site.

S. E. Singer and D. A. Baker⁽³⁹⁾ have prepared a detailed analysis of the problem of considering atmospheric diffusion, including review of previous applications of methods available, assessment of ranges of applicability of methods, and effects of assumptions made. The unclassified information of interest has been excerpted and is presented here for the convenience of the reader:

Meteorology

The meteorologist considers the atmosphere in the vertical plane in terms of "soundings" which are basically plots of altitude against temperature. If the decrease in air temperature with altitude, the lapse rate, is that predicted by the ideal gas law, the lapse rate is said to be adiabatic; and it can be shown that a parcel of dry air moved vertically within such an atmosphere is neither aided nor hindered in its upward or downward movement by the surrounding air. A lapse rate which gives a greater decrease of temperature with altitude than adiabatic is termed superadiabatic or steep, and vertical movement of a parcel of air within such an environment is aided actively by the environment. A lapse rate less than adiabatic implies that the atmosphere will resist any tendency for a parcel of air to move up or down within it. Lapse rates less than adiabatic can range through the isothermal case to an actual inversion, i.e., increase of temperature with altitude. [If the lapse rate is large enough, a critical lapse rate can be reached where the air will spontaneously overturn - Editor]

The atmosphere, or a given vertical portion thereof, is said to be stable or unstable depending upon the degree and the direction of the departure of the lapse rate from the adiabatic. In a stable atmosphere a contaminant, such as that resulting from fission product release, will diffuse slowly in the vertical, whereas it will diffuse rapidly in an unstable atmosphere.

A real atmosphere will rarely be characterized for long or in great depth by a constant lapse rate. As a result of both dynamic and diurnal variations, it is common to encounter a combination of lapse rates over any reasonable atmospheric depth. A common combination might give the overall picture shown in Fig. 3-13.

FIG. 3-13
COMBINATION OF LAPSE RATES (REF.39)

As a more illuminating example of the effects of lapse rate on vertical diffusion, consider the release of a contaminant at any point within the region AB. Since the lapse rate within this section of the atmosphere is superadiabatic (unstable), diffusion would be relatively rapid within this section. However, diffusion upward beyond point B would be hindered markedly. The inversion in this case would establish effectively an upper boundary to the overall diffusion, and the steep lapse rate would assure the contaminant's reaching the ground rapidly regardless of the original height of release within the AB layer. On the other hand, a release above point C would be prevented from reaching the ground by the very same inversion.

The altitude of release or the height to which a contaminant released on the ground will rise before it reaches thermal equilibrium with the atmosphere is thus of great importance, not only in the special case of inversion limitation just cited, but for other reasons as well. Obviously the higher the initial height of rise, the greater the opportunity for diffusion or dilution of the initial concentration, and the greater also the opportunity for geometric factors to diminish "shine" radiation from a radioactive contaminant.

In a review of the formulas available for computing cloud rise, three such relationships were found in general use or recommended for use in the analysis of reactor hazards problems. The first⁽⁴⁰⁾ is a somewhat simplified form of an equation of Sutton's. The second one given⁽⁴⁰⁾ is Machta's formula, and the third appears to be a modification of Sutton's.⁽⁴¹⁾

A comparison of these formulas indicates that not only are they markedly different, but each contains at least one parameter which is not considered in the other two. This circumstance, together with the implied⁽⁴⁰⁾ uncertainties with regard to cloud rise estimates led to a comparison of all three formulas. Such a comparison, based on an isothermal atmosphere, is given in Fig. 3-14. Because of the lack of common parameters, it was necessary to normalize to an arbitrary heat release of 10^5 calories.

FIG. 3-14
COMPARISON OF THE CLOUD RISE FORMULAS. (REF. 39)

Diffusion

Although a number of diffusion models have been used in reactor hazards analyses, many empirical and even arbitrary, only one model has received serious general attention: the theoretical formulation of Sutton. For the case of a continuous ground point source, his formula for the concentration at a distance along the downwind axis is:

$$\chi(x) = \frac{2Q}{u\pi C_y C_z X^{2-n}}$$

Where

$\chi(x)$ = concentration

u = average wind velocity

C_y = horizontal diffusion coefficient, meters^{n/2}

C_z = vertical diffusion coefficient, meters^{n/2}

n = atmospheric stability parameter

Q = emission rate

The quantity n is determined from the additional expression:

$$\bar{u} = \bar{u}_1 \left(\frac{Z}{Z_1} \right)^{\frac{n}{2-n}}$$

Where $1 > n > 0$

(n is small for steep lapse rates, unstable conditions, and large for inversion conditions). $\bar{u}$ and $\bar{u}_1$ are mean wind speeds at the altitudes defined by Z and Z_1 . C_y and C_z are, in theory, deducible from eddy velocity measurements. In practice, however, C_y and C_z are obtained by solution of the equation for known or assumed values of all the other parameters, and n is usually assumed to take on certain values recommended by Sutton.

Sutton states that this and his allied equations for diffusion over level surfaces, in conditions of small temperature gradient and moderate wind will give satisfactory results up to several kilometers.

Sutton's restriction on distance is a fundamental one which can be shown to follow from the assumptions basic to his diffusion model. Further, and more explicitly, the term u refers to a single and constant wind speed. Common experience indicates the unrealistic nature of such an assumption except over comparatively small distances.

In spite of Sutton's own reservations with respect to the distances over which his theoretical diffusion model is applicable, and with respect also to the stability conditions permissible, the model has been applied to a large number of hazards studies without

regard to these reservations. The nomographic forms into which Sutton's work has been transformed by such workers as Holland and Gifford(40) make disregard of the distance reservation unfortunately easy.

The "selection" or empirical determination of appropriate values for n , C_y and C_z are rather difficult to assess, since it is not practical to determine these quantities independent of n . Holland's work(40) and that of Stewart(42) show relatively good agreement, on the order of 20-30%, between Sutton's recommended values and those found by experiment. It is very disturbing to note, however, that the agreement is best for inversion conditions, those for which Sutton considers his work least valid. Moreover, both Holland and Stewart found a dependence of the diffusion parameters on wind and terrain. While these effects are, meteorologically, quite reasonable, neither is recognized explicitly in Sutton's formula. Gifford(40) notes that the corresponding curves for the earlier data obviously define that Sutton stability parameter, n , only poorly. It shares this characteristic with all other methods of determining this parameter. The n -problem has been met squarely by Stewart et al., who simply assume that $n = 0.25$ in all meteorological conditions. Although this decision must have caused these authors some concern, they should be commended for breaking the bottleneck.

These remarks of Gifford led to a sensitivity analysis to determine the effect of an arbitrary or injudicious selection of n on χ , the concentration at a given distance. The results are shown in Fig. 3-15. The abscissa is the product of concentration, $\chi(x)$, at a given distance and a normalization constant, K , obtained by assuming that all other parameters in Sutton's formula were independent of n .

FIG. 3-15
EFFECT OF VARYING n IN SUTTON DIFFUSION FORMULA. (REF. 39)

Sutton recommends the following values of n for various atmospheric conditions:

<u>Condition</u>	<u>n</u>
Steep lapse rate	0.20
Small temperature gradient	0.25
Moderate inversion	0.33
Large inversion	0.50

It is apparent from Fig. 3-15 that arbitrary assignment of 0.25 for n for the large inversion condition gives an apparent error in concentration of about an order of magnitude. This graph illustrates also the fact that the effect of n on χ increases with downwind distance: at 1 km, changing χ by 0.1 units change χ by a factor of 2; at 100 km, the factor becomes 3.3.

There is one other aspect of Sutton diffusion theory which merits attention. This amounts to the continuing depletion of the diffusing cloud which may arise from fallout or impaction of the contaminant. Consideration of such deposition has been made by Chamberlain, who modifies Sutton's equations to account for this factor. Understandably, taking deposition into account required a knowledge of particle size and mass. Since these data are lacking completely for the aircraft reactor hazards case, Chamberlain's work serves only to define roughly certain limiting cases. For example, one may compute, assuming certain particle physical properties (reasonable or unreasonable), what the maximum deposition level is at a certain distance from the source.

From this review of the techniques available for computing atmospheric diffusion, it is concluded that there is no truly valid means for making such computations for all distances of interest and all possible atmospheric stability conditions. On the basis of general acceptance and very limited experimental confirmation, it appeared reasonable to use Sutton's methods, providing that only steep lapse rates (good diffusion) and short distances were considered. For distances greater than those recommended by Sutton, some modification of the formula to consider terrain and variation of his parameters with distance appears possible.

Modification of Diffusion Theory

In reviewing a number of hazards studies which predicted very large exclusion radii for poor diffusion conditions, the frequent assumption of a very low wind speed which was held constant in direction for tens and even hundreds of miles was particularly disturbing. There are two fundamental objections to such an assumption: a wind which is low in speed is also variable in direction by virtue of the meteorological conditions which produce it. This is true to such a degree that winds below a certain speed 3 knots for Air Weather

Service observation) are not assigned any direction - they are reported only as "light and variable." Secondly, the assumption of a low-speed wind which is constant in direction for very large distances implies a constancy of those factors for long periods of time as well as for long distances. Neither the time nor the distance constancy is compatible with real meteorology - light winds are the result of a flat pressure gradient which affords the best possible opportunity for wind perturbations by stronger meteorological systems, terrain influences, and diurnal changes.

To illustrate this point, an analysis was made of wind observations for Albuquerque, for conditions of low wind speed. Vector diagrams of half-hourly wind observations were plotted in such a manner as to obtain the actual path which would be described by a balloon or a cloud floating with the wind. The ratio of the total distance of travel to the net displacement was then computed. For a constant wind direction, the path is a straight line, and hence the ratio is unity. For a wind varying in direction, the path is tortuous, and the ratio is greater than 1. The results of this work were then plotted as a function of wind speed.

These results illustrate pointedly the variability in wind direction at low wind speeds. At a speed of 4 knots, for example, the total to net displacement ratio is approximately 2. The implication here is that if one assumes a speed of 4 knots for a diffusion calculation, the exclusion radius so derived should be divided by 2 to obtain a more realistic result. Since poor diffusion cases involve low winds speeds, the effect of a refinement of this sort is to make the associated exclusion radii considerably less pessimistic than those usually presented.

Regarding the persistence of a poor diffusion condition, it was reasoned that it would be far more realistic to assign a limit to this condition consistent with meteorological or climatological experience rather than to assume that a poor diffusion situation would persist almost ad infinitum.

In making a general hazards analysis, that is, an analysis independent of locale, there is an obvious difficulty in establishing this limit however. Thus the authors would be inclined to consider approximately 12 hours as a general, poor diffusion limit consistent with the following rationale: at most locations in mid-latitudes an inversion will set in roughly at sundown if the skies are clear; it will deepen throughout the night to a maximum depth of a few thousand feet, but it will be dissipated rapidly after sunrise. This is surely a very common case of poor diffusion, perhaps the most common. On the other hand, another analyst might insist on

assuming a much longer limit, arguing, for example, that the appearance of a complete cloud cover somewhat after sunset but before sunrise could result in persistence of the inversion and the accompanying poor diffusion. This possibility, as a generality, must be conceded. Thus the approach of establishing a reasonable upper limit to the persistence of poor diffusion, in a generalized hazards analysis, becomes somewhat philosophical.

In the case of a hazards analysis of a specific location, however, this approach is certainly valid, since meteorological records for a specific location can establish rather rigorously what the persistence of poor diffusion conditions has been at that particular place and thus permit reasonably exact prediction of what it is likely to be.

In further attempts to inject realism into the analysis of poor diffusion, the authors were impressed by the lack of a meaningful definition for "poor." What one hazards analyst considers poor can be considered perfectly miserable diffusion by another, and vice versa. Unfortunately, exclusion radii are very sensitive to this definition; even more unfortunate, it is frequently difficult to deny the slight possibility of even miserable diffusion conditions, the vagaries of weather being what they are.

Thus, it was concluded that an analysis of a poor diffusion case was likely to depend strongly on the pessimism (rarely, optimism) of the analyst, and that no model or refinement one could devise was likely to be susceptible to rigorous proof, with the exception of the wind direction variability case. These efforts to modify poor diffusion theory led to the further conclusion, subjective though it may be, that the majority of existing models or analyses are unduly, and sometimes illogically pessimistic.

In addition to the general study above, a number of specific investigations have been conducted. The methods used and the results are of some interest.

The plume case in which the continuous source strength is determined by the leak rate of the vapor container has been treated in some detail by R. F. Valentine.⁽⁴³⁾ He has assumed that the container leaked at a rate proportional to the square root of the pressure. Seepage of specific fission products was considered and the cloud concentration at any point in time and space was determined on the basis of Sutton's treatment. The fission product inventory was that expected after 3000 hours operation at 270,000 kw. For reasons given, the fission product load was assumed to be 2.77% of the total core fission product inventory at shutdown divided as follows:

<u>Selected Isotopes</u>	<u>Per Cent</u>
I, Xe, Kr, Br	2.42
Cs and Rb	0.34
Ba and Sr	0.0092

This constitutes an example of arriving at the radioactive load in a logical and reasonable manner for a specific reactor rather than arbitrarily choosing a total fraction for release. This approach, giving more realistic results, reduces markedly the assessment of damage and pays off in site requirement reduction.

From the concentrations in the plume and considering the concentrations produced on the ground by various removal mechanisms, it is next necessary to determine the exposures and doses to people downwind. As can be seen by examining Fig. 1-2, these doses are acquired in various ways.

One form of radiation is direct from the cloud to the receptor in the form of gammas. Another form is by inhalation directly from the cloud. Fallout from the cloud along the way and rainout of material from the cloud at any point lead to ground contamination. This in turn produces two dosage sources; one is direct radiation from the ground and the other is radiation by injection of products grown in the ground either directly (as for vegetables) or indirectly (as by consumption of cow's milk). The dose received is modified by the amount of time the receptor spends in the area and the **degree of protection** he has. Calculation of the dosages were made by Valentine⁽⁴³⁾ for the plume case specifically. The results are shown in Fig. 3-16 through 3-21 for the case of seepage from the container in proportion to the square root of the pressure which varies with time.

For the general release pattern of

100% of noble gases
70% of halogens
10% of remaining mixture

from the portion of the reactor affected by the accident, J. W. Healy⁽³⁴⁾ has made estimates of the maximum distance of possible lethal effects for a ground level source as a function of power level under neutral and moderate inversion conditions. The reactor is assumed to have operated for 200 days. The worst conditions can be represented by:

$$\log_{10} \text{ Distance} = -3.2 + 0.84 \log_{10} \text{ Power (Power} < 10^5 \text{ kw)}$$

Distance in miles, Power in kw

using the release pattern above.

FIG. 3-16
SELECTED FISSION PRODUCTS (REF. 43)

FIG. 3-17
STRONTIUM 90 (REF. 43)

INHALATION DOSAGE DOWNWIND FROM POINT OF EMISSION
FOR CONTINUOUS SEEPAGE OF FISSION PRODUCTS FROM
PLANT CONTAINER WITH CORE MELTDOWN

INHALATION DOSAGE DOWNWIND FROM POINT OF EMISSION FOR
IODINE FISSION PRODUCTS
CONTINUOUS SEEPAGE OF FISSION PRODUCTS FROM PLANT
CONTAINER WITH CORE MELTDOWN

FIG. 3-20
EXTERNAL DOSAGE (REF. 43)

FIG. 3-21
INHALATION DOSAGE FOR SELECTED
FISSION PRODUCTS (REF. 43)

DOSAGE DOWNWIND FROM POINT OF EMISSION FOR CONTINUOUS SEEPAGE
OF FISSION PRODUCTS FROM PLANT CONTAINER WITH CORE MELTDOWN

Calculations on the same basis give estimated maximum distance at which a dose to the lungs in one hour can be 100 rads. The most pessimistic results can be represented by

$$\log_{10} \text{ Distance} = -1.85 + 0.82 \log_{10} \text{ Power (Power} < 10^5 \text{ kw)}$$

Same units

Similarly maximum distances for long evacuations can be represented by

$$\log_{10} \text{ Distance} = -1.78 + 0.76 \log_{10} \text{ Power (Power} < 10^4 \text{ kw)}$$

and the maximum distance for crop hazard by

$$\log_{10} \text{ Distance} = -0.08 + 0.62 \log_{10} \text{ Power (Power} < 10^4 \text{ kw)}$$

(The power may be taken as the power of the affected portion of the reactor times the fractional release from a container in a 24 hour period, to estimate effects of small leaks - Editor)

Many other calculations of doses have been made. Each reactor hazard evaluation report contains such calculations, and many attempts at generalizing have been made. Of particular interest are some treatments of the general problem by Chamberlain and Megaw,⁽⁴⁴⁾ by Marley and Fry,⁽⁴⁵⁾ by J. J. Fitzgerald,⁽⁴⁶⁾ and by Stewart et al.⁽⁴²⁾

On the bases of doses calculated for specific releases and leak rates and for direct radiation from the gamma sources, the adequacy of the leak rate specification can be determined.

3.1.3 References

1. "Title 10 - Atomic Energy," Federal Register, 22, No. 19, Jan. 29, 1957.
2. "Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water," National Bureau of Standards Handbook 52 (Supt. of Documents, Washington, D. C., 1953).
3. "Permissible Dose from External Sources of Ionizing Radiation," National Bureau of Standards Handbook 59, Issued Sept. 24, 1954, (Supt. of Documents, Washington, D.C.).
4. "The Biological Effects of Atomic Radiation," National Academy of Sciences, National Research Council, Washington, D.C. (1956).

5. "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants," WASH-740, Appendix H. (U.S. Atomic Energy Commission, March, 1957).
6. B. Leonard, "Hazards Associated with Fission Product Release," Paper No. 428, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - Sept., 1958, (United Nations, Geneva, To be published).
7. L. S. Taylor, Health Physics, 1, No. 1, (June, 1958).
8. K. Z. Morgan, et al., "Maximum Isotope Contamination for Short Exposures," Paper No. 79, Proc. of the International Conference on the Peaceful Uses of Atomic Energy, Vol. 13, p. 139, (United Nations, N.Y., 1955).
9. W. Banks, "Recommendation of the International Commission on Radiological Protection," ICRP/54/4.
10. J. J. Fitzgerald and J. L. Flynn, "Exclusion Area Formula for Power Reactors," GE-KAPL document undated (prob. 1956 or 1957) no number.
11. J. J. Fitzgerald, "Criteria and Methods for Evaluating Environmental Reactor Hazards," KAPL-1527, (March, 1956).
12. R. O. Brittan, "Some Notes on Reactor Site Selection," Presented by American Delegates to the International Standard Organization Conference at Harrogate, England, June, 1958.
13. C. F. Miller, "Gamma Decay of Fission Products from the Slow Neutron Fission of U^{235} ," USNRDL-TR-187.
14. J. O. Blomeke and M. F. Todd, "Uranium-235 Fission-Product Production as a Function of Thermal Neutron Flux, Irradiation Time and Decay Time," ORNL-2127 (5 Volumes), August, 1957..
15. L. B. Borst, "Estimates of Amount of Radiation and of Accompanying Energy Liberation from Fission Products," Radiochemical Studies: The Fission Products, NNES, Vol. IV-9, Book 1, p. 344, Paper No. 34, (McGraw-Hill, New York, 1951). (Curves prepared by J. A. Wheeler.)

16. K. Way and E. P. Wigner, "Rate of Decay of Fission Products," Radiochemical Studies: The Fission Products, NNES, Vol. IV-9, Book 1, p. 436, Paper No. 43, (McGraw-Hill, New York, 1951) or Phys. Rev. 70, 115, (1946); and Phys. Rev. 73, 1318, (1948).
17. S. Untermeyer and J. T. Weills, "Heat Generation in Irradiated Uranium," AECD-3454, (February, 1952).
18. I. L. Faller, T. S. Chapman, and J. M. West, "Calculations on U^{235} Fission Product Decay Chains," ANL-4807, (May, 1952).
19. C. J. L. Lock, "Fission Product Formation in a Homogeneous Power Reactor," AERE-C/R. 1715 (June 28, 1955).
20. H. F. Hunter and N. E. Ballou, Nucleonics 9(5), C 2-7 (1951).
21. J. Moteff, "Fission Product Decay Gamma Energy Spectrum," APEX-134, (1953).
22. F. H. Clark, "Decay of Fission Product Gammas," NDA-27-39, (December, 1954).
23. J. B. Sampson et al., "Poisoning in Thermal Reactors Due to Stable Fission Product," KAPL. - 1226, (October 4, 1954).
24. W. H. Walker, "Fission Product Poisoning," CRRP-626, (January, 1956).
25. R. W. Stoughton and J. Halperin, ORNL, Private Communication, (January, 1954).
26. J. O. Blomeke, "Nuclear Properties of U^{235} Fission Products," ORNL-1783, (October, 1955).
27. C. D. Coryell and N. Sugarman (eds.), Radiochemical Studies: The Fission Products, NNES, Vol IV-9, Appendices A and B, (McGraw-Hill, New York, 1951).
28. J. M. Hollander, I. Perlman, and G. T. Seaborg, Revs. Mod. Phys. 25, 469-651, (1953).
29. D. J. Hughes and J. A. Harvey, "Neutron Cross-Sections," BNL-325 (July, 1955).

30. L. E. Glendenin and E. P. Steinberg, ANL, Private Communication (February 28, 1955).
31. J. R. Stehn and E. F. Clancy, "Fission Product Radioactivity and Heat Generation," Paper No. 1071, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - September, 1958, (United Nations, Geneva, to be published).
32. R. O. Brittan, "Fission Product and Activated Material Inventories for Reactors," for presentation at the 1959 Waste Disposal Hearings of the JCAE.
33. G. W. Parker and G. E. Creek, "Experiments on the Release of Fission Products from Molten Reactor Fuels," Paper No. 1074, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, September, 1958, (United Nations, Geneva, to be published).
34. J. W. Healy, Private Communication.
35. J. J. Fitzgerald and J. L. Flynn, "Exclusion Area Formula for Power Reactors," KAPL, unpublished.
36. L. Geller and R. Epstein, "A General Method for Evaluating Containment Shielding Under Normal and Emergency Conditions," Paper No. 435, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - September, 1958, (United Nations, Geneva, to be published).
37. F. D. White and D. H. Pack, "Meteorology as Related to Reactor Site Selection," Problems in Nuclear Engineering - Selected Papers, Vol. I, (Pergamon Press, N.Y., 1957).
38. O. G. Sutton, "The Problems of Diffusion in the Lower Atmosphere," Quart. J. Roy. Meteorol. Soc., 73, 257, (1947).
39. S. E. Singer and D. A. Baker, "Exclusion Radii for a WS-125A Test Base," AFSWC-TN-57-23 (deleted).
40. "Meteorology and Atomic Energy," AECU-3066, U.S. Weather Bureau, (July, 1955).
41. J. J. Fitzgerald, H. Hurwitz, Jr. and L. Tonks, "Method for Evaluating Radiation Hazards from a Nuclear Incident," KAPL-1045, (March, 1954).

42. N. G. Stewart, H. J. Gale and R. N. Crooks, "The Atmospheric Diffusion of Gases Discharged from the Harwell Pile (BEPO)," AERE HP/R 1452, (1954).
43. R. F. Valentine, "Hazards to the Area Surrounding PWR Due to Atmospheric Diffusion of Radioactivity," (September, 1957).
44. A. C. Chamberlain and W. J. Megaw, "Safe Distances in Reactor Siting," AERE HP/M 109.
45. W. G. Marley and T. M. Fry, "Radiological Hazards from an Escape of Fission Products and the Implications in Power Reactor Locations," Paper No. 394, Proc. of the International Conference on the Peaceful Uses of Atomic Energy, Vol. XIII, p. 102, (United Nations, N.Y., 1955).
46. J. J. Fitzgerald, "Reactor Safeguards," Sect. 8 of Dangerous Properties of Industrial Materials, by I. Sax, (Reinhold, N.Y., 1957).

3.2 Area II - Codes and Criteria

3.2.1 Committee Work

The American Standards Association Sectional Committee N6, "Reactor Safety" was formed and is sponsored by the ASME and ANS. This committee has set up a Subcommittee #2 on "Reactor Containment" under the chairmanship of Alf Kolflatt of Sargent and Lundy of Chicago. The work of this Subcommittee #2 is to produce a procedural standard for design, construction, and testing of reactor containment systems which will serve as a guide to legislative bodies and lead to systems which have adequately considered the public safety.

This subcommittee has examined many facets of the problem and has found that six existing codes should be referred to for conformation.

1. ASME Unfired Pressure Vessel Code Section VIII together with Code Cases applicable to containment vessels for nuclear power plants. The ASME Code applies to steel containment vessels designed for over-pressures in excess of 15 psig.

2. American Petroleum Institute Standard 620. Recommended Rules for the Design and Construction of Large, Welded, Low-pressure Storage Tanks should be followed for vessels that are designed for less than 15 psig internal pressure.

3. ACI-ASCE Joint Committee 323 - Report on Tentative Recommendation for Prestressed Concrete.

4. Building Code Requirements for Reinforced Concrete - ACI-318-56.

5. Pacific Coast Building Officials' Conference - Uniform Building Code.

6. National Fire Protection Association (International) - 1958 Annual Report prepared for 62nd NFPA Annual Meeting, May 19-23, 1958, at Chicago, Illinois.

Two other subcommittees of ASA-N6 are developing procedural codes which will tie in with the work of Subcommittee #2. Since various facets of the containment program overlap, and are not independent, certain divisions of effort were necessary. Subcommittee #1 on "Reactor Siting" and Subcommittee #7 on "Maximum Accidents" are tied into Subcommittee #2 as follows:

Subcommittee #7 will specify procedures for determining the mechanical and radioactive loads on the containment barrier. Subcommittee #1 will specify procedures for fixing the allowable maximum leak rate and auxiliary shielding requirements for the barrier necessary to insure that maximum permissible emergency exposures off-site are not exceeded (on the basis of the radioactive load).

3.2.2 Codes

(1) Because of the wide adaption of ASME Boiler Code by states as statutory requirements, many of the designs for containment vessels having over-pressures above one atmosphere have been based on, and in many cases have adhered to its Section VIII.

Some changes in this code of significance to reactor containment work are being proposed by ASME Nuclear Standards Board Subcommittee on Boiler Code Changes. The Subcommittee is now in the process of preparing a proposed new code Section to cover the specific case of containment vessels.

In addition, deviations from the code have been allowed in several rulings in specific cases by the ASME Boiler Code Committee. These cases and rulings are collected and presented here in detail for easy reference.

Case No. 1224,-1

Inquiry: Neither Section I nor Section VIII of the ASME Boiler and Pressure Vessel Code as now written precisely covers pressure vessels that are an integral part of a nuclear installation. Under what rules shall they be constructed?

Reply: The Committee recognizes that in the design of nuclear installations, requirements will differ from those of conventional boilers and pressure vessels.

1. It is the opinion of the Committee that vessels that are an integral part of nuclear installations built in accordance with the requirements of the ASME Boiler and Pressure Vessel Code as modified or defined in this and subsequent cases, meet the intent of the Code, and each vessel shall be marked as required by the section to which it is built including the appropriate Code Symbol. In addition, the words, "Case No. _____" shall appear on the Data Report.
2. All vessels that are an integral part of nuclear installations shall be constructed in accordance either with the requirements of Section I or with the requirements of Section VIII for vessels that are to contain lethal substances.
3. It is intended that jurisdiction over piping external to vessels shall terminate at:
 - (a) The first circumferential joint for welding and connections; or,
 - (b) The face of the first flange in bolted flange connections; or
 - (c) The first threaded joint in that type of connection.
4. Definitions.

The Reactor Vessel is that vessel in which nuclear fuel is present and which the nuclear chain reaction takes place.

Primary Vessels are those vessels, other than the reactor and containment vessels, which contain or may contain reactor coolant. These vessels may be heat-exchangers, pressurizer tanks, drain tanks, dewatering vessels, etc. Secondary vessels are all other vessels which do not contain reactor coolant or are not otherwise subject to irradiation. Secondary vessels are covered in appropriate sections of the Code.

Containment Vessels are those outer vessels which enclose the reactor vessel or portions of the primary coolant circuit or both. The containment vessels are not normally pressurized and are built to contain the lethal radioactive substances that may be released in case of an accident or failure of the reactor vessel or the primary coolant circuit or both.

Case No. 1226,-1,-2,-3

Inquiry: May containment vessels as defined in Case No. 1224-1 for nuclear reactor installations be built under Section VIII, Code for Unfired Pressure Vessels, without stress-relieving as specified in Par. UW-2(a) ?

Reply: It is the opinion of the Committee that these containment vessels fall under the category of those containing lethal substances but still may be built without stress-relieving provided the following requirements are met:

1. Plates and forgings of containment vessels exposed to the elements (not inside a heated enclosure) shall conform to specifications SA-300 for plates and SA-350 for forgings. These and other materials and the construction shall meet the impact test requirements of Par. UG-84 at a temperature not less than 30° F below the lowest recorded ambient temperature of the area in which the containment vessel is to be erected, except that the lowest test temperature may be assumed to be -50° F for any part in the United States.
2. Welding procedures and operators shall be qualified in accordance with Section IX.
3. All doors, nozzles, and opening frames shall be pre-assembled into shell plate and stress-relieved as complete assemblies for welding into the shell. Also, special consideration should be given to make the design of the reinforcement for large openings as strong as the shell (see Par. UA-7).
4. The thickness of the shell and head plate shall not exceed that for which stress-relieving is required in accordance with Par. UCS-56, except that for materials listed under Group P-Number 1 in Table UCS-23, stress-relieving is not required in thickness over $1\frac{1}{4}$ in., and up to $1\frac{1}{2}$ in. inclusive, provided a pre-heat of 200° F is used during welding.

5. All longitudinal and circumferential joints shall be of the double-welded butt type and shall be fully radiographed in accordance with Par. UW-51. All welds on doors, nozzles, openings, and frames, and all welds that cannot be radiographed shall be examined for cracks by magnetic particle or fluid penetrant method of inspection.
6. The case number shall appear on the Manufacturers' Data Report Form.
7. Provisions for corrosion shall be made in accordance with Par. UG-25. The mandatory requirements of Par. UCS-25 are not intended to apply to containment vessels.

Case No. 1228

Inquiry: In a two-stage construction of a containment vessel built under Case No. 1226 for a nuclear reactor, will the vessel be acceptable if during the pneumatic test certain of the welds in the lower part of the vessel are not available for inspection because of concrete shielding or concrete supports placed on inside and outside of the vessel after construction of the lower part?

Reply: It is the opinion of the Committee that containment vessels built under Case No. 1226 for nuclear reactors will meet the intent of the Code when erected in two stages, provided:

1. All inaccessible welds are double-butt welded, radiographed, and fully inspected and tested for leak tightness using a gas medium such as Halide leak detector test prior to being covered.
2. All other requirements of the Code are met including a pneumatic test of the completed vessel.
3. Inspection of welded joints in the lower part of the vessel during the pneumatic test will be waived where such joints are covered by concrete during the construction of the vessel, provided there are no openings or penetrations of that part of the vessel covered by concrete.

Case No. 1234

Inquiry: Under what special rules shall a nuclear reactor vessel or a primary vessel, as defined in Case No. 1224, be built in order to be acceptable for Code construction?

Reply: Pending development of more complete rules to cover nuclear vessels, it is the opinion of the Committee that a reactor vessel or a primary vessel shall meet the requirements of this case in order to meet the intent of the Code and to be stamped in accordance with Case No. 1224. Where differences exist the requirements of this Case take precedence over the Code rules for the subjects covered.

The requirements of this Case are as follows:

1. Thickness of each part of the vessel shall not be less than that determined by the Code rules using the applicable formula for the part with S values from the appropriate table in Section I or Section VIII.
2. The combination of stresses evaluated under item (1) with thermal stresses due to temperature distributions at any level of steady power operation, including internal heat generation, shall not exceed $1\frac{1}{2}$ times the S value.
3. For operating metal temperatures up to 800°F the maximum allowable bolt design stresses as used in Code formulas may be based on heat treated properties for operating metal temperatures 100°F or more below the tempering temperature, provided the stresses do not exceed $1/3$ of the yield strength at temperature.
4. Due regard shall be given to the creep and stress-rupture properties for prolonged exposure at temperature in order to assure adequate safety under all conditions of operation.
5. a. Compensation shall be made for all openings, regardless of diameter. The compensation shall be on either the reinforcement basis or the ligament efficiency basis as given in Section I or Section VIII.

Any compensation required shall be integral with the vessel wall or the nozzle or some with each.

- b. When all or part of the required compensation is attributable to the nozzle, the nozzle shall be attached by full penetration welds through either the vessel or the nozzle thickness. When no portion of the nozzle is required for opening compensation, the nozzle need not be attached by full penetration welds, except as limited by d.

- c. Each design detail shall be carefully considered to provide against operational failure such as might occur from thermal stress or external pipe reactions.
 - d. It is the intent of the foregoing that full penetration welds be used wherever possible, for the purpose of achieving continuity of metal and facilitating the required radiographic examination. Non-full penetration welds are intended to be allowed only when the spacing of the openings is such as to make the achievement of full penetration welds through the vessel or nozzle walls physically impossible.
6. All welds which are subject to stress caused by pressure shall be radiographically inspected, except where this is impractical for non-full penetration welds at close-spaced openings as permitted in item 5. Wherever possible all weld joints shall be designed to permit radiographic examination to Code standards. When the weld joints cannot be designed to permit radiography to Code standards, radiographic examination shall nevertheless be made in accordance with best obtainable practice. In addition, when radiography to Code standards is not feasible or when any radiography is impractical, welds shall be examined by other non-destructive methods such as liquid penetrant, magnetic particle, or ultrasonic, to prove their soundness. None of this item applies to seal welds.
7. The Code rules are intended to provide minimum safety requirements for new construction, and not to cover deterioration which may occur in service as a result of corrosion, erosion, radiation effects, instability of the material, or operating conditions such as transient thermal stress or mechanical shock and vibratory loading; nevertheless particular consideration shall be given to these effects with a view to obtaining the desired life of the vessel.
8. In view of these severe service requirements, particular consideration shall also be given to materials, construction, and inspection, including supplementary methods of non-destructive testing, so that soundness and good practice will result. Due regard shall be given to such items as smoothness of welds and to location and detail of structural attachments.

Case No. 1235

Inquiry: May containment vessels for nuclear reactor installations be built in accordance with Section VIII, Code for Unfired Pressure Vessels, without pressure-relief devices as required by Pars. UG-125 to UG-134?

Reply: It is the opinion of the committee that, since it is intended that these vessels be designed and built to safely contain all the lethal radioactive substances that may be released in case of a maximum credible accident affecting the reactor vessel or primary coolant circuit or both, and because of the hazardous character of the materials which might be released, pressure relief devices are not required. If for any reason such devices are installed, adequate provision shall be made for safe disposal of the effluent.

Case No. 1238,-1

Inquiry: May intermediate containment vessels, such as shield tanks, within the outer or final containment vessel as defined in Case 1224 for nuclear reactor installations, be built under Section VIII, Code for Unfired Pressure Vessels, without the stress-relieving specified in Par. UW-2(a) ?

Reply: It is the opinion of the Committee that intermediate containment pressure vessels surrounding the reactor vessel which are not required to retain radioactive materials under normal operating conditions, need not be built in accordance with special requirements for vessels containing lethal substances. All other intermediate containment pressure vessels surrounding the reactor vessel shall be considered to fall under the category of those containing lethal substances but still may be built without stress-relieving required in Par. UW-2(a) provided the requirements of Case No. 1226, omitting Par. 1 are met.

Case No. 1245

Inquiry: May a low-pressure rupture disk be used in the discharge pipe immediately adjacent to the steam safety valve outlet in a nuclear system to prevent atmospheric pollution which might result from small intermittent or persistent safety valve leakage when the safety valves are arranged to discharge into the containment vessel in which the nuclear system is located?

Reply: It is the opinion of the Committee that the use of a low-pressure rupture disk in the discharge pipe immediately adjacent to the steam safety valve outlet in a nuclear system to prevent atmospheric pollution in the containment vessel which might result from small intermittent or persistent safety valve leakage is permissible if the following conditions apply:

- a. The rupture disk shall be designed to burst at a maximum of 13 psi, including manufacturing tolerances.
- b. The safety valve settings are reduced so that the valve setting plus 13 psi shall not exceed the design pressure of the reactor or its steam separating drum.

- c. The internal diameter of the disk holder shall at least be equal to that of the safety valve discharge connection so that when the disk bursts there will be no restriction on the safety valve discharge.
- d. The rupture disk holder shall be designed to include a 1 in. drain opening on its inlet side. This drain opening shall be coupled to the required drain opening at the lowest point in the outlet of the safety valve body. The combined drains shall be piped to a suitable drain tank.

(2) The following information is of interest with respect to API Standard 620, "API Recommended Rules for the Design and Construction of Large, Welded Low-Pressure Storage Tanks" (February 1956).

This standard is comparable to the ASME Boiler Code for unfired pressure vessels in scope. It may be followed if the design pressure is below 15 psig, the lower limit applicability of the ASME code. It has an additional restriction, in that the temperature of the gas which it must hold may not exceed 200°F. It has some advantage in the fact that the ratio of maximum allowable stress values in tension to specified minimum physical property tensile strength is $3\frac{1}{3}$ instead of 4 if the design pressure is below 5 psig. Like the ASME code, a pressure relief valve is specified. Proper application of the code to design entitles owner to an official API symbol stamp of approval. Application is limited to the area not covered by API Standard 12C, "Specification for Welded Oil Storage Tanks." The standard may be obtained from American Petroleum Institute, Division of Refining, 50 West 50th Street, New York 20, N.Y. for \$2.00.

3.3 AREA III - PROOF OF CONTAINMENT

3.3.1 General

It is necessary to prove that the containment provided is adequate, both initially and on a continuing basis. Simply stated, the structure is adequate if integrity is always maintained to the extent that radiation injury and contamination are restricted to permissible levels. This implies that some means of testing be employed to show that the structure can withstand the mechanical and radioactive design loadings. Containment structures are placed in two categories to facilitate discussion of the proof testing:

1. "Semi-gas-tight" enclosures typified by conventional buildings treated to reduce inherent leakage to a practical minimum.

2. "Gas-tight" enclosures typified by welded steel vessels. For these two categories, the testing methods and evaluation of these methods will be discussed.

3.3.2 Proof of Strength

(1) Semi-gas-tight enclosures

This type is designed in accordance with usual structural codes and standards. If the prescribed codes and standards are adhered to, adequate strength is assured and strength proof tests are not required.

Some incidental strength proof testing of "semi-gas-tight" enclosures is accomplished during leak tests and leak rate tests, since design pressures are simulated for the purpose of these tests.

(2) "Gas-Tight" enclosures

This type of containment is designed to withstand internal pressures which might occur during or subsequent to severe accidents to the reactor plant. The design pressures usually have associated with them elevated temperatures which may induce thermal stresses. The range of overpressures for which this type of containment structures have been designed extends from 2 psig (CP-5) to 52.8 psig. Shell design temperatures range from nearly ambient to 650°F. In addition to calculated overpressures, these structures in some cases must withstand calculated underpressures (partial vacuum) produced as hot gases and vapor cool after an accident with attendant contraction and condensation. These structures must also be designed for loads due to natural phenomena, including negative pressure differentials produced by wind and barometric changes. Underpressures (partial vacuum) are relieved by vacuum vents which must be designed and maintained to limit the negative differential to a safe value. The range of negative pressures considered lies between -0.08 psi and -2.0 psi.

Containment structures of this type are strength proof tested by subjecting them to a pneumatic pressure in excess of the design pressure. When (in nearly every case) design is according to the ASME Code for Unfired Pressure Vessels,⁽¹⁾ or the API Standard 620,⁽²⁾ test is conducted at a pressure 25% above the design overpressure. No provision is made in the tests for elevated temperature, or for concentrated loading.

During pressurization for the proof test of welded steel vessels, it has been common practice to halt the compressor when the pressure reaches 2 to 10 psig to make preliminary tests for leaks and to observe the vessel for signs of distress. Continuing pressurization, other observations may be made at higher pressures; such as strain gauge tests at large openings, air-locks, etc. On reaching 125% of the design overpressure, test pressure is maintained for a period of one hour, completing the proof strength test. The pressure is then lowered to design pressure to conduct leak rate tests.

Specifications usually require that air-locks, equipment and freight doors, and other large attachments, be completed in the shop and tested operationally and under pressure prior to shipment to the site.

In a few cases, sampling of vessel contents has been required to determine whether a combustible mixture exists during pressurization and testing. In a large number of cases, such sampling has been required before weld repairs are made after depressurizing. In many cases, neither of these precautionary measures have been required.

At the present time it is generally conceded that proof of strength tests will be performed only during the construction phase; and that they are of no value for periodic proof of integrity tests during the life of the reactor plant. Additions to the plant would, however, be tested the same as original construction.

3.3.3 Proof of Leak-tightness

(1) Semi-gas-tight structures

For these structures, either positive or negative pressure differentials are created by means of blowers.

Before actual quantitative leakage rate tests are started, it has been the practice, and would normally be mandatory, to conduct various leak tests or surveys. Leak surveys, both internal and external may be made since pressures may be small (of the order of one inch of water). By these surveys, it is the hope that all detectable leaks may be located and repaired, and that after retesting of the repairs, the over-all leakage rate is reduced below the maximum permissible rate. In some cases where the

repair of a leak is difficult, or where the repairs would entail a serious delay of schedule, the rate of loss via such a leak is measured and taken into account when evaluating the over-all rate. In all such cases, the leak is repaired and tested independently after conclusion of the general test.

Luckow and Patterson⁽³⁾ describe methods for calculating and measuring leakage rates for semi-tight buildings. When ambient atmospheric pressure changes occur, the leakage is determined by the amount of air transferred between the building and the atmosphere to establish pressure equilibrium. This amount is related to the square root of the pressure difference thru a "leakage coefficient" analogous to the familiar orifice equation.

The proposed method for measuring this over-all leakage coefficient involves pressurization to 1 inch of water and measuring pressure drop as a function of time. This is repeated by evacuating the building to an equivalent negative differential and measuring pressure rise. The approximate equivalence of the coefficients for "in" and "out" leakage can thus be determined.

Certain inherent difficulties are encountered in making measurements. This difficulty arises because it is impossible to isolate the building from its environment during the measurement. Fluctuations in ambient temperature and pressure introduced errors in these measurements. Measurements must be taken during a calm, because wind generated pressure differentials are small themselves, so the region of interest is the region in which it is difficult to take data. However, it is stated that the results gave an approximation which can be used as a check on values determined by the method presented for predicting and calculating the leakage.

The extreme case of wind-caused building leakage occurs when the leaks are concentrated at the points of highest and lowest surface pressure. However, an assumption can be made that the leaks are uniformly distributed over the building surface. This may portray most cases accurately since all but the minute leaks are readily detectable and repairable.

An analyses of a specific building (the Ford Reactor building) was made and checked by test. First, the building effective leakage coefficient was measured in both directions as outlined above. Using this coefficient and weather data, calculations of leakage were made for a representative transient in both the "distributed" and "point" leak cases. The conditions imposed were (1) the pressure in the building equal to the highest pressure; (2) the incident angle of the wind equal to 45 degrees; and (3) a wind velocity equal to 10.5 miles per hour. The result indicated that the pressure would fall quite rapidly approaching its equilibrium value asymptotically. The pressure would decrease to within 10% of equilibrium

pressure in 3.75 min for the distributed case and in 5.9 min for the point case. The leakage due to the transient is more important than assuming equilibrium for the entire 24-hour period. Based on the weather conditions of the representative transient, the distributed leakage including transient would be 0.07% greater than equilibrium for the whole period and the point leakage would be 0.17% greater.

To test the validity of the proposed leakage prediction scheme and to establish whether the building may best be described by the point or the distributed case, an independent measurement of building leakage was undertaken. This independent measurement involved introduction of adulterant (tracer) gases (CO_2 and Freon 12), allowing time for complete diffusion and pressure equalization, and then measuring concentration of adulterants by mass spectrometer over a 28-hour period. The amount of air change, or leakage, was deduced from the concentration change as a function of time. In the latter test leakage was due only to natural phenomena, wind and ambient temperature-pressure changes. Results were compared with predicted leakage based on leakage coefficient measured in the former tests and weather conditions prevailing during the tracer gas test, and showed leakage rates of the same order.

The methods set forth may also be used to determine the necessary capacity of the clean-up system required in the "controlled leakage" approach to limiting the release of radioactive materials. For this method, the air is drawn out of the building and exhausted through a clean-up system which removes part or all of the noxious constituents.

The "controlled leakage" scheme is proposed by Binford and Burnett⁽⁴⁾ for a conventional building, following a radioactive material release from the reactor. It is proposed that if air is exhausted from a building at a rate such that all movement of air, other than movement through the exhaust system, is into the building, then the building is "gastight".

If the building housing the ORR is gastight under the above definition, then the radioactive gas so exhausted can be disposed of by scrubbing, diluting with large amounts of air, and passing up an available stack. The rate at which gas can be disposed of in this manner will depend upon the hazard resulting from the emission of the radioactive gas from the stack. Control of this hazard will dictate the maximum rate of exhaust from the building and thus determine the type of building construction required. The proposal is to replace random, uncontrolled leakage concentrated gas from the building at the surface of the ground with controlled leakage of dilute gas at an elevation of 200 to 250 ft. It is assumed that a suitable filter is placed in the exhaust system so that all particulate matter will be removed and that a scrubber which will remove 99.90% of the iodine is also incorporated in the system.

(2) "Gas-Tight" Structures

In addition to the strength proof test for "gas-tight" containment structures, there has generally been required some proof of compliance with requirements for permissible leakage rates. The criterion for this test is a maximum permissible leakage rate set forth in the specifications or in a test procedure based on commitments in a hazards summary report. The basis for establishment of the permissible leakage rate is discussed elsewhere in this report; however, a lower bound has been established by accuracy limitations in detecting and measuring the loss with credence. As techniques for measurement of leakage rates are improved, or new methods of greater accuracy are developed, it may be feasible to prescribe lower allowable rates which may be based on some uniform code, or set of rules, as proposed elsewhere in this report.

While going to design pressure for the leak rate test, leak detection tests are usually run. These take the form of measuring the pressure loss in a given period. If this loss is greater than a calculated maximum allowable, leaks are located by soaping, or sound, and repaired. In other cases Freon is injected during the early stages of pumping, and leaks located with mass spectrometer instruments. More generally, these preliminary leak tests are conducted by applying soap solution to all welds and closures. If leaks are found, the vessel is depressurized and the repairs made; the pressure brought back to preliminary level, and repairs tested with soap solution.

The most difficult problems posed in determining the leakage rate for containment structures of "gas-tight" construction are those encountered in detecting and measuring very small leakage losses from vessels of the magnitude normally required. There have been several methods employed or proposed, each of which has advantages and disadvantages for application to initial ("bare shell") testing, or for final and periodic testing. Included in these are (a) the so-called absolute temperature-pressure method; (b) the so-called reference method; (c) a method whereby leakage is measured by measuring the amount of make-up air required to maintain the status quo conditions; (d) the method in which an adulterant gas is introduced into the vessel contents and its escape detected by "sniffing" all welds and seals, by regional checks, by patch tests, by collection in an over-all envelope, or by sampling and testing for reduction in concentration; (e) the method in which, after the initial leak test and radiography of all welded joints and dye-penetrant or magnetic particle checking of penetrations, the general tightness of the vessel is assumed to be secure and further testing is accomplished by constructing boxes around each penetration, checking the boxes for leak tightness, drawing a vacuum on each box and measuring pressure rise; and then integrating these individual leakage rates to represent the over-all leakage rate of the vessel.

(a) The absolute temperature-pressure method has been applied for a number of containment vessels for both initial and final leakage rate tests with varying degrees of success. The degree of success depends on the validity of assumptions made regarding the influence of certain factors on the reliability of the calculations; and the accuracies to which the readability and repeatability of the readings are held. In initial tests, in some cases, shell skin average temperatures were taken to correct for change in physical size of the vessel due to thermal expansion and contraction; whereas, in other cases this was ignored. Also, in some cases, for all tests, water vapor conditions were monitored and corrections made for dry air; whereas, in other cases this was ignored, or steps taken in the hope of eliminating conditions which would introduce this error. Accuracies depend on the capabilities of the instrumentation and the care and precision used in making readings. In all cases an effort was made to provide instruments delicate enough to permit reading to accuracies necessary to reduce errors to amounts consistent with the magnitude of the specified permissible leakage rate. For instance, in the initial test for the EBWR containment vessel,⁽⁵⁾ the results showed that pressure readings to 0.01 in. Hg and temperature conversion to 0.1°F allowed excessive scatter in the points plotted, which had to be counteracted by extending the test until sufficient data were obtained to show a definite trend. For the final test and first periodic test, an absolute manometer was built which could be read to 0.001 in. Hg, and a method for calculating, direct from millivolt values was devised to eliminate the inaccuracies introduced by conversion to °F. These means reduced the scatter of the plotted points greatly and permitted reduction in the duration of the test to a day or two instead of eight days.

(1) J. C. Heap⁽⁶⁾ describes the direct temperature-pressure method and gives the theory upon which the calculations are based.

By taking temperature and pressure measurements, it is possible to establish the weight loss of air over a period of time for a containment system. The direct pressure-temperature measurement system leakage is analyzed below by assuming the equation of state for a perfect gas for the conditions at two different times and that the volume of the containment system at the two different periods has remained constant. The containment system at time t_0 is:

$$p_0 V = w_0 R T_0 \quad (I. 1)$$

where

p_0 = containment system absolute air pressure at initial selected state, psia.

V = containment system free air volume ft^3

w_0 = containment system air weight at initial selected state, lb.

T_0 = containment system absolute air temperature at initial selected state, °R.

R = gas constant for air, ft/°R

and at time t_1 is:

$$p_1 V = w_1 R T_1 \quad (I. 2)$$

where

p_1 = containment system absolute air pressure at a time after initial selected state, psia.

T_1 = containment system absolute air temperature at a time after initial selected state, °R.

w_1 = containment system air weight at a time after initial selected state, lb.

Dividing Eq. I. 2 by Eq. I. 1 and solving for the air weight at the second selected time gives:

$$w_1 = w_0 p_1 T_0 / p_0 T_1 \quad (I. 3)$$

The weight loss of the air between the two periods is:

$$w_0 - w_1 = w_0 (1 - p_1 T_0 / p_0 T_1) \quad (I. 4)$$

As the weight loss of the air is a function of time, a time basis must be incorporated to provide a comparison between tests conducted at different times. The most logical basis is twenty-four hours, as this includes daytime and nighttime thus giving a proximity to a cycle. The inclusion of a time factor requires the supposition that the leakage when correlated with the proposed basis be the ratio of twenty-four hours over the interval of time in hours; hence, Eq. I. 4 becomes:

$$\Delta w = 24 w_0 / \Delta t (1 - p_1 T_0 / p_0 T_1) \quad (I. 5)$$

where

$\Delta w = 24 (w_0 - w_1) / \Delta t$, the weight loss of the air for a selected time interval.

Δt = selected time interval expressed in hours.

The distribution of the radioactive particles for a nuclear incident is frequently considered proportional to the free volume or weight of air in the containment system. The leakage of the system is expressed as the air weight difference between the initial and final selected

states on a projected time basis divided by the air weight of the initial selected state; thus,

$$\Delta w/w_0 = 24/\Delta t (1 - p_1 T_0 / p_0 T_1) \quad (I. 6)$$

If the initial and final temperatures for the system are the same, then Eq. I 6 reduces to:

$$\Delta w/w_0 = 24/\Delta t (1 - p_1/p_0)$$

In former tests, efforts have been expended to obtain identical initial and final temperatures by having a minimum test run of a day. Whenever possible a test is started between midnight and six o'clock in the morning as this offers the greatest opportunity of obtaining equal temperatures for a twenty-four hour test.

Corrections for change in physical size of the vessel due to thermal expansion and contraction and for dry air are not considered in this paper, but can obviously be added if deemed advisable.

The question pertaining to the number and location of temperature sensing elements remains controversial. A second controversial question is that pertaining to whether or not blowers should be installed in the vessel to induce air circulation. In some cases they are used; whereas, in other cases they are not called for. Agreement should be reached on both of these questions, probably through an experimental program.

With respect to the temperature measurements, Zissi and Diehl⁽⁷⁾ report test results for the West Milton sphere in which the absolute temperature-pressure method was used for a "bare shell" test. In a "preparatory test" it was determined that the temperature reading given by a "thermistor" placed at the geometric center of the sphere was coincided with the arithmetic mean of twenty "thermistors" arranged throughout the sphere, for the equilibrium period during the early morning hours of a normal day. In addition to temperature, a number of other factors were dealt with in the West Milton test. Error introduced by change in physical size of the sphere by thermal expansion and contraction was assumed negligible or within the readable accuracy of the instrumentation; temperatures of the steel shell, therefore, were not taken. Also, to eliminate errors due to evaporation and condensation of water in the sphere, water which was condensed during pressurization was drained out prior to start of the leakage rate test. However, it is not certain that this step positively prevents this error since subsequent condensation may occur in case local areas of the shell reach temperatures below the existing dew-point within the shell.

Heineman's⁽⁵⁾ procedure for the absolute temperature-pressure test performed for the EBWR initial leakage rate test is typical.

I. Preliminary preparation while reactor is in operation.

A. Set up air compressors near permanent inlet penetration. Total capacity should be approximately 600 cubic feet per minute to pressurize to 15 psig in about 12 hours.

1. Piping manifold and delivery line can be completely installed at this time.
2. Install duplicate pressure gauges on this system.
3. An isolating valve is required in the delivery line. This valve must be tested for air tightness before installation. A globe type valve with composition disk, or a tapered plug valve (lubricated), should be used. Delivery line and valve should be 3 inches or larger.

B. Set up all instrumentation, as follows:

1. Five thermocouples (calibrated, iron-constantan) suspended near center line of building at the following elevations:
 - a. Above central portion of ceiling midway between ceiling and overhead tank.
 - b. Below lower ceiling slab elevation (clear of crane).
 - c. About 2 feet above main floor.
 - d. Midway between condenser floor and pump floor.
 - e. About 5 feet above basement floor.
2. Thermocouple leads are run to thermocouple cold junction box on condenser floor above television camera and connected to spare leads to the potentiometer instrument in control room.
 - a. The corresponding terminals in control room are wired through a selector switch to the manual potentiometer and ice bath reference couples in the usual manner.
 - b. A "K2" manual potentiometer should be used for this test.
 - c. Thermocouples must be iron-constantan.

3. Wet-bulb and dry-bulb thermocouples mounted on water bath with duct and blower with Variac speed control are located above the central portion of the ceiling slab near the first temperature sensing couple and on the condenser floor near the building center line. The leads from these couples are connected through to the multi-point instrument in the control room in the same manner as described above and thence to the selector switch and "K2" instrument. Water supply and drain tubing must be provided for each bath as well as 110V electric supply to the fans.
 4. The 63 inch barometer, standard barometer, 30 inch manometer, and weight tested pressure gauge (0-20 pound) are located in the control room with tubing, valves, and connection to shell.
- C. Set up two blowers above ceiling for drawing air up one side and down on the opposite side. Provide electrical power leads to these blowers.

II. Preparation which must await reactor shutdown.

- A. Adjustment and settings for power plant components, valves, instrumentation, etc.
1. By means of a clamp-on ammeter reduce the motor loadings of the air-conditioning blowers to about one-half the rated overload current by partially closing the discharge louvers. Set air-conditioners at about 70°F.
 2. Adjust the water supply to both wet-bulb baths to obtain maximum overflow through drain without overflowing bath. Set Variacs to obtain correct wet-bulb depression with minimum flow.
 3. Trip solenoid latches on inlet and exhaust butterfly valves. Check for air-tightness.
 4. Check vacuum breaker valves for operation and tightness.
 5. Make sure that freight door is closed and bolted up tight.
 6. Remove or open any containers in building such as paint cans, drums, bottles, etc.
 7. Open two water box vent valves on condenser in building.
 8. Close circulating pump discharge valves, condenser inlet valves and turn off control power.
 9. Open condenser vacuum breaker.

10. Open vent condenser in Liquid Recovery System.
11. Turn off hand water valves to overhead water storage tank water regulator and also the emergency cooler valve.
12. Secure reactor air exhaust blower.

III. Test Procedure

- A. Make sure that air-conditioners are running and start blowers in "attic."
 1. Close outer emergency air-lock door leaving inner door open.
 2. Close inner main air-lock door leaving outer door open.
- B. Start compressors and pressurize building to 15 psig.
- C. While pressure is building up check all possible leak sources; such as, air-lock door gaskets, air-lock shafts, equalizing valves, portholes, bolts, etc., conduits, cable penetrations, cable troughs, Micarta disks and Amphenol fittings, valve stems, and all new welds.
 1. Close inner emergency air-lock door and open outer door and check inner door, shaft seal, equalizing valve, etc.
 2. Disengage inter-lock on main air-lock and crack equalizing valve on inner door, then quickly close and seal outer door and check all leak sources at outer door, conduits, etc.
- D. With pressure at 15 psig close isolating valve in air inlet line and stop compressors.
- E. For a period of 24 hours, or more, take all temperature readings, wet-bulb and dry-bulb readings (record all of the foregoing readings in millivolts), and total pressure reading at one-half to one hour intervals. From the average of the millivolt readings for the five temperature thermocouples, the average of the two dry-bulb and two wet-bulb temperatures, and the total pressure readings calculate the quantities and corrections in accordance with the following formulas:

$$W_{st1} = \frac{-0.24 \times \Delta T + W_{st2} \times h_{fg2}}{(h_{g1} - h_{f2})}$$

$$W_{st2} = \frac{53.3 \times T_{wb}}{144 \times (P_{tot} - P_{sat2}) \times v_{g2}}$$

$$P_{st1} = P_{tot} \left(\frac{W_{st1} \times 85.7}{W_{st1} \times 85.7 + 53.3} \right)$$

in which:

T_{db} = dry bulb temperature, °R

T_{wb} = wet bulb temperature, °R

W_{st1} = weight of vapor per 1 lb. dry air at T_{db}

ΔT = wet bulb depression

W_{st2} = weight of vapor per 1 lb. dry air at T_{wb}

h_{fg2} = heat vaporization at T_{wb}

h_{g1} = total heat of vapor at T_{db}

h_{f2} = heat of liquid at T_{wb}

P_{tot} = total pressure in vessel, psia

P_{sat2} = saturation pressure of vapor at T_{wb}

v_{g2} = specific volume of vapor at T_{wb}

P_{st1} = partial vapor pressure in vessel

Corrected Dry Air Volume (STP) =

$$V = 400,000 \times \frac{K}{T_{av} + K} \times \frac{P_{tot} - P_{st}}{14.6959}$$

in which

$$T_{av} = \text{Av. Air Temp. in MV} \quad K = \frac{T_{av}(mv)}{T(^{\circ}F) - 32} \times 491.69$$

P_{tot} = Total Pressure in psia

P_{st} = H_2O vapor pressure in psi (P_{st1} above)

It is preferable that a complete set of readings be taken every half hour, but in no case at greater intervals than one hour. Pressures shall be read to an accuracy of ± 0.001 inch Hg; and temperatures shall be read to an accuracy of ± 0.0001 millivolts. Temperature conversions shall be made to within $\pm 0.1^{\circ}F$ for wet and dry bulb temperatures.

Slide rule calculations are permissible for determination of vapor pressures (to 3 significant figures) but all other calculations shall be done on a calculating machine (desk type) to preserve values to five (5) significant figures.

The conduction of the tests on EBWR, with descriptions of instrumentation and results are given by Heineman and Fromm.⁽⁵⁾ The number and arrangement of resistance thermometer elements for taking

air temperatures, thermocouples for taking shell skin temperatures, blowers mounted in the vessel for air circulation, air thermocouples, and wet-bulb-dry-bulb baths are shown. Both the initial bare shell test and the final acceptance test, as well as the first periodic test, are covered.

(b) The reference method consists of measuring pressure differentials between the leaking vessel being tested and a tight reference vessel placed within the containment vessel. Basically the two vessels are connected thru a manometer, having both been pressurized to design pressure initially. As the containment vessel leaks, its pressure drops below that of the reference vessel. The stated justification for this system is that the reference vessel can be so placed as to have the same mean temperature as the vessel being tested, thus eliminating temperature effects by automatic compensation. Since the experiment is not time independent, it is implied that no sensible lag exists as the reference vessel air temperature adjusts to the changing air temperature of the vessel being tested. There is also the implication that the reference vessel does not itself leak appreciably in comparison with the vessel being tested. In practice, various factors deny this automatic compensation. Hence temperature and pressure must be measured with this method as well as for the absolute method (and to the same accuracy) so that corrections may be made.

This reference method was first proposed by Untermeyer and Layton⁽⁸⁾ and used by them to test the VBWR. Since then it has been applied to tests of a number of containment vessels both for initial and final leakage rate tests. The degree of success has depended on the care and accuracy in taking readings. Up to the present time only very meagre temperature data have been taken during tests employing the reference method. Omission of any real temperature survey is based on the assumption that complete temperature compensation is achieved in this method. This assumption is not permissible except in the rare case where the average air temperatures in both vessels are the same, and the initial and final temperatures are the same. In lieu of a temperature correction as such, it can be shown that the correction can be made in terms of absolute pressures at initial and final conditions in the vessel and the reference volume, if it is assumed that the average air temperature in the vessel is equal to the average air temperature in the reference volume (which condition does obtain twice during a normal day). This, of course, would require accurate measurements of absolute vessel pressure throughout the test period.

Heap⁽⁶⁾ has provided the correct analysis for such a system:

It must be kept in mind that for a gas, the pressure is a function of temperature. Using the equation of state for a perfect gas and assuming the free air volume of all systems remains constant, the

conditions at two different times for the reference system and the containment shell are:

at time to

$$p_0 V = w_0 R T_0 \quad (\text{II.1a})$$

$$p_{R0} V_R = w_R R T_{R0} \quad (\text{II.1b})$$

where

p_0 = containment system absolute air pressure at initial selected state, psia.

p_{R0} = reference system absolute air pressure at initial selected state, psia.

V = containment system free air volume, ft.³

V_R = reference system free air volume, ft.³

w_0 = containment system air weight at initial selected state, lb.

w_R = reference system air weight at initial selected state, lb.

T_0 = containment system absolute air temperature at initial selected state, °R.

T_{R0} = reference system absolute air temperature at initial selected state, °R.

R = gas constant for air, ft/°R

and at time t_1

$$p_1 V = w_1 R T_1 \quad (\text{II.2a})$$

$$p_{R1} V_R = w_R R T_{R1} \quad (\text{II.2b})$$

where

p_1 = containment system absolute air pressure at a time after initial selected state, psia.

p_{R1} = reference system absolute air pressure at a time after initial selected state, psia.

w_1 = containment system air weight at a time after initial selected state, lb.

T_1 = containment system absolute air temperature at a time after initial selected state, °R

T_{R1} = reference system absolute air temperature at a time after initial selected state, °R

For the reference system to be valid, the weight of the air in the reference volume must remain constant from one state to the next.

The pressure differential between the reference system and the containment system for the different periods are:

$$\Delta p_0 = p_{R0} - p_0 = R (w_R T_{R0}/V_R - w_0 T_0/V) \quad (\text{II.3a})$$

and

$$\Delta p_1 = p_{R1} - p_1 = R (w_R T_{R1}/V_R - w_1 T_1/V) \quad (\text{II.3b})$$

where

Δp_0 = differential pressure, reference system minus containment system at initial selected state, psia.

Δp_1 = differential pressure, reference system minus containment system at a time after initial selected state, psia.

Solving for the weight of air in the containment system at the two selected times yields:

$$w_0 = V/T_0 [w_R T_{R0}/V_R - \Delta p_0/R] \quad (\text{II.4a})$$

and

$$w_1 = V/T_1 [w_R T_{R1}/V_R - \Delta p_1/R] \quad (\text{II.4b})$$

The weight loss of the air between the two selected states then becomes:

$$w_0 - w_1 = V w_R /V_R [T_{R0}/T_0 - T_{R1}/T_1] + V/R [\Delta p_1/T_1 - \Delta p_0/T_0] \quad (\text{II.5})$$

Using the 24 hour basis as discussed in the direct pressure-temperature measurement system the leakage will be:

$$\Delta w = 24/\Delta t \left\{ V w_R /V_R [T_{R0}/T_0 - T_{R1}/T_1] + V/R [\Delta p_1/T_1 - \Delta p_0/T_0] \right\} \quad (\text{II.6})$$

The leakage of the system expressed as the ratio of the weight loss to the initial weight is:

$$\Delta w/w_0 = 24/\Delta t \left\{ V w_R /V_R w_0 [T_{R0}/T_0 - T_{R1}/T_1] + V/R w_0 [\Delta p_1/T_1 - \Delta p_0/T_0] \right\} \quad (\text{II.7})$$

or by substitution of Eqs. II. 1a and II. 1b yields:

$$\Delta w/w_0 = 24/\Delta t \left\{ p_{R0} T_0 / p_0 T_{R0} [T_{R0}/T_0 - T_{R1}/T_1] + 1/p_0 [T_0/T_1 \Delta p_1 - \Delta p_0] \right\} \quad (\text{II. 8})$$

For the case where $T_0 = T_{R0}$ and $T_1 = T_{R1}$ Eq. II. 8 reduces to:

$$\Delta w/w_0 = 24/\Delta t p_0 [T_0/T_1 \Delta p_1 - \Delta p_0] \quad (\text{II. 9})$$

Only with the further restriction that $T_0 = T_1$ does the fractional weight loss equal the fractional change in pressure.

(c) Measurement of Make-up Air. This method was first proposed for containment vessel testing by Heineman.⁽⁹⁾ It was considered for testing the EBWR shell but was never applied, nor has it been up to the present. Heineman proposed that leakage be determined by measuring the amount of air which must be added to the vessel to maintain the pressure at its initial value. The make-up air is supplied from a compressed air cylinder which is placed on a platform scale. A flow control valve is used to maintain constant flow during a given observation. The flow is to be measured by means of a rotary type flow meter as a check on the scale readings.

This scheme, because of the relatively high tare weight of the bottle, dictates the use of a scale with a range which may be too great for the accuracy required. If this proves to be true, an alternate method of supplying the air by water displacement of the air from a small tank may be used with greater accuracy.

This method requires reliable monitoring of air temperature so that the proper test period may be selected. It is possible that tests of relatively short duration could be made successfully with this method.

(d) The method in which an adulterant gas is introduced into the vessel and its escape detected by various means has been used many times in other applications. It has not been used for leak rate tests of gas-tight containment vessels. Leak surveys by sniffing all welds and seals have been used for preliminary tests and repairs before leakage rate tests. Detection and measurement by collection, by regional check, by patch tests, and by means of an overall envelope have been proposed for determining total leakage rate. Also, determination of leakage rate has been proposed by sampling and testing for reduction in concentration of the adulterant gas. Adulterants proposed include helium, freon, and radioactive gas. The adulterant method has an advantage, in that it may be possible to run the tests at low pressures. This feature becomes very important

in the case of periodic tests after all equipment and components are in place and where some of these components cannot be subjected to high over-pressures required for the measurement of leakage rate by the first three test methods described above. It might also be possible to conduct this type of test while the plant continues in operation, thus avoiding costly, or intolerable, shut-down for many hours, or days, required to conduct loss of pressure tests. As has been stated previously, it is generally conceded that repetition of proof of mechanical strength in periodic tests will not be required, so that low pressure tests may be adequate. If helium were used as the adulterant it would also be possible for operating personnel to continue normal access to the plant during the test. An objection to low pressure tests, as noted above, is the fact that leakage which would occur at full design pressure may not show up at all at low pressures. [NOTE The use of helium gas in leak detection has been detailed by Metz⁽¹⁰⁾ - Ed.].

(e) Method of Checking Individual Penetrations. This method was adopted in leakage rate testing of the containment sphere for the British fast reactor at Dounreay Scotland. It is first necessary to perform the strength proof test. During this test, or earlier in the shops, all welded joints are radiographed or checked by dye-penetrant or magnetic methods. A soap bubble check is performed on all seams during the pressure proof test. After all seam leaks have been stopped and all faulty welds repaired it is reasonable to assume that the structure is generally sound and tight. If this is true, any leakage must come from positions of penetration, which are usually quite high in number. The method therefore proposes to check the leakage at each penetration and to obtain the total leakage by integration.

The scheme proposed for the test is to build a tight box at each penetration exterior to the shell. A vacuum is then drawn on this box and the pressure buildup in the box due to leakage from within the sphere is determined for a period of approximately 10 min. This is done for each penetration. All leakages for such tests are to be added up and related to the total volume of the sphere to give a leak rate in per cent of contained air per 24 hours. The sensitivity of this test can be one or two orders of magnitude greater than that from integrated pressure tests. It may well prove to be the answer to one of the biggest problems, that of periodic checking of the leak tightness of containment vessels. If the philosophy leading to the principal basic assumption of the test is acceptable, this type of test may even prove to be most satisfactory for initial integrated tests.

Should such a method be suitable for both initial and periodic leak rate testing, it is possible to design these individual penetration enclosures into the structure so that they are easily available for tests at any time. The tremendous advantage of not having to pressurize the vessel makes it possible to determine leak-rate any time. It is even possible to set up a continuous test.

3.3.4 Experience in Performing Proof Tests

Brittan, Heap, and Heineman⁽¹¹⁾ have surveyed the methods used in performing leak rate tests. The procedures and test experiences for a number of actual structures have been considered by them. The sources of information are informal reports and conversations, and are not readily available. The notes which they have accumulated or prepared on the subject are reproduced here:

(1) The specified leakage rate has been expressed in a number of different ways. Many of these are vague and ambiguous. For instance, the rate has often been specified as "per cent loss per 24 hr" without stating whether this be per cent of vessel volume, per cent of "contained air", per cent of "stored air" (air added during pressurization), per cent of design pressure, or what.

In the case of EBWR, the rate was specified as 500 cu ft at STP per 24 hr for the initial, or "bare shell" test after completion of the vessel and prior to installation of any internal or external concrete or back-filling of the excavation. This is approximately 0.1% of the "stored air" expressed in terms of "free air". For the "final test" at EBWR, the rate was specified as 1000 cu ft at STP per 24 hr, which is approximately 0.25% of the "stored air" in the "free volume" after installation of all internal concrete and equipment. For the VBWR at Vallecitos, the rate for the "bare shell" test was given as one per cent of the rated pressure for the vessel, which implies fraction of "stored air", per 24 hr. At EBR-II the maximum permissible rate is given as 750 cu ft/24 hr. of air at absolute-design pressure, or "compressed air", which represents about 0.125% of the gross vessel volume. The rate for the Enrico Fermi power plant containment vessel is given as 500 cu ft of free air per 24 hr which is about 0.06% of "stored air" in the "free volume". For one power plant the rate was specified at the 25% overpressure and had to be calculated for the design pressure, since for safety, the test could not actually be conducted at a pressure greater than this. In addition, in this case, the rate was expressed as 0.5% loss without stating whether this be "contained" or "stored" air. In view of this confusion in designation of the permissible leakage rate, the need for a uniform and definite terminology is glaringly apparent.

The least ambiguous of these specifications is that used for VBWR. This scheme also ties in best with the fission product leakage which is usually designated as a function of the square root of the pressure. It is consistent with determining quantity leaking out using an expression analogous to the usual orifice flow equation. An equally unambiguous designation would be in terms of mass of air per unit time. A drawback to such a procedure is that one has an extra step to go through in converting measurements.

A further difficulty occurs in translating the specified leak rate for air to that for fission products. Suppose that the fission products released to the contained air are uniformly distributed and the air pressure is increased one atmosphere by heating. If the pressure is reduced back to its original value by leaking the hot air out, only half of the air, and hence half of the fission products will have leaked out. For other overpressures the fraction of fission products which can possibly leak out is just the overpressure in atmospheres divided by the overpressure plus one atmosphere. If the pressure rise is determined in part by expanding compressed material as a gas, the relation is more obscure. This "maximum fraction" should be considered in every case, but is especially influential where the overpressure is small.

By manipulation of the expressions for fractional weight loss based on "total" air and "stored" air, it can be shown that

$$\frac{\text{fraction of "total" weight leaking out}}{\text{fraction of "stored" weight leaking out}} = \frac{\text{gage pressure}}{\text{absolute pressure}}$$

Furthermore, it can be shown that the expressions for weight loss fractions based on initial stored air are related to those based on final stored air by:

$$\frac{\text{fraction of initial stored air leaking out}}{\text{fraction of final stored air leaking out}} = \left(1 + \frac{\Delta p}{p_0}\right) \left(1 + \frac{T_0 - T_1}{T_1}\right)$$

Since the fraction based on final stored air is

$$f_{\alpha} = \frac{w_0 - w_1}{w_1} = \frac{1}{p_{1\text{gage}}} \left[\Delta p_1 - \frac{T_1}{T_0} \Delta p_0 \right],$$

it is implied that if one is willing to sacrifice assurance that the manometer is working and thus make $\Delta p_0 = 0$, the fractional leakage based on final stored air is equal to fractional pressure change without temperature correction and can be compared with the criterion provided it is less than 0.9 of the maximum allowable leakage. Of course $\frac{T_1 - T_0}{T_1}$ must be less than about 0.1 if $\frac{\Delta p}{p}$ is less than 0.005. The development here also requires that $T_0 = T_{R0}$ and $T_1 = T_{R1}$.

(2) Two other factors which affect the accuracy and reliability of test results are changes in physical size of the vessel due to thermal expansion and contraction, and water vapor conditions. Up to the present time, even for "bare shell" tests, none of the leakage tests employing the reference method has made corrections for thermal changes in size, nor for water vapor conditions. In an effort to eliminate errors due to water vapor conditions, vessels have been drained of condensate just prior to starting leakage rate measurements, and in some cases,

immediately after final readings the drain valve was opened to make sure that no further condensation had taken place. In all cases where this was done, it was found that condensation had not occurred during the test in sufficient quantities to form a puddle at the bottom of the vessel, or that interim condensation had re-evaporated. In some cases (at Dresden and others) a hygrometer has been used for a general check on water vapor conditions.

(3) One containment system tested by the reference method had temperatures read only to the nearest 1°F. Although no corrections were made for temperature changes, the inherent error was calculated and found to be approximately $\pm 50\%$ of the permissible leakage rate. In this same instance the allowable leakage established for the reference system was so large as to negate results if the actual leakage were allowed to approach even a small fraction of the allowable. In this test, and some others, the leakage rate test was conducted at a pressure far below the design pressure at which the permissible leakage rate was specified, and the results extrapolated to the higher pressure. Such extrapolation may not be reliable, since leakage which might occur at the higher pressure may not show up at all at lower pressures.

(4) As indicated earlier, a choice can be made as to whether to correct the readings for changes in absolute pressure during the test, or changes in absolute temperature during the test. Using Heap's notation as before, for the reference system

$$w_R = \frac{(p_0 + \Delta p_0) V_R}{RT_{R0}} = \frac{(p_1 + \Delta p_1) V_R}{RT_{R1}}$$

so that

$$\frac{T_{R0}}{T_{R1}} = \frac{p_0 + \Delta p_0}{p_1 + \Delta p_1} \quad (a)$$

also for the vessel being tested,

$$\frac{w_1 - w_0}{w_0} = \frac{p_1 T_0}{p_0 T_1} - 1 \quad (b)$$

If the reference system temperature coincides with the main vessel temperature both initially and finally,

$$\frac{T_{R0}}{T_{R1}} = \frac{T_0}{T_1} \quad (c)$$

From (b) using (a) and (c) then, it is found that the fractional weight change is given by

$$\frac{\Delta w}{w_0} = \frac{w_1 - w_0}{w_0} = \frac{P_1}{P_0} \frac{P_0 + \Delta P_0}{P_1 \Delta P_1} = \frac{\frac{P_1}{P_0} (\Delta P_0) - (\Delta P_1)}{P_1 + \Delta P_1} \quad (d)$$

This can be used in place of II.9, which requires temperature correction.

(5) It is reported that the use of Freon injected as a containment and detection of leaks using a halide detector is of questionable value, since the high sensitivity of this method detects many leaks which are so minute that they cannot be located with soap solution in order to make repairs. It is also implied that leaks so minute need not be repaired. Some leaks which cannot be repaired without depressurization may be plugged temporarily; if a leak cannot be stopped, its rate can be measured and accounted for.

It might be noted here that soap solution tests, or "sniffer" tests, of welds and penetrations cannot in most cases, be utilized except prior to the initial leakage rate test, since the majority of these containment structures are partially or wholly buried, are partially or completely covered with thermal insulation, and may be partially or completely lined with concrete.

(6) In tests on the Dresden sphere, two reference systems were employed and independent readings taken for each system. Thermocouples were located at strategic points on the surfaces of the reference containers for a general check on temperature conditions, also a hygrometer was used as mentioned earlier. The average leakage rate calculated from the two reference systems was 0.016%, $\pm$ 0.014% which is far below specified maximum rate of 0.5%.

In connection with this test an analysis was made to estimate the size of a leak required to realize the maximum specified leak rate. In this analysis it is assumed that the pressure remains at 29.5 psig for 24 hours, that the rate of discharge is 0.5% of the initial stored air in 24 hours, that the temperatures remain constant, that flow is steady and isentropic. The sphere free volume is 3.6×10^6 ft³, containing 275,000 lbs of air at N.T.P. During the test, air is added to bring the pressure to 29.5 psig or 44.2 psia. The new air weight is $275,000 \times (44.2/14.7) = 825,000$ lbs. The stored air = $825,000 - 275,000 = 550,000$ lbs. In 24 hours, 0.5% of this or 2750 lbs will leak out. The flow rate is then $2750/(3600 \times 24) = 0.032$ lbs/sec. The critical pressure is $0.532 \times 44.2 = 23.5$ psia. Since this is greater than the pressure to which the air is being discharged (14.7 psia), the formula for isentropic flow becomes

$$W = \frac{0.532 C A P}{\sqrt{T}} \quad \text{or} \quad A = \frac{W \sqrt{T}}{0.532 C P}$$

Letting $T = 560^{\circ}\text{R}$ and the coefficient $C = 0.7$, the area is

$$A = \frac{0.032 \sqrt{560}}{0.532 \times 0.7 \times 44.2} = 0.046 \text{ sq. inches}$$

corresponding to a diameter of only 0.24 inches.

If this is projected for the leak rate experienced on the test of 0.015%, the diameter is only 0.037 inches.

3.3.5 References - Area III.

1. "ASME Boiler and Pressure Vessel Code." (Am. Soc. Mech. Engrs., N. Y., N. Y., 1956 to date).
2. "API Recommended Rules for the Design and Construction of Large, Welded, Low Pressure Storage Tanks (Tentative)," API Std. 620, First ed., Feb. 1956, (Am. Petroleum Inst., Div. of Refining, N. Y., N. Y.).
3. W. K. Luckow and J. Patterson, "Air Leakage from Semi-Airtight Buildings," Paper No. 1009, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - Sept. 1958, (United Nations, Geneva, to be published).
4. F. T. Binford and T. H. Burnett, "A Method for the Dispersal of Volatile Fission Products from an Accident in the Oak Ridge Research Reactor," ORNL-2086, (1956)
5. A. Heineman and L. Fromm, "Containment for the EBWR," Paper No. 1891, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - Sept. 1958, (United Nations, Geneva, to be published).
6. J. C. Heap, unpublished work.
7. J. Zissi and E. Diehl, "Leak Rate Test - Power Plant Building-West Milton Site," KAPL report M-6065, (Sept. 1953).
8. S. Untermeyer and D. Layton, "Leakage Testing on the Boiling Water Reactor Enclosure," VAL-33, (Sept. 1957).

9. A. H. Heineman, "Suggested Method for Determining Leakage from EBWR Building Shell," unpublished memo. (Feb. 1955).
10. H. J. Metz, "Notes on Helium Leak Detection," CF-58-1-20, (Jan. 1958).
11. R. Brittan, J. Heap and A. Heineman, "Evaluation of Leak Rate Tests," unpublished collection.

3.4 AREA IV - Loads on the Barrier

An extensive amount of information is available on this topic. Sources used result from private and government-sponsored work. Some of the material is obtained from work in fields unrelated to atomic energy; some is for work done in the atomic energy field on problems not directly related to containment; some is from work directly on the containment problems.

Several surveys have been incorporated. In each case the surveyor's references have been accepted without examination by the Editors, and included with the list, in order, at the end of Section 3.

The work is presented in the order prescribed by Section 2 for easy reference.

3.4.1 Mechanical Loads

3.4.1.1 Basic Energy Sources

The basic energy sources leading to the release of reactor contents and the production of mechanical loads, such as a pressure and temperature increase or concentrated loads, are nuclear and chemical. The primary problem is to determine the energy release time history of these sources, the material motions, and the pressures near the release region. These pressures and motions may be restrained by the primary structure or, if failure occurs lead to a further energy transfer.

(1) Nuclear

Methods for determination of the nuclear energy release are available, but they are all predicated on a knowledge of the system's reactivity time history which is strongly dependent upon the specific reactor and not readily generalized. One must have a super-prompt critical excursion in order to develop large and damaging pressures from nuclear release alone. An excursion quells itself by radically altering the density of the system, thereby reducing reactivity. Even a so-called autocatalytic system eventually expands to sub-criticality. The general relations for determining the energy releases, given the shutdown mechanism, are well known and will not be repeated here.

In spite of variation in reactors and in reactivity time history, one is impressed by the similarities in the character of extreme nuclear excursions. A characterization of the general form of the energy history of an excursion as a function of prompt effective neutron lifetime was proposed by Brittan⁽¹⁾ at a conference in May of 1957. The proceedings of this conference were not published, nor are they available to anyone but

participants. Except for this proposal, conference proceedings have since appeared in one form or another in later publications. It seems reasonable for the records to incorporate this proposal here in abbreviated form.

All nuclear energy releases, which in themselves contribute to the containment problem, result from input reactivity excursions into the region above prompt critical. The releases are primarily self-terminated by reactivity reduction which is a function of the energy release.

Regardless of the manner in which reactivity is added, the power-time relationship will have a characteristic peak which reaches a maximum when the reactivity is reduced to prompt critical. The power rises and falls from this peak in an exponential manner. For reactivity reduction, which is a linear function of the energy release, the power spike is characterized by a time width at half height ($1/2 P/P_{\max}$) of about 3.5α where α is the reciprocal of the reactor period. The period is defined as the ratio of power to the time derivative of the power ($P/\dot{P}$). Hence $1/\alpha$ is the time required for the power to change by a factor of e .

An approximation to the power-time relationship based on the exponential rise and the approximate width at half height is shown in Fig. 3-22.

FIG. 3-22
GENERALIZED POWER AND ENERGY HISTORIES FOR A BURST
(REF. 1)

The energy-time relation shown is obtained by integration of the power-time relation. For comparison, a typical experimental power-time trace is shown along with its energy-time integral. Only the energy-time information is important to problems of containment.

Because this is a generalized curve, the time coordinate has been shifted to the power peak location and given in terms of αt e -folding times. The shift is equal to $\ln(P_{\max}/P_0) + 1.057$; so, $\alpha t = \alpha t - \ln(P_{\max}/P_0) - 1.057$. The ordinates have been normalized to give fractions of peak power and total energy release.

Thus, the power curve first fits the expression

$$P/P_{\max} = (P_0/P_{\max})e^{\alpha t},$$

where P_0 is the power at the start of the excursion which is many times smaller than $P_{\max}$. This holds until $P/P_{\max} = 1$. The half width of the truncation is $\alpha t = (3.5/2) - \ln 1/2 = 1.057$ e-folding times. The curve to the right of $\alpha t = 0$ is obtained as a mirror image about $\alpha t = 0$. A very good fit to a typical experimental curve from the value of αt at $P/P_{\max} = 1/2$ to $\alpha t = 0$ is

$$(P/P_{\max}) = 1 - A(|\alpha t|)^{3/2}$$

where

$$A = 1/2 \cdot (1/1.75)^{3/2} = 0.216$$

The "tail" of the true curve is exaggerated. For the assumption that $P_{\max}/P_0$ is large, the tail is unimportant.

The energy-time curve is

$$E(t)/E_{\max} = \int P dt / E_{\max}$$

where $E_{\max}$ is taken as twice the integral of $P dt$ up to $\alpha t = 0$.

$$E_{\max} = 4.1 P_{\max}/\alpha \text{ if } P_0 \text{ is neglected with respect to } P_{\max}$$

and

$$E_{\max} = 3.8 P_{\max}/\alpha \text{ for the true curve shown.}$$

Using these curves, the time-energy relation can be found to a satisfactory approximation for any excursion above prompt critical in any reactor providing $E_{\max}$ and α are given. If the characteristic neutron lifetime of the reactor and the available reactivity above prompt critical are given, α may be obtained from:

$$\alpha = \frac{\text{available reactivity (above prompt critical)}}{\text{lifetime}}$$

The neutron lifetime for thermal reactors is about 100 times larger than that for fast reactors. It is difficult to get α larger than about 200 sec^{-1} for thermal reactors and about $20,000 \text{ sec}^{-1}$ for fast reactors.

The most significant aspect to be considered is that 95% of the energy is released during the time the power level remains above a value equal to $1/10$ of peak power and that this 95% release always takes place

in about 7 e-folding times. 50% of the energy is released in 2 e-folding times. Thus for thermal reactors, the time energy release is ten's of milliseconds and for fast reactors, ten's of microseconds.

The BORAX explosion had an α of about 400 sec^{-1} and a total nuclear energy release of 140 megawatt-seconds. On this basis, the peak power reached is about 1.4×10^{10} watts, and 95% of the total energy is released in about 17 milliseconds. On the basis of a TNT equivalent blast energy release of 1000/cal/g, this corresponds to about 70 lb of TNT. Scale experiments and other estimates indicate that a TNT charge of 10 to 20 lb would have done the same damage.

Various methods of adding reactivity along with detailed differences in the method of shutdown will be encountered in the numerous specific problems with which one must deal. However, for the purpose of examining the containment problem, the time scale for the energy release relationships will be practically unaltered. When only bad accidents are considered, the width at half height for the power curve will not vary by more than about $1/4 \alpha t$, and the time for 95% energy release will vary even less, percentagewise. The proposed relationships can be considered adequate to describe the time-energy behavior of nuclear excursions for the purposes of the containment problem.

An enormous number of kinetic calculations and experiments have been performed for every reactor, too many to summarize completely here. Some representative examples are the work by Brittan⁽²⁾ where generalized energy release analyses are presented for space-independent, one energy group systems (which form a good model for fast reactors); the basic experiments on such systems, reported for Godiva⁽³⁾ and the associated analytical work done earlier at Los Alamos by K. Fuchs⁽⁴⁾ and by G. Hanson;⁽⁵⁾ the extension of the Godiva analyses to space-dependent calculations for condensed systems, taking into account motion of material, as coded for the IBM 704 as AX-1 code by Stratton et al.⁽⁶⁾

The work by H. Bethe⁽⁷⁾ on maximum energy release for sodium cooled intermediate reactors and R. O. Brittan⁽⁸⁾ for the ZPR-III hazards analysis are the first attempts at analyses of originally uncondensed systems. These led to a space-dependent analysis of uncondensed fast systems by H. Bethe and G. Tait⁽⁹⁾ which has since been extended and modified by V. Jankus.⁽¹⁰⁾

The kinetics experiments initially performed on BORAX by ANL and later extended to SPERT by Phillips, and those on KEWB by NAA-AI have provided the experimental basis and stimulated numerous analytical studies of details of the shutdown mechanism for thermal reactors. The Borax data was analyzed by J. Ulrich⁽¹¹⁾ to yield relations between pressures, energy release, and reactivity. The most significant outcome of this latter

study on the water reactor is that the pressures developed before nuclear shutdown are proportional to the second derivative of the reactivity. For the fast reactor, this same relation is also implied in the work of G. Hansen⁽³⁾ and has later been shown analytically by V. Jankus.⁽¹⁰⁾

The results of fast reactor studies show a high dependency on the form and substance of the state equations for the core material. These are not well known, especially when considering the extremely severe excursions of interest in containment problems. It is important that the reader realize that excursions which are not severe enough to rupture the primary containment system (consisting of the reactor pressure vessel and coolant system) are of no interest here. Furthermore, a point often overlooked is the existence of an upper limit on the theoretical maximum nuclear energy release for given classes of systems. This upper limit may be determined as a function of the rate at which reactivity can be added, the strength of the shutdown system, and the lifetime of the particular reactor.

The shutdown mechanism depends on the material motion which produces density decreases, size increases and displacements. This motion which is characterized by the mechanical and thermal inertia. If the reaction rates are very high (short periods or e-folding time) heat cannot be transferred sensibly until higher temperatures are reached where the radiation becomes a significant mechanism. The mechanical inertia does not appear to be important for e-folding times >50 microseconds and thermal inertia is sufficient to prevent significant heat transfer for e-folding times <2000 microseconds.⁽²⁾ Reaching the former limit restricts the consideration to fast reactors exclusively. This latter limit can be reached only with great difficulty in a thermal reactor.

Jankus⁽¹⁰⁾ develops a method of analysis of the nuclear energy released and pressures developed in a fast reactor which is valid for violent excursions. Excursions investigated in this analysis are assumed to start at low power level so that there is no appreciable heating for several periods. It is also assumed that the reactivity is above prompt critical and that the period is so short that mechanical inertia causes the expansion to lag the heating. In addition it is required that a sufficient amount of energy, Q^* , be generated to expand the system to homogeneity (until voids are filled) before pressure is experienced. It is further assumed that large rates of addition of reactivity are possible.

Jankus shows that the pressure is proportional to the second time derivative of the reactivity; and that the ratio of total energy to Q^* is a sensitive function only of the size of the core, the equation of state of the material, the maximum reactivity reached, the lifetime, and Q^* . The effect on results of the approximations used is discussed in some detail, and the results of a few calculations using the IBM 704 AX-1 code mentioned

previously are included as corroboration in the discussion. The more detailed calculations show that the analysis of Jankus yields pessimistic results by factor of as much as 1.5. It is pointed out that energy available for work is always less than total energy developed by at least Q^* .

Some of the calculated results for a typical fast reactor are, in part, shown in the following table. The assumptions for Q^* and the equation of state are somewhat arbitrary, and the results shown here are meant only to show variation with reactivity insertion rate.

(dk/dt) /sec	Total Energy - Q*		$k_{\max}$ $\times 10^3$	Total Energy 10^{16} ergs for Q* = 0.35×10^{16} ergs	Max. Press. Megabars
	Q*				
10	0.84		0.55	0.65	0.064
50	1.56		1.26	0.91	0.119
100	2.2		1.81	1.13	0.167
500	5.9		4.16	2.4	0.45
1000	10.3		5.96	4.0	0.78

k = reactivity above prompt critical

t = time

Q^* = pressure threshold energy (see above)

Most recent and more detailed studies for fast systems are reported by Stratton et al.⁽⁶⁾ using an IBM 704 code (as noted earlier). This more general method involves integrating the fission energy source term, equations of motion, of state, etc., stepwise in space and time during the history of the nuclear burst.

The method is applied to an analysis of the Godiva prompt burst experiments. The experimental results, covering the excess reactivity range up to ten cents over prompt critical, include total fissions, the maximum fission rate, and the width of the burst in time at one-half the maximum value of the fission rate. The theoretical results match the experimental results satisfactorily. The calculation will give (in addition to the above quantities) internal temperatures, pressures, material velocities, and fission rates as a function of radius of the assembly and time during the burst. In particular, the calculation can predict the energy of motion of the material during the burst. This quantity is a better measure of the explosive energy than is the total fission energy generated during the burst.

The method is extended, for hypothetical bursts and several reactivities, to bare assemblies of lower U^{235} enrichment and to various reflected assemblies. Simple rules are described which enable one to scale burst yields, kinetic energies, etc., from one assembly to another.

To analyse conceptual fast reactor accidents, two spherical models are used. These are: (1) a homogeneous core in which all core material is assumed to be molten and at a relatively low density, and (2) a heterogeneous system in which the uranium is assumed to be in the form of spherical shells each of which is separated by low density material or void. In case (1), a nonlinear equation of state is appropriate, and the concept of "threshold" is important. In case (2), a normal equation of state is applicable.

The analytic theory is applied to the first model. This theory shows a strong dependence of the explosive energy release on the "free" energy, Q , which can be produced without causing disassembling accelerations. For $Q = 0$, the total energy release varies as the cube of the excess reactivity. In the limit of large Q , the explosive energy varies as the $3/2$ power of the excess reactivity and is generally much larger than the total energy for a comparable system with $Q = 0$.

The differential calculation is applied to both spherical models. In this case the motion of various portions of the core and reflector is followed in detail. A general survey is described in which the various parameters of importance to conceptual accidents are isolated and described.

Comparison to the analytic theory is made for a few special cases.

(2) Chemical Energy

It should be noted that most thermal reactors providing useful powers cannot conceivably have severe enough nuclear energy releases to alone be significant to our problem. However, even the less severe nuclear energy releases may serve to put the core metal in a state where conditions are appropriate for a very rapid metal-coolant reaction to take place. In other words, if and only if the metal can heat up and pass through the liquid phase can significant metal-water reactions take place.

The principal difficulty with regard to metal-water reactions as basic energy source is the fact that data is not available on reaction rates in terms of the important parameters of the problem. Nearly all the metal-water reaction experiments have been concerned with % completion of reaction, rather than determination of the relation between basic reaction rates and the fundamental parameters. This has led to a basic study of rates by a group at ANL under R. C. Vogel. Fortunately, as part of the job of planning their work, a literature survey was made by L. Baker and C. H. Smith and recorded in unpublished form. The methods used, parameters measured, advantages and disadvantages, and results are all presented. The information on methods and results are recorded here. The unclassified portion of their reference bibliography has been added to that at the end of this section.

The purpose of this survey is to summarize and to criticize both the experimental methods that have been used to study metal-water reactions and the results of these studies.

Thermodynamic data indicates that most of the metals commonly used in reactor cores have the potential to react with water with the liberation of considerable heat. The amount of energy released by such a reaction can be comparable to that released by the fission process during a power excursion.

This literature survey has been limited to the following metals and certain of their alloys normally used or considered for use as nuclear reactor core materials:

1. Zirconium
2. Aluminum
3. Stainless Steel
4. Uranium

Studies dealing only with low temperature steam corrosion have been omitted.

Molten metal-water reactions fall into the category of rapid chemical reactions. The most critical requirement of an experimental method to determine rates of rapid reactions is as follows:

1. The reactants must be brought together at the reaction temperature in a time which is short compared to the period of observation.

Either of two methods may then be used to follow the reaction:

- 2a. Some index of the extent of reaction can be measured continuously as a function of time, or
- 2b. The reaction can be quenched after a period of time and the extent of reaction determined.

It is often impossible to carry out rapid reactions under isothermal conditions. A given experimental method must then be evaluated in terms of how well reaction temperature is controlled.

Molten metal-water reactions are heterogeneous so that the control of interfacial area must also receive important consideration.

In addition to the controls that the experimental method imposes on the reaction, it is necessary to consider what measurement techniques are used to define the reaction conditions of temperature and pressure and to follow the extent of reaction. The measurement techniques must be judged in terms of their accuracy and time response.

The experimental methods that have been used can be divided into six broad categories as follows:

- I. In-pile heating in MTR
- II. Slow heating of metal immersed in water or steam.
- III. Pouring or spraying molten metal into water.
- IV. Molten metal drops falling into water.
- V. Dispersion of molten metal by an explosive charge.
- VI. Dispersion of metal by a condenser discharge.

Each category will be discussed separately. A final summary of results is included.

I. In-Pile Heating in MTR

References

Phillips Petroleum Company^(12,13,14)

Westinghouse^(15,16)

Method

Heating - The metal under study was either alloyed with U^{235} or formed as a clad around U^{235} fuel, placed in an autoclave and then immersed in the neutron field of a reactor for a specified period of time (6-15 seconds).

Contacting - The metal was either surrounded by water the entire time or surrounded by water vapor and allowed to flow or drip into liquid water.

Time of heating/contacting - It was estimated to require 0.1 - 0.2 seconds to reach the melting point.⁽¹⁵⁾ Heating continued, however, for the entire period of immersion.

Reaction temperature - Largely unknown. The temperature could be determined only from a heat balance. It depends upon heat loss rates from the metal to the autoclave materials and upon the rate of chemical reaction.

Interfacial Area - Completely unknown.

Results

Aluminum

- a. Samples that did not melt gave no reaction.
- b. Molten samples reacted.
- c. Sporadic explosions were obtained only with samples that reached temperatures slightly above the melting point.

Zirconium

- a. Molten samples gave reactions up to 80 per cent completion with some sporadic explosions.

Nichrome

- a. Molten samples gave up to 30 per cent reaction with sporadic explosions.

Discussion of Results

- a. The sporadic explosions and pressure surges may have been due entirely to localized overheating and the release of gas pockets. It is not at all certain that the apparent pressure transients are related to metal-water reactions. It is also likely that some of the transients were due to electrical disturbances since the authors report considerable difficulty with the instrumentation.

II. Slow Heating of Metal Immersed in Water or Steam

References

Battelle Memorial Institute⁽¹⁷⁾

WAPD⁽¹⁸⁾

Method

Heating - An induction furnace was used to heat a supported solid Zircaloy-2 sample^(17,18) and to heat a molten Zircaloy-2 sample contained in a graphite crucible.⁽¹⁷⁾

Contacting:

- a. The specimens were mounted inside a Vycor tube and protected by an argon gas blanket during the heating period. When the solid sample reached the reaction temperature, 50 psia steam was passed through the reactor, and when the molten sample reached the reaction temperature, a jet of 50 psia steam was directed upon the surface of the molten Zircaloy-2
- b. The samples were heated to the desired temperature in an initial water atmosphere (18)

Time of Heating/Contacting

- a. Steam was present in the solid sample system from 0.08 to 120 minutes and was present in the molten sample system from 1 to 25 seconds (17)
- b. The sample was heated to the desired temperature in 5 to 10 seconds. The contacting period was the time required to collect successive 250 ml volumes of hydrogen and the time periods were from 10 seconds up to 55 minutes. (18)

Reaction Temperature

- a. The temperatures of the solid specimens and of the melts were measured initially. There is some doubt as to what the reaction temperature is because of the rapid chemical reactions of the steam with the molten samples and due to the increased heat losses from the solid sample with a steam surroundings rather than an argon gas atmosphere. (17)
- b. The sample temperature tended to remain constant during the reaction period and ranged from 1300 to approximately 1860 C.

Interfacial Area - Initially the interfacial areas for both solid and molten samples are known accurately. The interfacial area of the sample during melting was probably quite different from the initial area (18)

Results

Zircaloy-2

Battelle Memorial Institute⁽¹⁷⁾

- a. Reactions below 1690 C and reaction times between 5 seconds and 2 hours were best correlated as:

$$v^2 = 0.1132 \times 10^6 \theta \exp [(-34,000 \pm 1440)/RT]$$

with v in units of ml of hydrogen per square centimeter and θ in units of seconds.

- b. The reaction rate between molten Zircaloy-2 and steam does not correlate with the initial temperature. However, the reaction rates are higher than those predicted from the extrapolation of result a.
- c. Pressure appeared to have little effect between 20 and 50 psia.
- d. Between 4 and 13 per cent of the hydrogen formed in the reaction of steam with Zircaloy-2 was absorbed in the sample.
- e. As the reaction temperature was increased, the ratio between the thickness of the zirconia layer and the alpha-phase layer decreased.

WAPD⁽¹⁸⁾

- f. Reaction rates were fast at the temperatures used, but were not explosive or violent even at temperatures above the melting point.
- g. The oxidation rates of Zircaloy-2 near its melting point in water were approximately the values that would be obtained from the extrapolation of the oxidation of zirconium in air data.
- h. The scales formed during the oxidation were essentially the same in appearance and order of formation as those formed during the scaling of zirconium in oxygen.
- i. Small amounts of uranium and boron do not affect the high temperature oxidation rates of Zircaloy-2 in water.

Discussion of Results

The Battelle reaction rate data⁽¹⁷⁾ for Zircaloy-2 steam are lower than WAPD data⁽¹⁸⁾ for Zircaloy-2 water reaction rates at the same time and temperature. The Battelle data are probably lower because of the uncertainty of the reaction temperature. The comparative data imply that the WAPD reaction mechanism is a metal-steam reaction.

This method is good for corrosion studies, but is not applicable for molten metal-water reactions.

III. Pouring or Spraying Molten Metal into Water

References

Aerojet(19,20,21)

Aluminum Co.(22,23,24)

Argonne(25)

Mine Safety Appliances(30-34)

Method

Heating - Bulk metal samples were heated in crucibles.

Contacting - The metals were then sprayed⁽²¹⁾ or poured into liquid water.

Time of Heating/Contacting - There is reason to believe that traces of air or water vapor were in contact with the bulk melt for extended periods of time; however, the principal contact between fresh metal surface and water is made as the stream breaks up beneath the water surface. Contacting time for most of the surface is, therefore, very small.

Reaction Temperature - The initial temperature of the molten metals were accurately known. Rapid cooling occurred in every case; however, no attempts were made to estimate or measure the cooling rates.

Interfacial Area - Largely unknown. Reasonably accurate estimates were obtained⁽²¹⁾ where the spray formed spherical particles that were not entirely oxidized. It was assumed that the initial particle size did not change throughout the reaction. Measurement of particle size thus led to an estimate of the total interfacial area. The other experiments did not produce well-defined particles so that no accurate estimate of the area was possible.

Results

Aluminum

- a. Aluminum Company(22,23,24) showed that explosions sometimes occurred when 50 lb batches of aluminum were discharged into water. Conditions leading to explosion are summarized as follows:

1. Crucible exit hole diameter must be greater than $2\frac{3}{4}$ ".
2. Metal drop distance must be less than 10 feet.
3. Water depth must be less than about 30 inches and more than one inch.

Paint or grease on the bottom of the water container was effective in preventing explosions. The authors concluded that aluminum-water reactions were not involved. The observations were explainable in terms of the rapid generation of steam. The absence of a flash indicated also that the H_2-O_2 reaction was also not involved.

- b. Argonne(25) found no reactions and very minimal steam production using fine jets of molten metal.
- c. Aerojet(19,20) and Mine Safety Appliances(26-30) found no reaction when molten aluminum was dropped into water without additional dispersion by a blasting cap.
- d. Aerojet(21) in the "explosion dynamometer" obtained negligible reaction below 1200 C. Above an initial metal temperature of 1200 C, evidence of strong reaction was obtained. The 1200 C temperature was beyond the range of any of the other investigators who used pouring methods. Aluminum at 2070 C gave a stronger pressure pulse than Zr, stainless steel, NaK or black powder.

Zirconium

- a. Aerojet(19,20) found black glass-like coatings (.005 inch) after pouring 454 grams of molten Zr into water. Calculations, considering the film thickness and an average droplet size, indicated that 16 per cent of the metal was oxidized before the metal was quenched.
- b. Aerojet(21) in the "explosion dynamometer" (spraying) found that strong explosions occurred with zirconium over the temperature range 1900 - 2600 C. There was no

clear-cut effect of temperature. The individual drops were collected and the thickness of the oxide layer determined. The extent of reaction varied from complete for small (10-20 micron) drops to about 50 per cent for 100 micron drops to about 20 per cent for 1000 micron drops. These results are approximately what one would obtain if 25 microns of metal had reacted uniformly from the surface of each drop.

- c. Mine Safety Appliances (26-30) found that zirconium reacted up to 60 per cent in 750 psi steam and to a lesser extent at lower pressures. Considerably less reaction was noted when the zirconium was dropped into room temperature water.

Zircaloy-2 and Zirconium + 1% Beryllium - Results were identical to zirconium within the rather large experimental error.

Stainless Steel, Type 303 - One run was reported in the Aerojet explosion dynamometer (21) in which a strong reaction was reported at 2270 C.

Nickel - Nickel at 1900 C gave essentially no reaction in the explosion dynamometer. This observation constituted an important verification of the method.

Uranium - A single measurement in the explosion dynamometer of uranium at 1766 C showed that a very mild reaction occurred.

Discussion of Results

The results obtained with the explosion dynamometer are qualitative in nature; nevertheless, they show that explosive reactions can occur and define the required conditions rather well. The nature and rapidity of the reactions are illustrated in Fig. 3-23. The rate of rise of the pressure is necessarily limited by the rate of injection of the molten metal so that even faster reactions are likely to occur under different conditions. In spite of the apparent violence of the reactions, it seems well established that the reactions are not self-sustaining, do not go to completion and are not ignitions. It was also likely that no hydrogen-oxygen reaction was involved since "de-gassed" water was used and no air space was available to the hydrogen.

FIG. 3-23
RESULTS OF AEROJET EXPLOSION DYNAMOMETER TESTS
(REF. 21)

IV. Molten Metal Drops Falling into Water

References

Battelle⁽¹⁷⁾

Method

Heating - One end of a supported rod was heated inductively until a single drop fell.

Contacting - The drop fell through an inert atmosphere into a water vapor-argon mixture and finally into a 25 foot column of liquid water.

Time of Heating/Contacting - The metal was probably in contact with some water vapor during the entire heating operation. It was only in contact with nearly pure steam for about 50 msec while in flight before entering the liquid.

Reaction Temperature - The initial temperature of the metal was presumed to have been the melting point. The time of fall through

the water column required to quench luminosity was recorded. If it could be assumed that the end of luminosity corresponded to a definite temperature, then a qualitative time-temperature history was obtainable.

Interfacial Area - The use of single drops made accurate area estimations possible. The drops after reaction had a characteristic cup-shaped indentation on one side which complicated the situation.

Results

Zirconium

- a. The time to the end of luminescence rose from 2 to 10 seconds as the initial water temperature was raised from 90 to 200 F. The corresponding depth of fall through the water ranged from 8 to 24 feet.
- b. 39 to 45 per cent of the zirconium in nominal 0.2-inch drops reacted although no particularly rapid or explosive reactions occurred.
- c. The observed oxide layer thicknesses corresponded approximately to those extrapolated from solid metal-steam reaction rate data.

Discussion of Results

The fact that the drops initially were at the melting point and possibly not fully melted could mean that the drop surfaces were molten for only a short time. This severely limits the value of the data.

V. Dispersion of Molten Metal into Water by an Explosive Charge

References

Aeroproject General Corporation^(19,20,21)

Method

Heating - An induction furnace was used to melt the sample to be studied in a bottom-tapped crucible.

Contacting - The sample was dropped into the water as soon as possible after reaching the melting temperature. The molten metal was dispersed by a blasting cap aligned directly beneath the outlet of the crucible and mounted 5 inches below the surface of the water. Protection of the metal from premature oxidation was done by a purified argon gas blanket during the heating cycle.

Time of Heating/Contacting - The molten metal stream was dispersed into the water instantaneously by the explosive charge.

Reaction Temperature - Largely unknown. The temperature of the melt was measured, but the reaction temperature could be determined only from a heat balance. It depends upon the heat loss rates from the metal to the surrounding water and upon the rate of chemical reaction.

Interfacial Area - Particle-size analyses of the recovered residue were made. Measurements were made, using a filar eyepiece in the metallograph of the thickness of the oxide layer and the diameter of the metallic core. These measurements determined the original droplet diameters before oxidation.

Results

Aluminum

- a. There was very little reaction when the molten drops were comparatively large. However, when the metal was dispersed, the estimated percentage of reaction was 75 per cent at a temperature of about 1565 C.
- b. The importance of temperature was shown. Aluminum was found to be essentially non-reactive at 930 C, but reacted explosively at 1565 C.

Uranium

- a. The residue was found to be partially atomized (approximately 8 per cent by weight for particles less than 140 microns in diameter), but the bulk remained as large, irregular platelets covered with a surface oxidation.
- b. The percentage of reaction varied from 30 to 50 per cent. There were indications of slight explosions in a few of the tests.
- c. The reaction of molten uranium with water was neither violent nor self-sustaining at a temperature of 1540 C.

Zirconium

- a. All or nearly all of the metal reacted violently in every case.
- b. For one particular case, sparks were thrown 50 feet into the air. The temperature was approximately the melting

point of zirconium (1830 C). From the equipment damage, a stress calculation indicated that a pressure of at least 2200 psi would be necessary to produce the effects noted. Then assuming 100 per cent reaction, the 454 grams of zirconium produced 235 liters of hydrogen and released 214 kilocalories of heat.

Discussion of Results

The validity of using a blasting cap for dispersal of the molten metal was obtained by comparing drop test runs with the explosion dynamometer (spraying) test runs. The curves were similar, and it was concluded that the energy added by the blasting cap (800 gram calories) was minor. It was hypothesized from this that all mechanisms for obtaining the dispersion of molten metal would probably lead to the same results, assuming an identical particle size distribution.

The rate of pressure produced by the reaction of molten metals with water is considerably less than those obtained from high explosives. The shock waves produced, however, make the problem of containment difficult.

In analyzing the Aerojet data it is suggested that the complete tabulated results be resorted to, since in their illustrations only a few representative experimental values are shown.

VI. Dispersion of Metal by a Condenser Discharge

References

ANL(31)

Bureau of Mines(32)

Columbia University(33,34)

KAPL(35)

NAA(36,37)

Fundamentals of fusing wires(38,39,40,41,42)

Method

Heating - The metal under study is in the form of a wire or ribbon and is heated rapidly to virtually any desired temperature by a momentary surge current from a bank of condensers or from a short-circuited DC generator.(33,34)

Contacting - The wire or ribbon is immersed in either steam or water throughout the heating process.

Time of Heating/Contacting - A considerable range can be achieved by variation of the electrical parameters in the discharge circuit.

- a. Several studies of exploding wires as high intensity light sources^(40,41) and fundamental studies of the physics of the disintegrations process^(38,39,42) have used very short heating periods (1-10) microseconds.
- b. Periods of the order of 100 microseconds were used by several investigators to study metal-water reactions.^(31,32,37)
- c. Slow discharges (3-50 milliseconds) were used^(33,34,35) to study metal-water reactions.

Reaction Temperature

- a. Rapid heating methods (1-100 μ sec) have produced temperatures up to 100,000 C.
- b. Slow heating methods (3 msec or longer) are limited to the melting points of the metals since electrical contact is broken.

Interfacial Area - Largely unknown. The particle size depends both on the rate of heating and on the amount of energy supplied and probably on the wire diameter.

- a. Rapid heating and high energies - the metals and/or metal oxides were reduced to colloidal particles.^(31,37)
- b. Rapid heating and low energies - A mixture of fine and coarse particles were obtained.^(32,37)
- c. Slow heating (20-50 msec) and moderate energy input⁽³⁵⁾ gave coarse particles. Energy inputs below that required for complete fusion caused the wire to part in only a few places.

Results

Aluminum

- a. Plott⁽³¹⁾ concerned himself exclusively with aluminum. It would appear that he obtained complete reaction every run,

however, he used more than sufficient energy to vaporize the metal. His results thus are of little use in evaluating kinetics of molten metal-water reactions.

- b. Data from NAA⁽³⁷⁾ are plotted in Fig. Per cent reaction as measured by hydrogen evolution as a function of the energy input was presented in tabular form. The values of energy input are subject to considerable uncertainty. It appears that significant reaction occurs only when enough energy is used to melt the metal.
- c. Columbia University results⁽³⁴⁾ indicated that aluminum is non-reactive at the melting point.

Aluminum + 3.5% Lithium Alloy - Data from NAA⁽³⁷⁾ makes it appear that the addition of lithium suppressed the reaction. This conclusion cannot be accepted without further proof. It is likely that more uncertainty exists in the energy input calculation in the case of the alloy since the electrical resistivity may not be known accurately.

Zirconium

- a. Data from NAA⁽³⁷⁾ leads one to the conclusion the molten zirconium will react considerably under these conditions.
- b. KAPL⁽³⁵⁾, using the slow discharge method found that large zirconium pieces and the balled ends of the wire continued to "burn" for some time after completion of the energy pulse.
- c. Columbia University⁽³⁴⁾ reported that a water reaction with zirconium occurred during the 0.8 to 2 milliseconds required to quench glowing fragments. A tendency to self-sustained reaction was noted.

Uranium

- a. The data of NAA⁽³⁷⁾ indicate that appreciable reaction can occur.
- b. Columbia University⁽³⁴⁾ reported that no evidence of self-sustained reaction was found.

Discussion of Results

The data of NAA⁽³⁷⁾ are the only quantitative metal-water reaction data that have been obtained by the condenser discharge method. No attempt was made to obtain rate data as distinguished

from measurements of the extent of reaction. Extent of reaction data, however, can be used to estimate rates if estimates of the total surface area and the quenching time can be made.

The work of NAA⁽³⁷⁾ is an excellent beginning to the study of metal-fluid reactions. Work is required to improve the accuracy of the measurement of energy input and particle size. Transient pressure and temperature measurements are required to obtain rate data.

Summary of Results

Aluminum

The only reports of aluminum reactivity at or slightly above the melting point resulted from the in-pile measurements and the exploding wire studies of Ruebsamen, *et al.* In both cases, only crude estimates of temperature could be made. The reported pressure surges obtained with aluminum in the in-pile studies were not necessarily indicative of aluminum-water reaction. The pouring and spraying experiments indicated strongly that aluminum below about 1200°C does not react appreciably. The explosion dynamometer work and the experiments with explosive dispersion showed that aluminum reacts strongly above 1200°C. Aluminum at 2070°C gave a stronger explosion on being sprayed into water than did an equal quantity of zirconium at the same temperature.

Uranium

The data on uranium are much more limited than that on aluminum or zirconium. Indications are that only surface corrosion occurs at temperatures up to 1700°C. Only mild reaction was noted in the explosion dynamometer and explosive charge methods. The exploding wire experiments show that uranium can react almost completely if a high degree of subdivision is obtained.

Stainless Steel

A single experiment in the explosion dynamometer indicated that stainless steel was similar to zirconium in its behavior toward water.

Zirconium: (Zircaloy)

The data and studies have shown that solid Zircaloy-2 follows the parabolic law approximately for both liquid water and steam up to the melting point. The reaction with water (steam layer) is somewhat more

rapid than that with steam; however, neither is very rapid nor violent. No evidence was reported in any of the studies of an ignition followed by rapid consumption of the metal such as occurs with most metals in oxygen. All of the data including the exploding wire studies, the single drop experiments, and the pouring and spraying work are consistent with the idea that rates of the molten metal reactions at temperatures above the melting point may be obtained qualitatively by extrapolating the low temperature data obtained with solid specimens. The relatively greater reaction obtained by the spraying, explosive dispersion, and exploding wire methods are probably due only to the greater interfacial area for reaction.

The basic metal-water reaction studies, especially those of Bostrom,⁽¹⁸⁾ have been applied to estimate the additional possible heat source and hydrogen evolution. A particularly complete treatment was rendered for PWR by Lustman.⁽⁴³⁾ This is summarized here in some detail to make readily available the methods and basic data, and the arguments used to lower the usual over-pessimistic maximum energy yields.

The following conclusions were drawn in the report for the PWR loss of coolant accident:

- a. Under conditions of zero heat loss, a self-sustaining reaction between Zircaloy-2 and steam is impossible.
- b. Quite moderate rates of cooling are sufficient to prevent heating-up due to the reaction particularly at low temperatures.
- c. The amount of reaction which occurs in the seed plates up to the melting point is less than 8% and diminishes with heating rate. The total amount of reaction which occurs during both heat-up and meltdown varies between 11 and 13%.
- d. The amount of reaction between molten Zircaloy droplets and water is between 13 and 15%.
- e. The amount and rate of reaction of Zircaloy with water is dependent on the emissivity of the zirconium surface and on the length of time the reaction has been taking place.

The metal is attacked at a diminishing rate until the "breakaway" point is reached. Then the rate of attack becomes constant. The "breakaway" point is dependent on temperature. For the first several minutes of a reaction the oxygen weight gain, W , is best represented by

$$W = K(T)t^n(T)$$

where

W is grams of oxygen weight gain per cm^2 of surface.

t is time in seconds.

K and n are functions⁽⁴⁴⁾ of absolute temperature T.

The heat evolved in the reaction

is tabulated for various temperatures.

Temperature (°K)	Ref. (44)		Heat of Reaction in Calories Entering Solid Phases per Gram of Oxygen Reacted
	K (T)	n (T)	
1200	4.15×10^{-4}	0.38	3920
1400	6.31×10^{-4}	0.462	3740
1600	9.6×10^{-4}	0.552	3580
1800	1.52×10^{-3}	0.66	3410
2000	2.5×10^{-3}	0.784	3230
2200	4.36×10^{-3}	0.945	3040

Heating curves for the Zircaloy in the core, Fig. 3-24, were calculated for no heat applied, 1 cal/gm/sec applied, and 0.1 cal/gm/sec applied. An initial temperature of 1000°K was assumed. The basic expression for the time needed to raise the temperature 200° C is:

$$(200^\circ \text{C})(7.27 \text{ cal/mol } ^\circ \text{C})(\text{wt. of half-section mols}) =$$

$$(\text{Decay heat cal/gm sec})(\text{wt. of half section gm})(t \text{ sec}) +$$

$$(\text{heat cal/gm } (\text{O}_2))[K(T)(t + t_0)^n - K(T)t_0^n]$$

Only nominal amounts of cooling are found necessary to prevent progressive oxidation and thus prevent melting. For example, cooling rates of $1.05 \text{ cal}/(\text{cm}^2)(\text{sec})$ are sufficient to prevent heating above 1400°K by 1 cal/gm/sec of applied heat.

The amount of reaction occurring during meltdown, Fig. 3-25, was determined by equating the heat of reaction per gram of oxygen at 2200°K to the heat of fusion of the half section, 5500 cal/mol.

Note that the total amount of hydrogen produced amounts to about 0.02 liters/ cm^2 of surface at STP. If 10% of the surfaces have reached melting, approximately 7000 liters of hydrogen at STP will have been released

FIG. 3-24
HEATING CURVE FOR CORE WITH VARIOUS AMOUNTS OF
ADDITIONAL HEATING (REF. 43)

FIG. 3-25
OXIDATION CURVE OF SPHERICAL PELLET OF 0.119 CM RADIUS
AT 2200 °K CORRECTED FOR DIMINUTION IN VOLUME (REF. 43)

The molten Zircaloy will react when it falls from the core into the water. As the droplet of Zircaloy oxidizes it will lose heat almost entirely by radiation into what can be considered a perfect heat sink. The equation for determining the time for a 200°K temperature drop is obtained by the following energy balance:

$$\begin{aligned}
 &(\text{Energy radiated over each } 200^\circ\text{K interval}) = \\
 &(\text{sum of energy supplied by oxidation reaction}) + \\
 &(\text{energy supplied by } 1 \text{ cal/gm of decay heat}) + \\
 &(\text{energy required to change temperature } 200^\circ\text{K})
 \end{aligned}$$

The oxygen weight gain equation must be corrected for the change in radius of the drop during oxidation:

$$W = Kt^n 4\pi(r_0^3 - W \times 0.1048)^{\frac{2}{3}}$$

The calculations indicate that the amount of reaction is very sensitive to the value assumed for the emissivity of the droplet (Fig. 3-26). The

FIG 3-26
COOLING CURVE FOR MINIMUM DROPLET IN
WATER AS FUNCTION OF EMISSIVITY
(REF 43)

emissivity of an oxide-coated zirconium droplet lies in the range about 0.7 which is relatively large. Calculations of the amount of reaction using emissivities of about 0.7 are in agreement with experiment.

For a reaction rate constant with time, the amount of reaction is inversely proportional to the square root of the drop radius (Fig. 3-27).

In addition to the above analysis for PWR, Lustman⁽⁴³⁾ also reviewed much of the work noted in the preceding ANL survey, reconciling the results with the findings of his analysis. This review is presented here in detail to augment the survey already presented.

Reactions Under Conditions of Rapid Heatup⁽⁴³⁾

The preceding analysis is applicable primarily to loss of coolant accidents where heating and resultant meltdown occur relatively slowly

as a result of decay heat. It was shown that under such conditions the amount of reaction obtained is predictable and is limited in extent. However, another class of accident conditions can be conceived as resulting from nuclear run-away incidents which involve extremely rapid heating to reaction temperatures.

FIG. 3-27
PERCENT REACTION VS. DROPLET
DIAMETER (EXPLOSION DROP TESTS
MOLTEN ZR IN WATER) (REF 43)

actants can be brought in contact and by the rate of removal of the gaseous hydrogen product and metallic oxide. Thus, it was predicted that for a considerable range between the melting point and the boiling point of zirconium, the amount and rate of reaction would be limited and relatively independent of metal temperature. As the zirconium temperature is raised still higher to the boiling point, the amount and rate of chemical reaction will again increase and would then be controlled by reaction kinetics applicable to homogeneous reactions. These predictions have not been analyzed quantitatively or verified experimentally.

A number of experiments have been performed in which zirconium and other reactive metals have been heated electrically or nuclearly to elevated temperatures in the presence of water. Ruebsamen, Schon, and

Saltsburg⁽⁴⁵⁾ has considered qualitatively some of the factors affecting chemical reaction under such conditions. As the temperature of Zircaloy is raised to the melting point and above, the amount and rate of chemical reactions are controlled by heterogeneous reaction kinetics as discussed in previous sections. As the metal temperature is still further elevated above the melting point, the amount and rate of chemical reaction will probably be controlled by the rate at which the metal and steam re-

Chrisney⁽³⁷⁾ heated aluminum, zirconium, and uranium alloy foils under water by capacitor discharge at rates in excess of $18,000,000^{\circ}\text{C}/\text{sec}$. Depending on the energy input into the samples, more or less complete reaction was noted. At energy inputs below $0.05 \text{ cal}/\text{mg}$ little or no reaction was noted. As the energy input increased, the percentage of reaction in general increased, and at values of 4 to $8 \text{ cal}/\text{mg}$, 100% reaction was noted. These effects may be correlated with the results shown in Fig. 3-27. At low energy inputs the resultant droplets of molten metal cooled relatively rapidly and showed only minor amounts of reaction. As the energy input increased, several effects may have operated to increase the amount of reaction: (1) The initial temperature may be well above the melting point. The oxidation rate at such high temperatures may become sufficiently greater than the amount of heat radiated from the molten droplets so that even under conditions of high emissivity, heatup and more complete reaction will occur. Note that oxidation heating increases exponentially with temperature, while radiation losses increases as the fourth power of temperature. (2) A rapidly heated metal may reach the melting point before an oxide of high emissivity has an opportunity to form on the surface. Thus heating due to oxidation and consequent complete reaction will occur when the emissivity has a low value characteristic of the metal. By the time sufficient oxide has formed to increase the emissivity, almost complete reaction will have occurred. (3) As noted in Fig. 3-27, the smaller the particle size, the more reaction will tend to occur. Thus, if with higher initial energy densities the state of subdivision of the particles increases, the more complete will be the reaction even under conditions where the reaction will not be self-sustaining. (A self-sustaining reaction is defined as one in which the metal temperature rises as a result of the reaction to a temperature above its initial temperature.)

Layman and Mars⁽³⁵⁾ heated zirconium foils under water at one atmosphere total pressure and at heating rates of about $3700^{\circ}\text{C}/\text{sec}$. The progress of the reaction was recorded by high speed motion pictures. It was noted that considerable reaction of the zirconium particles occurred after interruption of the current and while the particles were still suspended in the water. These results may readily be reconciled with those shown in Fig. 3-27. It may be noted that during the filmed time interval, one to two seconds, considerable reaction is still taking place during cooling of the particles. With an assumed emissivity of 0.8, the time required to cool to 1000°K would probably be about 5 sec, and the time in the elevated temperature range during which most reaction occurs would be about 1 to 2 sec. It is significant that over-all amounts of reaction of 11% and 25% were noted for the only two runs measured. These amounts of reaction are within the range of those reported in Fig. 3-27. It is also expected, from comparing the amounts of reaction occurring in the fine particles with those of coarser particles in Fig. 3-27, that very fine particles of zirconium would show more complete reaction than those of coarse particle size. One would expect the

residue from such a test to consist of particles of almost completely oxidized material as well as particles of incompletely oxidized metal. It is significant that Layman reported that certain of the particles were metallic but contained large amounts of oxygen. This is evidence of the internal diffusion of oxygen into the zirconium.

Lorentz⁽¹⁵⁾ heated zirconium-uranium alloys containing 4 to 40 wt % U^{235} by fissioning in the MTR at estimated heating rates of 750-2800°C/sec. Amounts of reaction ranging from 16 to 84% were noted. Again these results are not inconsistent with those presented in Fig. 3-27. It should be noted that the conditions of these experiments were such that the specimens were heated in small ceramic cups contained in a pressure vessel. It is conceivable, therefore, that the amount of energy lost from the molten samples would be decreased because of the reflectivity or insulating effect of the walls of the ceramic cup and hence the inability of the water atmosphere to absorb the radiated energy. Under such conditions it would be expected that larger amounts of reaction would be observed. This experiment emphasizes the necessity of carefully considering the environmental conditions under which these experiments are conducted.

In summary, these experiments can be interpreted as indicating that no new phenomena are encountered as a result of rapid heating per se. It would indeed be surprising, in view of the results discussed in previous sections, if rapid heating to temperatures above the melting point resulted in appreciably different amounts of reaction than does relatively slow heating to the same temperatures. As discussed above, the reaction kinetics at temperatures above the melting point appear to be characterized by a linear reaction rate; in other words, the presence of previously formed oxide films would not be expected to hinder further progress of the reaction so long, of course, as reaction rate is controlled by heterogeneous surface reaction kinetics. This conclusion appears to be borne out by the results of the experiments described above in which limited reaction is obtained under rapid heating conditions. The principal effect of rapid heating appears to be change in the state of physical subdivision of the metal reactant as a result of the particular experimental conditions employed and hence variable amounts of reaction resulting from the generally uncontrollable nature of such subdivision. It is proposed, therefore, that just as in the case of slow heating discussed in previous sections, the amounts of reaction obtained under conditions of rapid heating to temperatures above the melting point of zirconium but below the boiling point are capable of prediction from reaction rate data and knowledge of the important modes of heat losses so long as the state of physical subdivision of the metal can be satisfactorily characterized.

The question has been raised whether, if melting and reaction of a portion of the PWR seed plates were initiated by some mechanism, such reaction would result in further heating and spread of the reaction. It may

be predicted from the curves presented in Fig. 3-24 that, so long as the seed plates are in contact with water, the reaction will be rapidly quenched by conduction of heat to the water as soon as the external source of heating is removed.

Some qualitative experiments can readily be performed which illustrate graphically the influence of heat loss in preventing progression of the zirconium oxidation reaction. Zirconium foils, 1 to 2 mils thick, may readily be ignited with a match and will continue to burn the length of the specimen if suspended freely in air. Under such conditions sufficient heat is conducted to the metal adjacent to the location of the reaction to raise that portion of the foil's temperature to reaction temperatures. If, however, a cold object is placed even in light contact with this foil at some place along its length, the reaction will come to a halt one-quarter to one-half inch away from the position of contact. Sufficient heat is conducted to the cold body to prevent rise in temperature of adjacent portions of the foil to reaction temperatures. Allison⁽⁴⁶⁾ has performed a number of such experiments in the course of specifying conditions for safe storage of Zircaloy scrap. Allison found that progression of reaction along the foils of various widths and thicknesses was sensitive to both dimensions; in other words, for a given thickness of foil, a maximum width appears to exist beyond which a freely progressing reaction will not occur upon removal of the source of heat. These results are explicable on the basis that the rate of heat removal from the location of the reaction is a function of the cross-sectional area of the material available for heat conduction.

Reactions under Conditions of Dispersal of Molten Reactive Materials in Water

Varying results have been obtained when large masses of molten reactive metals have been dispersed in water. The results range from no observable reaction to quite violent explosions. Even in cases where violent explosions were obtained, it should be noted that complete reaction of the metal with water has rarely, if ever, been observed.^(20,25,47,24) In the experiments of Higgins⁽²⁰⁾ where relatively violent explosions were noted upon dispersal of molten zirconium into water by means of a blasting cap, the analysis of the amount of reaction observed in each dispersed size fraction showed that the course of reaction was predictable from reaction kinetic and heat loss data as shown in Fig. 3-27. Therefore, no new phenomena appear to be encountered under conditions involving reaction of large amounts of dispersed reactive metals other than occur in the cases analyzed previously for the reaction of single droplets. Some of the early results reported by Higgins revealed an apparent anomaly in case of the dispersal of aluminum in water. When such dispersal was performed at or shortly above the melting point of aluminum, little or no reaction was encountered. These results were interpreted by Lustman⁽⁴⁴⁾ on the following basis. The rate of oxidation of aluminum at temperatures under consideration (1500°F) is probably not sufficient to cause appreciable reaction

heating of the particles. Likewise, at such low temperatures, mechanisms of heat transfer other than by radiation through a steam blanket are important and the cooling of dispersed droplets probably occurs primarily by convective heat transfer. It was predicted that at higher aluminum temperatures, violent explosions should occur in case of aluminum as in case of zirconium. This prediction was indeed confirmed in recent experiments of Higgins⁽⁴⁸⁾ who showed the violence of the explosion attending the dispersal of aluminum in water continuously increased as the aluminum temperature was raised above 1200°C.

The observations by Russel,⁽²⁴⁾ and West and Weills⁽²⁵⁾ that the aluminum reaction may be quenched by suitable dispersal of the molten metal stream is consistent with the efficacy of radiation in extraction of heat. A well-dispersed stream of metal will be better capable of radiating its energy than a compact stream of metal. If emissivity of the metal particles and reflectivity of the surroundings are important, one would expect to note an effect of container shape and surface condition on reactivity. Such observations are reported in the references listed.

It is therefore quite understandable that a variety of results can be obtained from such experiments and it is also evident that little can be concluded as to the likelihood of metal-water reactions under experimental conditions where separation of the various important variables is not possible.

Accidents Involving Zirconium Fires or Explosions

A number of publications have appeared recently evaluating experiments and accidents in which zirconium fires or explosions were noted.⁽⁴⁹⁾ In these reports it is pointed out that accidents are more complete and more violent with moist rather than dry chips or powder. It is important to point out several important differences between these observations and the conditions under which zirconium-water reactions are likely to occur in case of the loss of coolant accident in PWR. Accidents which have been observed to occur in storage appear in all cases to be related to lumping of the powder or chips in a more or less compact mass presenting a high surface for reaction but relatively poor means for heat loss.

Practically all the accidents which have been observed are related to the storage of particles with surface-to-weight ratios greater than 10^{-2} to 10^{-3} square meters per gram. The minimum particle size estimated to result from a loss of coolant accident in PWR has been estimated above to have a surface-to-weight ratio of about 4×10^{-4} square meters per gram.

The accidents cited in these reports are related to the storage of dry or moist chips and not of material immersed in water. The conditions

for heat loss by conduction or convection through the water which are considered to be present at low temperatures in case of the PWR loss of coolant accident are not present in the case of storage of moist chips where the water phase is not continuous.

Note that an essential difference in environmental conditions exists in case of the experiments of Higgins,⁽⁴⁰⁾ those of Crooks,⁽¹⁷⁾ and those analyzed above as compared with the experimental conditions referred to earlier.⁽⁴⁹⁾ In the former cases each particle reacts independently of its neighboring particles, whereas in the latter case where the particles are in a compact mass the temperature of each affects the temperature of its neighboring particles. It is necessary therefore to first consider the reaction of single particles exposed to air or steam and then that of the masses of such particles. Note that a reactive metal exposed to air will attain an equilibrium temperature such that its rate of heat gain due to surface reaction will exactly balance its rate of heat loss due to convection and radiation. In case of bulky samples this temperature, of course, is infinitesimally above ambient temperature. On the other hand, in the case of finely divided metals this temperature can be well above room temperature and in certain cases can be high enough to show incandescence. This undoubtedly is the explanation for the sparking observed when grinding metals. Obviously, the lower the rate of oxidation of the metal, the lower its equilibrium temperature. Thus, if the rate of oxidation is decreased, for example, by formation of a thin adherent oxide film prior to exposure to air, the finely divided material can be safely handled. It is thus common practice in the powder metallurgy industry to first expose powders to slightly oxidizing environments after their production by hydrogen or vacuum reduction so that they can then be safely withdrawn into the atmosphere. If they are withdrawn into the atmosphere without such pre-oxidation, they will often spontaneously combust. This same explanation probably also holds for the observation that the metals, copper and lead do not spark on grinding, stainless steel sparks rather feebly, whereas iron and zirconium spark brightly.

In case of a mass of scrap consisting of fine chips, powder, or turnings stored either in open or closed containers, any spark, flame, or combustion which raises, even locally, the temperature of the metal will result in increased oxidation and hence increased temperatures. Since no means are available for removing the heat of reaction, the temperature of the mass of the metal will rise adiabatically similar to the rate of temperature rise shown in Fig. 3-24. Note that in case of combustion of a mass of chips or powder, the surface available for heat loss relative to the surface available for reaction increases directly with the total mass. Thus, under storage conditions, the temperature of ignition or reaction will, as estimated by Andersen and Belz,⁽⁵⁰⁾ decrease with increasing mass. If the particles are moist, the decrease in reaction rate by consumption of available oxygen will be greatly delayed because of the evaporation of steam and

the availability of oxygen from the steam. In addition, the combustion of hydrogen produced in the zirconium-steam reaction will aid in further dispersing the mass of chips and allowing access of the oxidizing environment. Note that if the chips are immersed in water and can lose heat from each particle by conduction and convection, any local heating will not propagate because a means of heat loss is available even at ambient temperatures. It may be concluded, therefore, that the incidence and consequences of zirconium fires occurring under dry or moist storage or conditions of inadequate protection against temperature rises are inapplicable to the results of zirconium reactions occurring in systems immersed in water.

In addition to the studies of Lustman⁽⁴³⁾ presented above, it is of interest to record some more recent information from reaction tests conducted by Higgins and Schultz.⁽²¹⁾ They report results of tests using an explosion dynamometer into which molten metals were sprayed, along with other data of importance to chemical energy released. The metals, zirconium, Zircaloy-2, aluminum, uranium, sodium-potassium alloy, and stainless steel, were tested. Explosive power, reaction time, reaction efficiency, and the effects of temperature and droplet size were investigated.

A combination of injection pressure and orifice size was selected to produce a wide range of droplet sizes (30 to 3000 microns). In most cases the distribution of particles followed a normal distribution curve with the mean particle diameter between 300 and 500 microns. The particles of zirconium and zirconium alloys, in most cases, were solid spheres; however, some hollow spheres were formed in most tests, usually in particles of larger size. Nickel formed relatively uniform spheres. On the other hand, the aluminum residue was irregular, spongy, and quite fragile. The stainless steel residue was composed of very irregular and porous particles having great specific surface areas. Irregular particles also made up the uranium residue.

Percentage of reaction versus droplet size curves for zirconium and zirconium alloys and for aluminum are given in Fig. 3-28. The greater reaction percentage of aluminum for a given particle size most likely results from the aluminum particles greater surface area for the same apparent particle size.

An examination of plots of initial metal temperature vs the peak pressure, total impulse, and work show that the zirconium curves extrapolate to the "no-reaction" level at the melting point. The reaction violence then increases with increasing temperature with an abrupt change in the rate of increase of reactivity at approximately 2400°C. In contrast, the reactivity of aluminum is nil to 1170°C, considerably above its melting point. At this initial metal temperature reactivity rapidly increases as temperature increases. (It should be noted that some chilling probably takes place initially on injection - Ed.)

FIG 3-28
PERCENTAGE REACTION VS. DROPLET DIAMETER
EXPLOSION DYNAMOMETER
FOR ZIRCONIUM, ZIRCONIUM ALLOYS AND ALUMINUM
(REF. 21)

The over-all efficiency of the energy-conversion process is defined as the ratio of the mechanical work done in the dynamometer to the theoretical chemical energy released by complete reaction of the injected metal. The principal losses are: (1) heat rejection of the thermodynamic cycle, (2) incomplete chemical reaction, (3) heat transferred to water and metal parts, and (4) mechanical losses such as gas escaping through the ejector port. The highest reliable over-all efficiency for zirconium and alloys was 2.32%. The losses due to incomplete chemical reaction in that test was probably near 70%. Thus the corrected over-all efficiency of converting liberated energy into work was 7.7%. Similar considerations gave for aluminum and sodium-potassium alloy, respectively, corrected over-all efficiencies of 1.5 and 1.2%.

The pressure pulses measured are stated to be primarily due to the liberation and expansion of the hot hydrogen gas released during the reaction. Plots of reaction pressure versus time are presented in Fig. 3-23. The pressure rise time indicates that metal-water reactions in these experiments are not high-order explosions. It takes milliseconds rather than microseconds to attain peak pressure and the rates of pressure rise are slightly greater than for black powder. At a temperature of 2070°C, aluminum is a more powerful reactant than zirconium.

The average rates of energy release for several tests were roughly approximated from the energy liberated in the reaction and time between first indication of pressure and when the gas appears to expand adiabatically:

System	Rate of Energy Release (Cal/sec)
Zr + H ₂ O	280 x 10 ³
Al + H ₂ O	1660 x 10 ³
U + H ₂ O	41 x 10 ³
NaK + H ₂ O	690 x 10 ³
303 SS + H ₂ O	46 x 10 ³
Black Powder	1050 x 10 ³

The "characteristic product" of a system is a rough approximation of the mechanical work performed by the explosion. It is obtained by multiplying the volume of the gaseous products of the explosion (at STP) by the heat of reaction assuming equipartition of the energy among the moles of the product. Several values are listed in the following table for comparison:

Certain Properties of Some Reacting Systems

System	Volume of Gases cc/gm(a)	Heat of Reaction cal/gm(a)	Partition Factor	Characteristic Product (a)
Nitroglycerin	712	1580	0.67	757,100
Black Powder	285	685	0.80	156,000
Li ——— LiOH	4615	6960	0.33	3,696,000
Na ——— NaOH	487	1900	0.33	305,250
Al ——— Al ₂ O ₃	1240	3550	0.75	3,300,000
Mg ——— MgO	922	3190	0.50	1,470,000
Zr ——— ZrO ₂	492	1330	0.67	438,180
U ——— U ₃ O ₈	250	437	0.89	97,010
SS(b) ——— Oxides(c)	584	118	0.76	52,440
NaK(d) ——— Hydroxides	375	1475	0.33	182,490

(a) For the metal-water reactions these values are based on the weight of metal only.

(b) Type 303

(c) Primarily Fe₃O₄ and Cr₂O₃.

(d) 56% K - 44% Na.

An analysis was performed on the residue from previous molten-metal drop tests⁽²⁰⁾ in which drops were dispersed in water by blasting caps. The energy yields for particles of equal size in the drop tests were approximately equal to those obtained by spraying the metal into the water. The energy added by the blasting cap is concluded to be insignificant.

A study was made of the oxidization rates of Zircaloy-2. The experiments were performed for all combinations of the following parameters:

Exposure time, sec	200, 600, and 1000
Steam pressure, psig	200, 600, and 1000
Sample temperature, °C	800, 1000, and 1200

Most of the anomalies appear at high pressure. But from visual examination, low pressures have little or no effect on the oxidation rate.

The data indicated that the test conditions covered a point where an abrupt increase in corrosion rate occurred. Such a point is called a "breakaway point." Photomicrographic evidence showed that the "breakaway" occurred without the spalling of the oxide film.

The actual location of the breakaway point is unknown but the post transition corrosion rate of Zircaloy-2 can be written:

$$\text{Rate, } B = 2.0 \exp(-32/RT), \text{ cm/sec}$$

where

R = gas constant = 1.99 cal/gm-mole °K, and

T = absolute temperature, °K.

The possible theory of metal ignition is a controversial and complex subject. The Eyring-Zwolinski theory of metal ignition⁽⁵¹⁾ was critically reviewed, reinterpreted, and improved. The discrepancy between this theory and the experimental results reported by Fassel, Gulbransen, Lewis, and Hamilton⁽⁵²⁾ was explained. Metal ignition was suggested as a result of microscopic, intergranular oxidation at the surface of the metal. This type of oxidation can produce localized hot spots at the metal surface thus vaporizing and igniting the metal.

At a sufficiently high temperature, there is evidence that the velocity of growth of the metal oxide film at a constant temperature is invariant with respect to time. This is the so-called linear rate law. The linear oxidation rate of the metal in atoms/sq cm/sec was written in terms of Eyrings formulation of absolute reaction rate. Then the velocity of

progression of the oxide-metal interface in cm/sec according to the linear rate law was given. In the next table are the results for systems studied in the report. Relatively good agreement has been found between rates based on theory and determined by experiment.

Maximum Oxidation-Rate Equations for Various Solids(21)

System	Average Temp. ° K	Max. Oxide Rate Equation ^(a) (cm/sec)
Ta + O ₂	873	$B = 5.2 \times 10^1 \exp (-29.5/RT)$
C + CO ₂	1273	$B = 7.3 \times 10^2 \exp (-59.0/RT)$
C + H ₂ O	1346	$B = 1.2 \times 10^2 \exp (-60.3/RT)$
ZnS + O ₂	1100	$B = 1.2 \times 10^2 \exp (-60.3/RT)$
U + O ₂	473	$B = 1.7 \exp (-16.6/RT)$
Mo + O ₂	633	$B = 2.2 \times 10^1 \exp (-23.3/RT)$
Nb + O ₂ ^(b)	700	$B = 1.3 \times 10^2 \exp (-43.6/RT)$
W + O ₂	900	$B = 8.0 \times 10^3 \exp (-47.4/RT)$
(Zircaloy-2 + H ₂ O)	633	$B = 2.0 \exp (-32.0/RT)$
(Zircaloy-B + H ₂ O)	1750	$B = 4.7 \exp (-39.2/RT)$
(Zircaloy-B + 2.2% Be) + H ₂ O	1750	$B = 7.5 \exp (-38.5/RT)$

(a)Note: R, as employed in table, is equal to
 $1.99 \times 10^{-3} \text{ k cal}/(\text{gm-mole})^\circ\text{K}$

(b)The experimental data for Nb and W are open to some question because of the possible influence of trace impurities.

3.4.1.2 Pressure Loads

If the primary systems are breached, possible internal pressure and temperature loads on the outer or last barrier must be determined in order to arrive at design figures. This once-in-a-lifetime loading, unfortunately, must be determined as a function of time in the general case. The aggregate pressure loading may have several different sources each varying differently with time making possible more than one maximum and the loading may be so rapid that a dynamic analysis of the strains is necessary. The parts of the loading considered are: a) expansion of core and coolant which creates partial pressures and transfers heat to the enclosed gas; b) release of heat from chemical reaction between the core materials and the environment which leads to creation of gaseous combustion products having significant partial pressures and causes transfer of heat to the enclosed atmosphere; c) heating of enclosed atmosphere and solid components by absorption of the beta and gamma energy of the released fission products and other radioactive materials.

Each of these partial pressures has a different time history as does the heat available. For example, the stored energy of a rapidly released coolant will lead to a pressure rise peaking in a few seconds and then decaying through transfer of heat to the contained air and structure. Some time after the loss of coolant, melting and vaporization due to after-glow heat may release fission products, spreading an additional energy source to points in the system where energy is absorbed by the air rather than the large heat sink of materials surrounding the core. This dispersal mechanism applies also to the core materials. In several investigations of specific systems, the total pressure-time history shows an early peak, followed by a decay lasting perhaps for several hours. The continuing heat source of the fission products gradually adds heat to the sinks and to the air in the container, resulting then in an increase in temperature and pressure which can be balanced only by heat dissipation to the region outside the container.

If the thermal impedance to heat dissipation is great enough, the pressure and temperature rise can continue to a peak higher than the initial one before a balance is achieved. In the case of a non-pressurized but chemically reactive coolant, the situation differs only in a shifting of the first peak to a later time as burning occurs. In such cases the fission-product heating may not result in separate peaking, but merely in augmenting the initial peak. In the case of a coolant fire, the pressure may be reduced below the outside pressure due to oxygen depletion. Complete depletion would result in removal of the partial pressure of oxygen, about 3 psi. This negative affect could be increased further by an increase in the barometric pressure outside the container ($1\frac{1}{2}$ psi seems a reasonable maximum). Rather than design the container for this large building load it would seem reasonable to allow air to bleed into the shell through a one-way valve.

The following information is cited to show what is known and considered in determining the pressure load:

(1) Expansion

(a) Thermodynamic treatment for expansion into a shell of a compressed liquid-vapor system has been given by J. C. Heap.⁽⁸⁸⁾

The liquid-vapor system may consist of either liquid or vapor alone as well as a mixture of both. General equations are presented for the solution of both the time dependent and independent problem.

The time independent treatment consists of going from one state point to another, assuming the initial state point is known and the final mixture is in a state of static equilibrium. Not considered are changes in potential energy, non-mechanical forms of energy changes, such as capillary, chemical, etc., and the omission of heat sources or sinks which may be present after the incident. The perfect gas laws are assumed valid and the specific heats assumed constant. Application of the first law of thermodynamics, Dalton's law of partial pressure, and various definitions yields a general solution of the problem under the preceding suppositions. Numerical examples and curves are presented relating pressure in the containment shell to containment volume required per pound of liquid vapor and the final quality and temperature of the water-vapor mixture in the containment shell. In particular, methods are given to solve for both the volume required to obtain a desired final pressure and the final pressure obtained by expanding into a known volume for the cases of where the final state is a mixture of gas, vapor, and liquid, and where the final state is a mixture of gas and vapor only. The results are displayed in Figs. 3-29 through 3-38.

General equations are derived for the time dependent problem based on an open system analysis with the working substance assumed to be homogeneous in composition, state, and velocity at the breach at any instance. The only energy forms considered are flow work, internal energy, kinetic energy, potential energy, and transferred heat. Chemical energy and nuclear energy terms are not included, although the treatment could easily be extended to do so. The complexity of the problem arises chiefly from the fact that flow is non-steady. Solutions can be obtained only by tedious numerical methods, implying application only to specific cases. Hence no parametric solutions are given.

(b) The matter of pressure loads is also pursued with respect to the Shippingport reactor.⁽⁵³⁾ The problem of time dependence is somewhat more difficult here because of the use of four interconnected containment vessels. The normal coolant pressures and temperatures are 200 psi and 535°F, an initially sub-cooled state.

FIG. 3-29
EXPANSION OF ONE POUND OF SATURATED
WATER INTO A CONTAINMENT SHELL OF DRY
AIR INITIALLY AT 14.7 psia AND 60°F
(REF. 88)

FIG. 3-30
EXPANSION OF ONE POUND OF LIQUID-VAPOR
(WATER) INITIALLY AT 500 psia INTO A
CONTAINMENT SHELL OF DRY AIR INITIALLY
AT 14.7 psia AND 60°F
(REF. 88)

FIG. 3-31

EXPANSION OF ONE POUND OF LIQUID-VAPOR (WATER) INITIALLY AT 1000 psia INTO A CONTAINMENT SHELL OF DRY AIR INITIALLY AT 14.7 psia AND 60°F (REF. 88)

FIG. 3-32

EXPANSION OF ONE POUND OF LIQUID-VAPOR (WATER) INITIALLY AT 2000 psia INTO A CONTAINMENT SHELL OF DRY AIR INITIALLY AT 14.7 psia AND 60°F (REF. 88)

FIG. 3-33

EXPANSION OF ONE POUND OF LIQUID-VAPOR
(WATER) INITIALLY AT 3000 psia INTO
A CONTAINMENT SHELL OF DRY AIR INITIALLY
AT 14.7 psia AND 60°F (REF. 88)

FIG. 3-34

EXPANSION OF ONE POUND OF WET MIXTURE
(QUALITY = 10%) INTO A CONTAINMENT SHELL
OF DRY AIR INITIALLY AT 14.7 psia
AND 60°F (REF. 88)

FIG. 3-35
EXPANSION OF ONE POUND OF WET MIXTURE
(QUALITY $\approx$ 20%) INTO A CONTAINMENT SHELL
OF DRY AIR INITIALLY AT 14.7 psia AND
60°F (REF. 88)

FIG. 3-36
EXPANSION OF ONE POUND OF WET MIXTURE
(QUALITY = 30%) INTO A CONTAINMENT
VESSEL OF DRY AIR INITIALLY AT 14.7
psia AND 60°F (REF. 88)

FIG. 3-37
EXPANSION OF ONE POUND OF WET MIXTURE
(QUALITY = 40%) INTO A CONTAINMENT
SHELL OF DRY AIR INITIALLY AT 14.7
psia AND 60°F (REF. 88)

FIG. 3-38
EXPANSION OF ONE POUND OF WET MIXTURE
(QUALITY = 50%) INTO A CONTAINMENT
SHELL OF DRY AIR INITIALLY AT 14.7
psia AND 60°F (REF. 88)

For a small break in the main coolant system, the pressurizer would attempt to maintain the prescribed pressure; hence, the water would be forced to flow out of the pressurizer until it is empty. Phase separation would occur in the flowing steam-water mixture when the water level dropped below the rupture. In the event of a large break, the pressurizer would not be capable of maintaining the flow of compressed water to the primary system, and the pressure of the system would instantly fall to saturation causing a two-phase flow out of the breach.

The internal energy inside the plant container would be increased with the entrance of the steam-water mixture; hence, increasing the intensity of the pressure and temperature. If the breach were of an extensive size, the mixing of the vapor and air could be violent, and the blowdown force would be sufficient to move the mixture from chamber to chamber fairly well, establishing uniformity throughout.

Condensation of the steam would occur on the cool surfaces inside the plant container during the discharge, increasing the surface temperatures and causing heat flow from the interior to the atmosphere and to the interior of the components.

Initially, pressures were estimated for a rapid expansion, neglecting heat transfer and assuming no contribution from reactor decay heat, primary system and pressure vessel heat. Using constant internal energy states, a trial and error solution was performed assuming the perfect gas laws were valid for the air and using steam tables for the water vapor. This led to an over-pressure of 52.8 psi. More realistic calculations described below showed this value to be conservative by less than 10%, so it was used as the design pressure.

An analysis of a blowdown for a rupture of a 15-inch inside diameter line of the coolant system was also studied in which more realistic assumptions were made. For making the computations, the continuity equation at critical flow, a transient partial differential force balance equation, a modified differential force balance equation, and transient heat balance equation were used. When the size of the rupture is known, the discharge rate may be calculated, yielding the change in internal energy of the plant container as a function of time. Assuming zero heat transfer with surrounding environments, an energy balance after a finite time would give the temperature and pressure of the plant container. The mixing process of the vapor and air being extremely complex, the rate at which the mixture reaches equilibrium cannot be computed with precision; therefore, assumption of instantaneous homogeneous-complete mixing and thermal equilibrium throughout the various chambers was employed as a reasonable basis.

A peak loading is reached in the plant container when the rate of pressure increase caused by boiling, decay, and blowdown is equal to the rate of decrease due to condensation.

The temperature gradient, heat transfer, etc., from the vapor to the condensing surfaces and then to interior materials are related through simultaneous partial differential equations. Complications are multiplied as the boundary conditions are functions of time. To avoid tedious solutions, the technique of taking a heat balance over an increment of finite time for the system is utilized where the initial ambient temperature and rates of heat transfer into and out are assumed constant during the time interval; hence, a new pressure and ambient temperature are attained. This variation gives rise to new temperatures and temperature gradients which in turn require new heat transfer rates. By this iteration technique the computations proceed from state to state.

The boiling and condensing surface coefficients play an important function in determination of the magnitude of the temperature and pressure peaks and were thoroughly examined.

The calculated relations for pressure and temperature due to expansion of coolant are all included in Fig. 3-39, along with chemical metal-water reaction contributions but without decay heat.

In attempting to decide about what assumptions to use regarding a system break, a scheme for determining maximum size for ductile rupture was worked out by Ketchum.⁽⁵⁴⁾ The method will not apply for creep, fatigue, brittle failure, and loads creating stresses over 70% yield point stress. The analysis depends on the growth of a minute crack to a long crack which reduces the strain energy needed for failure. Thus, at a particular crack length for a certain stress level, the system will rupture. The circumference of the rupture hole is then taken equal to twice the crack length. The actual crack length at failure cannot be accurately calculated but upper and lower bounds based on the ultimate and yield stress of the material can be found. All applicable failures thus far give crack lengths consistent with theory but the failures are too few in number to fully validate the theory.

The method uses the strain energy approach. Assuming failure due to bursting will occur somewhere between the strain energy due to the yield point stress and the ultimate stress gives the lower and upper limits of failure. The expression for reduced strain energy is derived from known expressions for the stress concentration around an elliptical hole. This leads to the limiting forms for crack length and hole diameter:

FIG. 3-39
PRESSURE IN PLANT CONTAINER AFTER COOLANT SYSTEM RUPTURE
(REF. 53)

$$\text{Maximum crack length, } L' = \sqrt{\frac{4(1+\mu)}{3\pi} \left(\frac{\sigma_u^2}{\sigma^2} - 1 \right)}$$

$$\text{Maximum hole diameter, } D_u = \frac{2L'}{\pi}$$

$$\text{Minimum crack length, } L = \sqrt{\frac{4(1+\mu)}{3\pi} \left(\frac{\sigma_{yp}^2}{\sigma^2} - 1 \right)}$$

$$\text{Minimum hole diameter, } D_{yp} = \frac{2L}{\pi}$$

where

σ = sum of squares of differences of principal stresses
 σ_{yp} = yield point stress
 σ_u = ultimate stress
 μ = Poisson's ratio

(c) One of the means of making more precise and realistic calculations of the expanding coolant pressure loads requires that heat transfer to sinks and to the structure be considered. This is so because whenever the expansion takes place in a period of more than several seconds, the pressures would be lower than the values calculated in going directly from the initial to the final states without heat loss.

To determine the possible reduction to be realized by considering available heat sinks, and to determine what heat transfer coefficients might be used, a series of tests were conducted by Sargent and Lundy and the Graver Tank Company.⁽⁵⁵⁾ This work is summarized here.

Transient thermal stresses would occur where a reactor incident releases a compressed liquid, liquid-vapor, or vapor, thereby heating the containment shell inner surfaces. The resulting stresses are functions of the vapor-air temperature change rate, the convection heat transfer coefficient, properties of the vessel materials, the geometric arrangement, etc.

To evaluate the convection coefficient and to determine the maximum pressure and temperature with several different internal arrangements, compressed water was released into a vessel simulating a containment shell. Evaluation of the convection coefficient can be obtained if the time rates of change of the following are known: vapor-air mixture temperature, vessel heat content, and inside surface temperature.

The containment shell was simulated by a vertical cylindrical shell with a flat plate for the bottom end and a hemi-elliptical shell for the upper end. Four different interior arrangements were utilized within the containment shell during the test runs.

The first arrangement had a horizontal dividing plate with a diametrically located rectangular slot which extended across the vessel and a closeable access manway. A water trough was suspended from the dividing plate. The slot and manway permitted passage of the water vapor from the lower section to the upper section. The water trough, when filled, provides a seal between the upper and lower vessel sections; hence, passage of the the vapor-air mixture from the lower to the upper section gives an intimate contact with water. The second interior arrangement had the steel division plate and water trough removed, thus allowing free passage of vapor throughout the tank. The third internal tank arrangement consisted of a concrete lining. The lining was used in establishing the effect of different internal surfaces and material on the pressure buildup and temperature of mixtures. The fourth internal arrangement consisted of a water trough suspended directly below the distribution pipe by welding plates to the trough and pipe. A thorough mixture of the flash steam into a water heat sink was insured by this setup.

The observed maximum pressure was considerably less than that resulting for zero heat transfer to the walls of the vessel and internal installations. The peak pressure and temperature normally occurred between 6 and 8 sec. after the compressed water release.

For the tests in which a steel surface was exposed to the vapor-air mixture, the maximum gauge pressure attained was approximately 55% of the theoretical pressure. When a concrete surface was exposed to the vapor-air mixture, the maximum absolute pressure obtained was approximately 80% of the theoretical value.

For the tests with the division plate and empty water trough in place, a maximum temperature difference of approximately 105°F was obtained between the top and bottom sections of the containment shell. The temperature difference after five minutes had abated to approximately 36°F. Slight pressure differences did exist between the two sections, but they were transitory and readily equalized.

Data comparison between tests with the trough empty and filled indicated that the maximum pressure and temperature in the upper portion were not appreciably effected.

Testing with the manway in the division plate open made no noticeable difference on the maximum temperature difference between the upper and lower sections or the change rate of the temperature difference with the respect to time for the upper and lower sections.

When the division plate was removed during the tests, only minor variations were observed between the temperature in the upper and lower sections.

For the test where the water trough was suspended from the distribution pipe, no rise in pressure was discernible although the ambient temperature in the vessel increased 19°F in 15 min after compressed water release.

The data indicates that the maximum transient convection heat transfer coefficient during the first 15 sec of the test was approximately 700 Btu/(hr) (sq ft) (°F). The average heat transfer rate over this period was sufficiently high as condensation of the water vapor occurred rapid enough to reduce the containment vessel pressure rise considerably below that of the theoretical maximum pressure with no heat transfer.

The pressure reductions realized were determined from the test results reported, and are summarized in the following table:

Test Set-Up Description	Peak Absolute Pressure (psi)		Time to Peak (sec)
	Observed	Theoretical Maximum Calculation	
Trough in position (full or empty) (steel)	27	49	3
Trough empty, manhole open (steel)	35	67	6
Trough removed (steel)	39	67	5
Trough removed (concrete)	53	67	7

A further difficulty encountered when trying to make the studies more realistic comes in determining flow rates through the break assumed. In many systems, two-phase flow is encountered. Experimentally backed information by Isben, May, and Cruz⁽⁵⁶⁾ is available for two-phase flow at low system pressures. The experimental work covers the pressure range from 4 to 43 psia and the quality range from saturated vapor to 1% (weight) vapor. Discharge pipes were scaled down to between 1/4-inch and 1-inch diameter plus some annuli. The data are correlated empirically, and compared with calculations based on a homogeneous flow model. Observed mass flow rates are higher than calculated by 0-10% at a quality of 1 and monotonically increase to about 60% at qualities of a few per cent. Other work is reviewed, and comparisons made with other models. The model having a water annulus and gas core yields overestimates when void fraction data is used to arrive at the functional cross section of the gas core. Agreement varies from a match at a quality of 1 to overestimates by 85% to 125% at qualities of a few per cent depending on void data selected. Comparisons with a momentum model show good agreement at high qualities, with the model giving increasingly larger overestimates with decreasing quality to about a 25% overestimate at 10% quality, the limit of reliability for the calculated values.

Although the observations have been at low pressures, these tests serve as a caution to users of analytical methods. Experiments are being started to extend this work to higher pressures (1600 psia) to check the analytical extensions.

(2) Chemical Reaction

In addition to the pressures from expansion, the contribution from chemical reactions between the environment and the core material (including coolant and products of previous reactions) after ejection from the primary system must be considered. These are typified by sodium-air or sodium-steam reactions, solid, liquid, and gaseous

uranium burning in air, and hydrogen-oxygen reactions. Some detailed consideration has been given to these typical reactions in general investigations, but mostly with regard to specific reactor projects where they would be particularly pertinent.

(a) The work reported by Gott and Flook⁽⁵⁷⁾ of England on investigations pertinent to the Dounreay Fast Reactor System provides some generalizations and bases for calculation of the pressure-time history for that installation.

Sodium and sodium-potassium are similar in that burning in air occurs in the vapor phase. The rate of vaporization from solid or liquid alkali metals is quite low compared to the usual fuels like gasoline because of the alkali metals' higher boiling point and fairly high latent heat of vaporization. This combined with the small calorific value, yields a low rate of combustion for the liquid or solid phase. These considerations would not apply if the entire sodium is available as a free vapor because it appears that complete combustion occurs as rapidly as the vapor forms.

The maximum internal pressure developed in this particular system is estimated first for a fire where the vapor is produced immediately after shutdown by the fission product heating of the reactor and second for leakages producing sprays and spillages to establish the extreme case.

To develop the first case it is assumed that: one, a leakage or failure of the coolant system causes overheating of the reactor to such an extent that vapor is formed in a direct ratio to the heat liberated within the reactor; two, all the vapor produced emanates from the reactor into the free volume and instantaneous combustion occurs with the air; and three, the reactor continues to function until boiling of the sodium has been completed.

The amount of sodium evaporated is proportional to the heat release in any time interval; hence, the amount of heat released by the burning sodium is proportional to the heat release from the reactor. The air temperature rise has been estimated from the known reactor fission product heat release vs time curve; with considerations given to the heat loss to the sphere and from the sphere to the atmosphere.

Only the air and steel of the upper hemisphere are counted as heat sinks, since the thermal mixing is considered nil throughout the other compartments. The pressure rise is assumed to be confined in the upper hemisphere. Heat losses to the floor and structural equipment are ignored. The heat lost from the air to the sphere is of

great importance and here heat transfer by natural convection has been assumed. High variations in temperature can exist over the sphere surface at the rates involved in a fire since the amount of heat conducted around the sphere is small.

Curves for temperature and pressure versus time in seconds after shutdown (Fig. 3-40) reveals a maximum air temperature and pressure rise of 320°C and 15 psi. The sphere temperature remains practically constant for the first 10 to 15 min while the air is heating up. The peak temperature of the sphere is not attained until 1 to 2 hours have lapsed because of its high heat capacity.

FIG.3-40
PRESSURE & TEMPERATURE HISTORIES FOR DOWNREAY SPHERE
(REF.57)

The studies indicate that to reach pressures and temperatures which would be calculated on the basis of complete combustion without heat loss would require an extremely rapid vaporization rate. Estimates of the burning rates made from properties of sodium indicate that a very extensive area of exposure would be required to insure an adequate yield rate. On this basis it seems evident that a few precautions would prevent a fire of the magnitude required.

The second study phase covers the question of sodium dispersing into the air. The total surface area exerts a great influence on the burning rate and is related inversely to the drop sizes. The maximum pressure of the cooling circuits is not adequate to produce good atomization for a small hole. A pipe tear produces better atomization, but not appreciably.

The number of holes and length of jet required reveal that in practice most of the fuel would gusher out unburned, hitting surfaces, and then collect in pools.

The study disclosed that sprays are not dangerous in promoting rapid pressure and temperature rises. Simple precautions restrict the extent of the rises. The use of heat insulation not only serves its intended purpose, but assists in localizing the spray.

It is pointed out that a sodium-concrete reaction can be very intense. It was not considered in this study, because the concrete has been completely clad with metal wherever exposure to sodium is possible, particularly surfaces capable of retaining sodium.

Two other sources of information on the sodium reactions have been reviewed. One of these is work by Humphreys⁽⁵⁸⁾ on explosive ejection of hot sodium into air in a container. The second is a study of NaK-water reactions for a simulated mechanical failure in the steam generator for the Enrico Fermi plant reported by Jones, et al.⁽⁵⁹⁾

In Humphrey's work⁽⁵⁸⁾ it is pointed out that of the methods of exposure of sodium to air which might exist abnormally in a reactor system, explosive ejection into the air is potentially more severe than either a spray or a liquid pool. The sodium-air reaction would occur primarily while the sodium was in flight. Thus, the heat of reaction would be directly transferred to the atmosphere with little energy initially being lost to structures or the building wall. The explosive ejection would also create a large discharge rate and fine particle dispersion causing the largest possible rate of reaction.

His investigations are limited to this most severe condition. The experiments determined the completeness of a sodium-air reaction and the resulting temperature and pressure inside an enclosure.

The experiments were run under ideal conditions with the most important parameter being the molar ratio of sodium to air. This ratio controlled the products formed. For a surplus of oxygen, Na_2O_2 was formed. For a surplus of sodium, Na_2O was formed.

In between, both Na_2O_2 and Na_2O were formed. In all cases the theoretical calculations for temperature and pressure were greater than the experimental results (Fig. 3-41). Note that the theoretical maximum calculations are for zero reaction time, for complete reaction, and for no heat conduction to the containment shell.

NOTES:

1. CURVES INDICATE MAXIMUM POSSIBLE PEAK PRESSURES AND TEMPERATURES BASED ON:
 - (a) 100% REACTION OF SODIUM OR OXYGEN, WHICHEVER IS LIMITING.
 - (b) INSTANTANEOUS REACTION (WHICH IMPLIES NO HEAT LOSS, OR, PERFECT DISTRIBUTION, INFINITE PARTICLE VELOCITY, AND INFINITESIMAL PARTICLE SIZE OF THE SODIUM).
 - (c) THE FOLLOWING AVERAGE SPECIFIC HEATS ARE ASSUMED:

$$C_p(\text{H}_2) = 4.82 + 3.3 \times 10^{-4}t - 4.7 \times 10^{-8}t^2 \text{ cal/mole}^\circ\text{C}$$

$$C_p(\text{Na}_2\text{O}_2) = 0.31 \text{ cal/g } ^\circ\text{C}$$

$$C_p(\text{Na}_2\text{O}) = 0.31 \text{ cal/g } ^\circ\text{C}$$

$$C_p(\text{Na}) = 0.31 \text{ cal/g } ^\circ\text{C}$$

(d) INITIAL SYSTEM CONDITIONS

$P(\text{ATMOSPHERIC}) = 14.5 \text{ psi}$
 $T(\text{ATMOSPHERIC}) = 20^\circ\text{C}$
 $T(\text{Na}) = 400^\circ\text{C}$
 $\text{O}_2(\text{ATMOSPHERIC}) = 20\%$

2. EXPERIMENTAL DATA ARE INDICATED AS FOLLOWS:

(A) = EXPERIMENTAL PEAK PRESSURE (MEASURED), RUN A.
 [A] = EXPERIMENTAL PEAK TEMPERATURE (CALCULATED FROM MEASURED PEAK PRESSURE), RUN A.

FIG. 3-41
 PEAK PRESSURE AND TEMPERATURE VS. Na/O_2 MOLAR RATIO
 (REF. 58)

The dispersion of sodium in air apparently resulted more from an initial sodium-air reaction than from an explosive ejection. This secondary dispersion would be less effective in dispersing the sodium throughout the entire building atmosphere in a large enclosure.

The effect of water vapor present at normal humidity was small, but not unimportant when considering the initial phases before secondary dispersion occurred. This is because of the very high reaction rates.

Humphreys also discusses application of his results to full scale containment vessels. The rate of pressure fall-off in a full-scale reactor containment building would be considerably lower than in the experimental vessel due to the $1/R$ dependence of the surface to volume ratio, and because the full-scale building surface heat capacity is comparatively smaller. The steel in the experimental reaction vessel has a relatively larger total heat capacity which contributed greatly to the rapid rate of system pressure reduction observed immediately after establishment of peak pressure. On this basis, the relative heat capacity of the experimental reaction vessel is about twelve times that of the EBR-II Reactor Building as an example. Accordingly, in actual applications, higher average wall temperature would be produced, with correspondingly reduced heat transfer rates and slower pressure fall-off.

Monson and Sluyter(60) made estimates for EBR-II based on Humphrey's work as follows: The ejection of some 14,000 to 23,000 lb of 750°F sodium with an ejection energy of approximately 180 lb to 310 lb of TNT would raise the building pressure to 75 or 80 psig. The building is designed for 24 psig with a safety factor of four for a normal temperature, and the steel shell retains its strength up to 650°F. Thus if the temperature of the metal did not rise too high, the building could successfully withstand pressures of 75 or 80 psig. The shell temperature is conservatively estimated by assuming none of the energy is absorbed by the concrete missile shield, all available energy is absorbed by that portion of the steel shell above the operating floor, and the available energy is 75% of the total energy. Assuming a total energy release equivalent to 29,000 lb of TNT, a final temperature of about 400°F is reached. Note that the gas temperature associated with pressure of 75 or 80 psig are of the order of 3900°F which can be regarded as a peak temperature. To obtain the design pressure of 24 psig would necessitate the ejection of 3000 lb of sodium and would create a peak gas temperature of 2100°F.

Because of the much larger mean diameter of an actual reactor building, the effective heat transfer rate to the wall during the pressure buildup period (effected principally by unreacted sodium and transport of reaction products) would be comparatively lower, and the peak pressure, therefore, somewhat higher. However, effective propagation by the secondary dispersion mechanism to the entire building atmosphere would be less likely to occur in the larger system.

One possibly beneficial effect of reaction product fallout in a reactor containment system should be noted. If a nuclear accident of sufficiently large magnitude to effect ejection of significant amounts of sodium were to occur, a reactor building atmosphere could be expected to include fission products and plutonium, as well as activated sodium. It is probable that upon conclusion of fallout, the atmosphere would consist essentially of nitrogen and volatile or gaseous fission products, the major portion of the nonvolatile fission product particles having been swept down by the falling reaction products. Thereafter, any atmosphere leakage from the containment shell would involve only the volatilized fractions of the radioactive material originally introduced.

The work on NaK-water reaction by APDA⁽⁵⁹⁾ also contributes significantly to the technology. The experiments involved a 1/2-inch diameter stream of leaking water at various pressures into a circulating NaK system initially at 50 psig. In addition, a more specific model of the boiler-NaK system was checked for effects from rupture of one tube of a bundle. The initial states covered parametrically in 52 tests are: gas surge volume on NaK loop, 0, 6 to 8, 92% of NaK volume; NaK loop pressure, 50 psig; NaK loop temperature, 200, 400, 600, 800°F; water loop pressure, 600, 900, 1200 psig; water loop initial pressure, 200, 400°F. The significant findings were:

- (1) Shock waves can be produced in the zero gas surge volume runs in about 20 milliseconds.
- (2) The 92% gas surge volume setup is the one which provides adequate time lags to allow time for the relief and isolating systems to function.
- (3) Uncontrollable variations in rate and mechanism of reactions were found.
- (4) The greater space provided by the 92% surge volume leads to a larger extent of reaction and hence a higher temperature, although the dynamic pressure over saving is reduced.
- (5) Except during shock periods when apparent pressures reached several thousand psi above initial water pressures, most of the pressure increases were 400 to 600 psi. Temperature increases were of the order of 1500°F.

Siler and Zick⁽⁶¹⁾ report that Detroit Edison Company's Engineering Department ran tests for APDA spraying sodium into an air-filled container. This series of experiments was meant to establish a realistic value of the pressure that might arise in a containment

vessel. The test giving highest pressures was obtained by a high nitrogen injection pressure with very fine nozzle dispersion forcing a stoichiometric amount of sodium at 850°F into the vessel at a rapid rate. The highest pressure attained in the 532-liter pressure vessel was 38 psig and occurred in about six seconds. With a burning rate of 10 lb of sodium per second per 1000 sq ft of surface and a particle size radius of 1 mm, the calculated maximum pressure was 40 psig, where radiation and convection losses were assumed. For systems having an excess of sodium, it should be noted that an increase in the containment vessel free volume does not result in lower maximum pressure for the sodium-air reaction.⁽⁶²⁾

The addition of heat and increase in pressures due to hydrogen-oxygen reactions must also be considered. Hydrogen evolves for example from a metal-water reaction. It could in the worst case accumulate in the container in a localized region, then detonate or deflagrate. Information from several sources is presented here for reference, and additional sources are noted. Shapiro and Moffette⁽⁶³⁾ treat hydrogen flammability and its relation to contain loading.

A mixture of flammable gases such as hydrogen and air may be diluted with one of its constituents or with a chemically inert gas such as water vapor until it is no longer flammable. The point at which a mixture will become flammable or non-flammable by a slight change in composition is known as the "limit of flammability." A "limit of flammability" is obtained both for simple burning and detonation or explosion with the "limits" dependent on the pressure and temperature of the mixture. Normal variations in pressure do not affect the limits of flammability appreciably with the range being slightly narrowed for pressures below 10 atm. Temperature, on the other hand, widens the range of flammability appreciably in a linear manner. The burning also depends on the propagation of a flame front. Convection currents due to the hot burnt gas rising, lower the flammability limits for upward propagation and raise the limits for downward propagation.

The addition of chemically inert gases lowers the range of flammability and can eventually eliminate any reaction from taking place. For 1 volume of any hydrogen-air mixture only 7.6 volumes of water vapor are required to make the mixture completely non-flammable. Fig. 3-42 summarizes known data on the flammability of hydrogen-air-steam mixtures. The spontaneous ignition temperature is around 520°C and varies with the amount of constituents present, the pressure, and the velocity of the mixture.

The probable effect of hydrogen flammability on the pressure buildup inside the containment vessel for PWR is shown to be non-critical. The hydrogen would be generated at a slow rate and would probably burn spontaneously, raising the pressure a few psi at a time when

FIG. 3-42
FLAMMABILITY LIMITS OF HYDROGEN-AIR STEAM MIXTURES
(REF. C3)

press the hydrogen-oxygen reaction, and that the addition of an inert gas would lead to the same results.

3.4.1.3. Concentrated Loads

The principal work lies in determining pressure-time-space relations so that the possibility of missile or spall formation can be determined and estimates made of the energy attenuation required to inhibit these possibilities. The work in part requires a detailed knowledge of the equations of state for the materials of the reactor and the primary structure surrounding it. For example, if the energy release takes place in a core submerged in a tank of water, it is necessary to determine the energy attenuation in the water between the core and the next media, e.g., pressure vessel, then shield. Information on equations of state and energy attenuation is meager, especially in the very high pressure regions.

(a) Some of the experiences of Armour Research Foundation obtained in analyzing explosive energy releases for nine reactors has been treated by F. Porzel.⁽⁶⁴⁾ He relates the history of a typical explosion in a reactor and the behavior of the surrounding primary structure. Much of this treatment is reproduced here.

The hypothetical system considered is a thermal core with a water reflector, placed in a reactor pressure vessel. This vessel is in turn set in concrete and has a removable cap held down by structure protected by a blast shield. An air-filled vault is located in the concrete below the pressure vessel. The entire system is enclosed in a containment shell.

the total pressure is much less than the containment vessel design pressure. If the hydrogen did not burn as it was generated, a mixture of steam, air, and hydrogen which will not detonate would be obtained. Detonation could occur in localized pockets of hydrogen, but the peak pressure should not exceed the containment vessel design pressure.

The significant contention here is that if the system is of the type to have a metal-water reaction, it is also probable that sufficient water vapor will be found in the container air to suppress the hydrogen-oxygen reaction, and that the addition of an inert gas

The energy release is supposed to occur in the core, at rates characteristic of high explosives and having an energy equivalence of a few hundred to a thousand kilograms of TNT. The resulting shock wave, roughly spherical, proceeds outward from the core through the pressurized water. Upon meeting the steel walls of the pressure vessel, the water pressures are increased several fold by a reflection process. Generally, the shock pressures at this point are so high (many thousands of atmospheres) that there is little hope of containing the explosion within the pressure vessel by any practical design.

When the shock in the water reaches the steam-water interface above the core, a several hundred-fold reduction in pressure occurs upon entering the rarified medium. This greatly reduced shock proceeds upward through the steam and is then increased several fold in pressure by reflection at the top of the vessel. This reflected pressure, is still greatly below the original pressure in the water, and constitutes the pressure loading on the cap. Fortunately, the loading is of short duration. It is the large reduction in pressure as the shock wave moves from the liquid to the gas phase which eventually saves the top plate from complete destruction.

A number of mechanisms occur for relief of pressure on the interior, but the most important relief which can be provided is a designed absence of strength at the bottom of the pressure vessel. When the downward shock breaks out from concrete to an air cell beneath the reactor, a severe reduction in pressure occurs. A corresponding rarefaction passes upward through the concrete, then through the water, at essentially the velocity of sound in each medium.

It is this rarefaction from the bottom of the vessel, moving upward with the velocity of sound, which literally tells the energy within the vessel that there is an effective escape route at the bottom of the vessel. By the time the explosion has expanded into the relatively immense volumes of air around and above the reactor, through the various avenues provided for it, the shock over-pressures are reduced to a few psi. This is sufficiently low that the energy of the explosion can be contained by an outer containment shell of reasonable thickness and structural strength. The rarefaction from the bottom thus provides the eventual relief of pressure within the vessel and in turn serves the important role of limiting the time duration of the pressure-loading on the cap structure.

The function of the hold-down assembly is to hold down the cap until the rarefaction from the bottom relieves the pressure on the cap. The pressure vessel itself cannot hold down the cap, because the steel below the water-line, will behave like a fluid at the shock pressures involved. In fact, the initial shock wave which passes outward through the concrete is usually strong enough to render the hold-down assembly of dubious value as a tension member for which it was designed. The tension members might fail under bending stresses at the pressures involved.

A "blast shield" is a device intended for use as an attenuator of shock energy. A wide variety of applications are possible when one is incorporated into the shield. The principal purpose of the "blast shield" is to reduce those pressures which cause failure of the hold-down structure by bending until the loading on the cap has been relieved by the rarefaction from the bottom.

Nearly every reactor design is of different geometry, and various components turn out to be critical components. Nonetheless, the various problems arise as a result of the indicated characteristic behavior when the energy release simulates a sharp explosion: i.e., initially intense pressure waves, pressure enhancement by reflection at dense surfaces, rarefaction and acceleration of material toward rarified media, significant differences in timing of the explosion processes, protection of critical members by blast shields, and eventual relief of pressure by dumping into large reservoirs of air.

A more detailed description of some of the hydrodynamic processes involved follows here.

The shock properties of interest for containment can be examined by considering the Rankine-Hugoniot equations. These are well-known relationships among pressure, density, shock velocity, material velocity, and other hydrodynamic variables across a shock front. The velocity equations are well known; the density relations have been derived by Porzel.⁽⁶⁵⁾

$$U^2 = \frac{(P - P_0)}{V_0 - V} V_0^2, \quad u^2 = (P - P_0) (V_0 - V)$$

Gases:

$$\frac{V_0}{V} = \frac{\rho}{\rho_0} = \frac{(2\epsilon + 1) P + P_0}{P + (2\epsilon_0 + 1) P_0} \quad (\text{reduces to standard form for}$$

$$\epsilon = \frac{\gamma}{\gamma - 1} = \text{constant})$$

Solids and Liquids:

$$P - P_0 = \frac{3}{n} \frac{\rho_0 C_0^2 \left[\left(\frac{V_0}{V} \right)^n - 1 \right]}{4 - \frac{V_0}{V}}$$

P = absolute pressure

V = specific volume

ρ = density

U = shock velocity

C = sound velocity

u = material velocity

γ = ratio of specific heats

n varies with material and process

subscript $_0$ denotes conditions in undisturbed media ahead of shock

Two comments about the Rankine-Hugoniot equations are pertinent to a discussion of reactor containment. If the initial energy release at the core is not instantaneous, the energy is first propagated outward through the pressure vessel as a rising pressure wave, rather than a sharp shock. Such a gradual rise in pressure could result in less structural damage than a shock of the same peak pressure. The first comment is that any rising strong pressure pulse, such as that shown on the left in Fig. 3-43, will quickly "shock up" during its passage through nearly any known material, rapidly reaching the configuration shown on the right. This steepening of the front depends on two simple, homely, and nearly universal properties of matter if it is squeezed by applying pressure:

FIG. 3-43
SHOCKING UP OF A PRESSURE PULSE
(REF. 64)

the volume decreases and the resistance to compression increases as the pressure is increased. Hence, the existence of a shock front becomes a property of the material, and not of the duration of energy release. The second comment is that the validity of the Rankine-Hugoniot equations do not rigorously require a sharp shock front, because they

are obtained by application of conservation principles across a region rather than across a mathematical discontinuity. In other words, the pressure wave which reaches the cap, bottom, and hold-down structure will most likely be a shock front, and if not, the Rankine-Hugoniot equations for a shock are quite likely still valid.

Some relationships for the peak energy density behind a shock front are shown in the following table.

Energy density, cal/cm³

Pressure	10 Bars	100 Bars	100 Bars	10,000 Bars
Air	0.72	15.5	240	1910
Water	1.4×10^{-4}	1.4×10^{-2}	1.4	53
Concrete	5.5×10^{-6}	5.5×10^{-4}	5.5×10^{-2}	5.5
Steel	1.2×10^{-6}	1.2×10^{-4}	1.2×10^{-2}	1.2

This tabulation provides an important comparison for the energy per unit volume in cal/cm^3 for several different materials at several different pressure levels. The outstanding result shown by this table is the very small amount of energy required in solid and liquid media to produce very strong pressure pulses. The pressure of one hundred atmospheres is especially interesting because it roughly represents the pressure in the shock front of a fireball from an atomic bomb in air just before the time of breakaway, which is obviously sufficient to raise the air temperature to several thousands of degrees Kelvin. The same pressure in water, concrete or steel is produced with very much smaller amounts of energy. This energy corresponds to a temperature rise of a tiny fraction of a degree centigrade, the order of numbers shown in the table.

The converse arguments show that enormously greater hydrodynamic energy can be stored within a compressible medium in comparison with an incompressible medium at the same pressure level. A factor of 100,000 times exists between steel and air at the 100-bar pressure level.

Figure 3-44 illustrates the entropy or waste heat concept, the basic idea behind the blast shield. The figure is a conventional thermodynamic pressure-volume diagram for unit mass of material. Energy is portrayed as an area on such a diagram. Starting from

FIG. 3-44
WASTE HEAT CONCEPT (REF 64)

ambient pressure P_0 , and ambient specific volume V_0 , the full curved line is the Rankine-Hugoniot curve, for the relation between pressure and density across the shock front. The peak total energy is the area under the rectangle below the pressure labelled P_s and between V and V_0 . The kinetic energy is the triangle in the upper right of the total energy rectangle. The internal energy is obviously the trapezoid occupying the rest of the rectangle.

If the shock pressure is below that required to achieve irreversible vaporization of the material through which the shock passes, the expansion can be described in a simple manner. A given particle will expand back down along the Rankine-Hugoniot curve to ambient pressure by adiabatic processes and reach a final specific volume substantially the same as its original specific volume, V_0 . Since the area under the curve represents the energy subsequently given up through expansion of the material, it has been labelled "hydrodynamic energy." The energy represented by the lens-shaped area in the trapezoid will not become available through subsequent expansion and therefore is labelled "waste heat." This latter energy appears as a temperature rise in the final product.

Some comments about waste heat are as follows. The waste heat concept is primarily a method of bookkeeping the energy and does not depend significantly upon the details of the expansion process. Next, it is fairly easy to find or to fabricate materials which "crush down" to small volumes at relatively low pressures. These are the ideal absorbers. In such cases, the Rankine-Hugoniot curve is sharply bowed, as shown in the Fig. 3-44 at the upper right. Nearly all the interval energy appears as waste heat and little if any appears as useful hydrodynamic energy. The kinetic energy fraction is still transferred to the next particle, through the principle of conservation of momentum, to sustain the shock. Finally, the subsequent expansion may be more complicated than shown here for a variety of reasons; heat transfer between phases, slippage along crystal boundaries, phase changes, etc. The net result is to terminate the expansion curve closer to V than to V_0 . These are equivalent to a hysteresis effect and lend further validity to the approximation that nearly all the internal energy is wasted behind the shock in an ideal absorber.

The waste heat principle holds for all solid and liquids, when the shock pressure is high enough. For example, in the typical reactor described earlier, 80% of the energy of the explosion may be dissipated in water by the time the shock front reaches the walls of the pressure vessel. At low pressures, or in the elastic range, the Rankine-Hugoniot curve and the waste heat area approach the configuration of the "ideal transmitter." This is the domain of ordinary elastic or acoustic theory.

In regard to energy partition in a blast wave consider that the hydrodynamic energy must clearly start at the explosive source with the fraction 1.0 of the energy release. Near the end of the vaporization (or bubble) stage, the fraction will fall to approximately 0.5 for most materials because of dissipation; the remainder appears as superheated vapor to drive the shock. Between the pressures just sufficient to cause vaporization and the strength of the material, the useful fraction of energy will continue to fall very rapidly, down to a few or a fraction of a per cent of the original energy at radii and pressures of practical interest.

The concept of energy partition as a constant fraction (so much into blast, so much into a bubble) is therefore essentially a misnomer. The hydrodynamic energy is initially 100%; the thermal energy is finally 100%; the partition depends on when, where, and how the energy is measured. If an attempt is made to estimate the original energy of explosion by any technique which measures the pressure or kinetic energy alone, at radii of practical interest, the results are bound to show very small values if these are interpreted as total energy and would lead to gross underestimates of the TNT equivalent. The hydrodynamic energy would be similarly small if it were measured in the shock wave from a high explosive charge. The familiar $W^{1/3}$ scaling law for blasts circumvents this difficulty, provided all details of the early history also scale in space and time.

On the basis of the foregoing considerations, the peak pressure versus distance curve may be described qualitatively as shown in Fig. 3-45. Beyond the close-in effects, there follows the range of pressures in which the shock wave is sufficient to vaporize the material behind the shock to a gas. During this phase, the peak pressure in a spherical wave decays with radius R roughly as $1/R^3$, but only roughly so.

FIG. 3-45
PRESSURE-DISTANCE AT SHOCK (REF.64)

Below the pressure sufficient to cause irreversible vaporization, the absorption of energy in solids or liquids through the waste heat process is very rapid. For good absorbers, the pressure decay is as high as $1/R^7$ or $1/R^8$. This rapid decay in pressure continues until it falls to the dynamic crushing strength of the material. At this pressure the waste heat mechanism becomes small, the wave becomes elastic, and pressure decays roughly as $1/R$.

The graph shows the advantage of capitalizing on the properties of the material between its vaporization pressure and its crushing strength.

In general, the hydrodynamic interest in the problem usually ceases once the dynamic crushing strength of the material is reached; this usually implies the statement that the shock is then too weak to do further structural damage. However, an important type of damage occurs from spalling of a rear surface, in which slabs of materials may be projected as missiles. The limit of spalling is set by the tensile strength of the material, which is considerably less than its crushing strength.

It is of interest to consider briefly the gradients in pressure behind the shock in various media without recourse to the complex algebra required to describe them. In general, the more incompressible a material is, the higher its peak pressure, the steeper the gradients behind the shock and the shorter the duration of the wave. For example, in the limit of weak shocks, the pressure gradient behaves with radius, R , like the quantity $\rho_0 C_0^2 / R$ (previous notation) and is steeper in solids and liquids than in air by the same ratios as appeared in the table for energy density. Thus the high pressures appearing in solid or liquid media need not be frightening. The decay and duration behind these shocks is such that the net static pressure impulse or dynamic pressure impulse is much closer or even below what they would have been in an air explosion at the same distance.

The rate of work, W , done by a shock front per unit area and time is the product $W = Pu$, analogous to a piston moving with force P at the material velocity u . If the peak pressure at the shock front

at a given time is the same in two media, then the rate of work in a gaseous phase relative to the rate of work in a dense phase is controlled by the material velocity, u , at the pressure level in question ($u \sim \sqrt{V_0 - V}$). Under this condition, the material velocity may be regarded as the geometric mean of the specific volume and fractional change in volume in each medium, or

$$u \sim \sqrt{V_0 \frac{\Delta V}{V_0}}.$$

The specific volume V_0 alone introduces a factor of 30 to 100 in favor of the gaseous phase. Moreover, the fractional change in volume V/V_0 for a gas is relatively large, being in the range from $3/4$ to $11/12$ for nearly all gases. But the fractional change in specific volume for a solid or liquid is very small, and this ratio again introduces many factors of ten to the relative rate of work in favor of the gaseous phase.

Such considerations show the enormous tendency for the shock energy to spill out into the gaseous phase - a factor of thousands at most pressure levels - with a corresponding depletion of the shock energy in the solid or liquid phases.

(b) Of particular concern is the lack of information on the state equation for a water-water vapor medium. The work on a water-air mixture is not applicable to the water-water vapor mixture, but is of interest for a liquid-gas system. The velocity of sound measurements provide a convenient determination of the pressure density relationship of the state equation. Information on water-gas mixtures was obtained by Karplus.⁽⁶⁶⁾

The velocity of sound, c , in a liquid containing gas bubbles is dependent on the densities ρ_G, ρ_L , the reciprocals of the bulk moduli of elasticity K_G, K_L , of the gas and liquid, respectively; and the volume concentration, x , of the gas as follows:

$$c^{-2} = [\chi \rho_G + (1 - \chi) \rho_L] [\chi K_G + (1 - \chi) K_L]$$

The development of the equation assumed: first, that the lowest natural frequency of the bubbles is well above the frequency considered; and second, that the distance between bubbles and the bubble diameters are much less than the wave length considered.

The experiment to determine the velocity of sound involves measurement of the wave length of sound, λ , at a known frequency, f . The phase velocity, c , is defined as

$$c = f \lambda.$$

The results of such measurements are given in Figs. 3-46 and 3-47.

FIG. 3-46

FIG. 3-47

VELOCITY OF SOUND IN H₂O-AIR MIXTURE (REF.66)

Consider now how the problem differs when the gas is the vapor of the liquid. The isothermal vibration of the gas bubble excludes the application of the results to a system consisting of steam and water. Since the water and the vapor are saturated, the pressure and temperature of the system are not independent. Thus an increase in pressure will cause some of the steam to condense and a decrease in pressure will cause some of the water to evaporate. This inapplicability led to further studies by Karplus⁽⁶⁷⁾ to determine basic pressure-energy relationships on a water-water vapor mixture. As a first step, an analytical study was made of sound velocity. The sound velocity, c , was found in terms of specific volume of fluid, V_f ; specific entropy of fluid, S_f ; the changes of these quantities upon evaporation, V_e and S_e ; and the quality, q , which led to the following equation.

$$c = \frac{qV_e + V_f}{\left[q \left(-\frac{\partial V_e}{\partial p} + \frac{V_e}{S_e} \frac{\partial S_e}{\partial p} \right) + \frac{V_e}{S_e} \frac{S_f}{p} - \frac{\partial V_f}{\partial p} \right]^{1/2}}$$

A plot of the velocity of sound versus pressure (Fig. 3-48) shows that the sound velocity is a linear function of the logarithm of pressure at very low steam concentrations. In Fig. 3-49 velocity has been plotted as a function of quality or phase concentration at selected values of pressure (or corresponding values of temperature). A monotonic increase in velocity with concentration at selected values of pressure (or corresponding values of temperature). A monotonic increase in velocity with concentration for all pressure is revealed. The rate of increase of velocity with quality, q , reaches a maximum at pressures in the region of 100 psi. As this critical point is approached, energy required for a phase change vanishes so that concentration becomes meaningless and the sound velocity quality curve will approach a horizontal line. In Fig. 3-50, the same curves are plotted on a log scale. It is found that in water containing just a few steam bubbles a very low sound velocity is present compared to the velocity in pure water.

Figure 3-51 is a representation of results extended to the condition of pure dry steam. It is observed that dry steam has a higher velocity than wet steam at the same temperature and pressure even when the water content of the wet steam approaches very near to zero. Velocity plotted as a function of pressure for constant entropy is shown in Fig. 3-51. Lines of constant quality and lines of constant temperature are also shown. At low pressures and high temperatures, velocity is reasonably independent of temperature and at higher pressures and lower temperatures these lines tend to curve down. When velocity is plotted as a function of constant enthalpy and pressure as in Fig. 3-52, the velocity is almost independent of pressure, provided condensation does not take place. Discontinuous velocity changes take place when the slightest trace of condensation occurs.

Karplus also developed an equation of state for the steam-water mixture giving velocity as a function of pressure and entropy. The specific volume, V , of a water-steam mixture in terms of specific entropy, S , and pressure, p , has the following form:

$$V = A_0 (S + 0.24)p^{-1} + B_0 (S - 0.76)p^{-0.7}$$

The accuracy of parameters A_0 and B_0 given by Karplus is better than 4% between 15 and 2500 lb in.⁻². Now, since

$$c = V \left(- \frac{\partial V}{\partial p} \right)_S^{-1/2}$$

the velocity of sound is readily obtained, because the work done going from condition V_1 to V_2 at constant entropy is

FIG. 3-48

VELOCITY OF SOUND IN H_2O -STEAM MIXTURES (REF. 67)

FIG. 3-49

FIG. 3-50
VELOCITY OF SOUND AS A FUNCTION OF STEAM QUALITY
(REF. 67)

FIG. 3-51
VELOCITY OF SOUND AT CONSTANT ENTROPY (REF. 67)

FIG. 3-52
VELOCITY OF SOUND AT CONSTANT ENTHALPY (REF. C7)

$$W = \int_{V_1}^{V_2} p \, dV = \int_{P_1}^{P_2} \left(\frac{\partial V}{\partial p} \right)_s dp$$

Finally, he developed the Rankine-Hugoniot curves for a steam-water mixture. The change of specific volume, V , and specific internal energy, E , across a shock front are given in terms of the pressure P by the well-known Rankine-Hugoniot relationships:

$$E - E_0 = (1/2) (P_0 + P) (V_0 - V)$$

Four additional relationships are obtained for a water-steam mixture. A final equation is obtained by eliminating E and q :

$$V = \frac{(P_0 + P) V_0 + 2G(E_0 - E_f + E_e V_f/V_e)}{(P_0 + P) + 2G(E_e/V_e)}$$

Where G is unity for a consistent system of units. V is then plotted against P for two cases of P_0 and V_0 in Fig. 3-53.

FIG. 3-53
RANKINE-HUGONIOT CURVES AND ADIABATS IN H_2O -STEAM MIXTURE
(REF. 67)

Some data are available on the behavior of pressure waves in steel. Katz and Peterson⁽⁶⁸⁾ have reported on results of shock collision on low carbon steel (Study of Shock Propagation in Ferrous Metals, Poulter Laboratories Technical Report 010-55, Stanford Research Institute, 12-30-55). This information is of principal interest near a high order energy release. A critical pressure was observed at 131 kilobars, where twinning occurs. There is a known phase transition from body-centered to face-centered cubic structure at this pressure. A value for shock attenuation below this pressure of about 0.8% mm was obtained. Empirical relationships between explosive thickness, contact zone widths, and collision zone widths were found. Thus with further work a correlation between stress and edge zone widths for normal shock waves impinging on steel might be obtained. The experiments determined a shock velocity in steel, independent of carbon content (0.01-0.4%) and heat treatment, of 4.93 ± 0.1 mm/ μ sec.

Application of explosive loading to the problem of determining the equation of state data for metals is given by Mallory.⁽⁶⁹⁾ The general method outlined determines the coefficients in the equations of state and provides a powerful general experiment tool. Aluminum is evaluated in the paper.

The pressure wave behavior in sodium has been approximated by most investigators to that in water, for which a considerable body of experimental information is available.^(70,71)

Cole⁽⁷⁰⁾ gives the relation

$$P_m = 2.16 \times 10^4 (W^{1/3} / R)^{1.13}$$

for the peak pressure in psi in a large body of water at a distance R feet from an equivalent charge of TNT W lbs. Sluyter⁽⁶⁰⁾ has determined that the body of sodium surrounding the core of EBR-II will have absorbed all but 10% of the energy in a shock wave resulting from a 300 lb TNT explosion at the core location, before the wave reaches the sodium boundary.

(c) One of the more serious difficulties lies in determining what non-nuclear forms of energy release are available to simulate the reactor energy releases. Proof of applicability of methods and data generated extensively with respect to high explosives would justify some of the analyses and permit H.E. to be used in correlating tests. One attempt at resolving this difficulty was reported by Larson and Olson.⁽⁷²⁾

To imitate the energy release rates of reactor malfunctions between an upper and lower bound, stainless steel cylinders were tested by the following three methods: first, detonation of explosive; second, ignition of a confined propellant; and third, sudden release of a gas under high pressure. The resulting air blast effects were measured at various distances from the source. Calibration was obtained by use of bare charges. The energies of the pressure waves were estimated, equivalent bare charges were calculated, and the resultant pressure, impulse, time distance relationships were checked against available experimental bare charge data and some theoretical point source calculations.

The division of energy employed to determine the energy in the pressure wave is of some interest.

It is presumed that first, part of the energy is absorbed by the vessel during the elastic and plastic deformation until rupture occurs; second, part of the energy is dissipated in accelerating the vessel fragment immediately after rupture; and third, the remaining energy forms the pressure wave in the air beyond the source.

The amount of plastic strain-energy absorbed by the vessel during deformation is given approximately by:

$$E_d = 2\pi R_1 \sigma_y t L \epsilon$$

where

E_d = plastic strain-energy absorbed by vessel, in. - lb.
 L = an equivalent length of a right circular cylinder with a total weight, wall thickness, diameter and material equal to the actual test vessel, in.

- R_1 = initial radius of cylindrical portion.
 ϵ = tangential rupture strain, in./in.
 σ_y = yield stress of material, psi.
 t = wall thickness of cylinder, in.

The velocity of a fragment of a vessel is:

$$V_0 = V_1 e^{C_D \frac{A}{M} \rho (x_1 - x_0)}$$

where

- V_0 = velocity of fragment at vessel surface, ft/sec.
 V_1 = velocity at known distance from vessel, ft/sec.
 C_D = coefficient of drag.
 A = average presented area, ft².
 M = mass, lb sec²/ft.
 ρ = air density, slugs/ft³.
 x_0 = initial known distance from vessel, ft.
 x_1 = known distance from vessel, ft.

The energy of fragment acceleration is

$$E_f = \frac{w}{g} V_0^2 \frac{1}{2}$$

where

- E_f = energy of fragment acceleration, ft-lb.
 w = weight of vessel, lb.
 g = acceleration of gravity, ft/sec².
 V_0 = velocity of fragment at vessel surface, ft/sec.

The change in internal energy of the nitrogen before and after the release is:

$$\Delta U = 778 w_1 C_v (t_1 - t_2)$$

where

- ΔU = energy of expansion, ft-lb.
 w_1 = weight of nitrogen in vessel, lb.
 C_v = specific heat at constant volume, Btu/(lb)-(F)
 t_1 = temperature in vessel, °R.
 t_2 = resultant temperature, °R.

The weight of nitrogen in the vessel is obtained from the ideal equation of state:

$$w_1 = 28.02 \frac{P_1 V}{RT_1}$$

where

- w_1 = weight of nitrogen in vessel, lb.
- P_1 = pressure in the vessel, psi.
- V = volume of the vessel, ft³.
- R = gas constant, 10.71 ft³-lb/in.²/°R/mole.
- T_1 = temperature in vessel, °R.

The resultant temperature is obtained by assuming adiabatic expansion; hence,

$$T_2 = T_1 \left(\frac{P_2}{P_1} \right)^{\left(\frac{\gamma-1}{\gamma} \right)}$$

where

- T_2 = resultant temperature, °R.
- T_1 = temperature in vessel, °R.
- P_1 = pressure in the vessel, psi.
- P_2 = final pressure, 14.7 psi.
- γ = ratio of specific heats for nitrogen.

Following this preliminary analysis, the peak overpressure in the blast wave was calculated from the Rankine-Hugoniot equation

$$\frac{P}{P_0} = \frac{2\gamma}{\gamma+1} \left[\frac{V^2}{C^2} - 1 \right]$$

where

- P = side-on excess pressure, psi.
- P_0 = ambient atmospheric pressure, psi.
- C = sound velocity in air ahead of shock, ft/sec.
- V = average velocity of shock wave, ft/sec.
- γ = ratio of specific heats.

The average velocity for an interval was obtained by measuring the transient time of the shock wave over a fixed distance interval between a pair of velocity gages. Velocity of sound was computed from

$$C = 1088 \sqrt{1 + \frac{T}{273}}$$

where

C = sound velocity in air ahead of shock, ft/sec.
T = temperature of the ambient air, °C.

The sensitivities of gages placed at the interval's midpoint can be calculated from

$$KA = \frac{H}{S} \times \frac{q}{P}$$

where

KA = gauge sensitivity, micromicrocoulombs/psi.
H = height of initial peak of pressure - time history on the film.
S = average voltage calibration step size on the record.
q = calibration charge, micromicrocoulombs.
P = peak overpressure, psi.

Peak pressure and positive impulse were computed from the side-on pressure-time histories of the shock wave. The peak overpressure for the test firings was computed by rearrangement of the aforementioned equation, i.e.

$$P = \frac{H}{S} \times \frac{q}{KA}$$

Positive impulse is defined as

$$I = \int_0^T P(t) dt$$

where

I = impulse, psi-ms.
P(t) = peak overpressure, psi.
T = duration of the positive phase, ms.

Numerical integration of I becomes

$$I = \frac{M \times Q}{KA \times US}$$

where

- I = impulse, psi-ms
- M = area under the positive phase of the pressure time history.
- Q = calibration charge, micromicrocoulombs.
- KA = gauge sensitivity, micromicrocoulombs/psi.
- U = time scale factor, scale units/ms.
- S = average voltage calibration step size on the record.

The equations for pressure and impulse versus distance were computed by the least square method from the data for the explosive phase tests:

$$P = 894 R^{(0.617283 \log_{10} R - 2.76141)}$$

and

$$I = 33.5 R^{(0.20191 \log_{10} R - 1.063899)}$$

To produce the same pressure-distance and impulse-distant relationships by a bare explosive, the total detonation energy was decreased by the amount of energy required to break the vessel and the amount necessary to accelerate the vessel fragments. Figure 3-54 shows comparisons made for effective charge weight of bare explosive scaled pressure and impulse versus scaled distance with the relations reported by A. Hoffman and S. Mills, Jr.⁽⁷³⁾ and with those of Brode's⁽⁷⁴⁾ theoretical point source energy curve.

The pressures and impulses in the propellant phase tests appeared to be almost independent of the propellant quantity used; hence, since unburned propellant was recovered, it was assumed that the blast came from the amount burned up to the time of vessel rupture.

A pressure-distance dependence similar to that for the explosive phase is presented in Fig. 3-55. The similitude of the pressure-distance, sizes and shape of fragments recovered and pressure time records signified that the propellant, although erratic, acted like an explosive in the production of pressure waves in air away from the source. An effective weight of explosive required to scale the propellant pressure versus scaled distance curve was determined as before. The energy in Brode's hot gas sphere solution agreed with the propellant curve when using the detonation energy of bare explosive.

FIG. 3-54
SCALED PRESSURE-IMPULSE-DISTANCE RELATIONS
(REF. 72)

FIG. 3-55
 FREE AIR BLAST PEAK PRESSURE AND
 IMPULSE VS. DISTANCE
 PROPELLANT IGNITED WITHIN SIMULATED
 REACTOR CORE VESSELS
 (REF. 72)

In the static phase test firings, the vessel parted at the primacord location, producing two large missiles traveling in opposite directions. Recordings revealed that a shock formed. These tests illustrated that sufficient energy was available to form a shock; however, the static pressure burst energy was considerably less than the propellant or explosive from the vessel fragments.

Physical evidences at the test site appear to corroborate the contention that propellants and explosives deform and fragment the test vessel in the same manner.

The three methods used on the tests encompassed the entire range of rates of nuclear energy releases possible. The explosive charge method provided the best reproducible imitation. The propellant method did simulate a slower reaction rate, gave a definite measurable shock and shock decayed with distance analogous to a bare explosive, but reproduction was not as easy as the explosive method. The static method reproducibility, as expected, was extremely poor because the available energy of the compressed gas is less than the energy released from a propellant or explosive. Good agreement was obtained with Brode's point source solution ⁽⁷⁴⁾ and with Hoffman and Mill's data ⁽⁷³⁾ on the explosive method. The propellant method results were predictable on the basis of assumptions made for the test in the comparisons.

The results of these BRL tests are not directly applicable to containment problems except in considering bare unshielded reactors such as might be encountered on critical experiments. Also the biological shielding normally provided imposes a very large inertial impedance and energy attenuator, so that the shock waves formed probably would not act on the containment vessel.

The media usually encountered directly around the reactor are arranged in complex combinations and geometries. Accepting explosives and propellants as adequate energy source simulations, experimental confirmation of design and determination of close-in damage phenomena have led to proposals for large scale testing.

This experimental approach to proving and improvement of actual designs was employed extensively with respect to the Air Force Nuclear Engineering Test Reactor (AFNETR) which has since been built. Success of this test program warrants considerable attention in this review. The tests are reported comprehensively by Baker and Patterson ⁽⁷⁵⁾ A one-quarter scale model of the reactor plant, except for the containment shell, was constructed at Aberdeen Proving Grounds with careful attention to the geometric scaling, even that of aggregates and reinforcing steel. The procedure was to release energy in the core section first from propellants and then from high explosives. Ever

increasing amounts were employed until major damage was experienced in the model. The propellant tests were conducted first, with minor results, up to the point where the quantity of propellant employed filled the entire core volume. The tests were continued with H.E., dropping back to a charge weight representing less than the maximum propellant charge on an equivalent energy basis. The maximum energy release reached corresponded to a 500 megawatt-second total energy release on the full-scale reactor, based on an assigned equivalent scaled energy value.

To understand the validity of the test and the problems involved in interpreting the damage, a discussion of the loading and structure response scaling is pertinent. Descriptions of the qualitative spatial behavior of the pressure with time, away from the detonation region, are available⁽⁷¹⁾ for media of liquids, solids, and gases. For an explosive, approximately one-third of the total heat of combustion is released during the detonation process, originating the initial pressure wave, and the remaining two-thirds is released during afterburning, if sufficient oxygen is available. Afterburning is a relative slow process and requires an excess of oxygen; thus, its contributions to the damaging potential of an explosive are usually small. In an underwater detonation of an explosive, for example, no excess oxygen is available and only the detonation heat is released.

A cube-root scaling law has been applied to ordnance problems and verified many times over the past years. Many general characteristics of explosions can be described by sample relationships holding over a very wide range of energy release. The peak overpressure and the impulse (time integral of the pressure at a point) divided by the cube root of the charge weight for a blast wave are uniquely established by the distance from the explosive center divided by the cube root of the charge weight. This is predicted by dimensional analysis and has been experimentally verified. This scaling applies to pressure waves traveling through a homogeneous medium and/or normally reflected from obstructions. The transient pressure duration, the time interval from peak pressure to ambient pressure, scales analogous to impulse. The pressure-time histories at the same scaled distances from two different size explosives have the same peak overpressure, but the duration varies directly as the linear scale factor.

The above pertains to the explosion. Regarding structural response, some difficulty exists in test-for-test confirmation of theoretically predicted responses by experiment. However, on a comparative basis it is found that responses are consistent. The response of a scaled-down structure subjected to an internal blast load compared to response of a full-scale structure has been treated theoretically. The analysis is based on equations of motion using presumptions that viscosity and thermal conduction are nil in the fluid media and the gravity and strain-rate effects are small for the solid media. It has been shown that, for a geometrically-scaled model subjected to a geometrically-scaled explosive charge, the displacement and

strain-time histories will be similar to that of the actual structure with the strain amplitudes unchanged, but the time and displacement will be compressed by the linear dimensional scale factor. For accurate duplication, it is necessary that all materials, densities, and elastic and plastic properties of the original be duplicated in the model.

[A word is also necessary regarding simulation of the actual energy release. If the real energy release is by fission chain reaction, a characteristic energy release is obtained which increases approximately exponentially with time to half-maximum power and then increases at an exponentially decreasing rate to maximum. About 95% of the energy is released in about seven exponential periods. An exponential period can vary from a few milliseconds to a few microseconds. The effective characteristics of energy release from propellants and explosives are different. These increase monotonically to completion of detonation or burning, then decrease abruptly to afterburning rates or zero. However, total release times for explosives are comparable to those for a fast reactor, and release times for propellants are comparable to those of thermal reactors. It is currently considered that the differences in shapes of the energy-time curves are not significant. --- Editor]

The authors⁽⁷⁵⁾ point out that high explosives have very high detonation rates and that the extremely rapid energy release subjects the surrounding media to severe shock waves regardless of the explosives' confinement. On the other hand, the attainment of a significant pressure increases with propellants requires attention to mechanical and inertial restraint.

For the type of reactor under test, the rate of energy release is less than that of an explosive by several orders of magnitude. The burning of properly confined propellant, then, would more closely simulate the actual case. However, the use of an explosive will probably always produce a greater effect close-in than will an equal amount of energy release at a slower rate. Therefore, in this application it is believed that an explosive will produce conservative results. [There is some disagreement regarding conservativeness further away from the source --- Editor]

Four propellant tests simulating full-scale energy releases between 6×10^6 and 7×10^7 calories and 6 high explosive tests covering the full scale range from 5×10^6 to 2.5×10^8 calories were performed. Transient loads experienced, and those predicted from free underwater and free air blast data for various parts of the model were obtained by the authors. In addition, impulses on the thermal columns roll-door have been inferred from its maximum momentum and compared to predicted underwater blast impulses. The data is too extensive to reproduce here so that the reference must be consulted directly.

It is of sufficient interest, however, to give the discussion and conclusions of the authors in some detail.

The power or energy excursions from the BORAX-II experiment cannot be accurately simulated by an explosive since the detonation of an explosive gives an initial power rate of extreme rapidity, independent of the explosive quantity used. In a given geometry configuration, the peak power can be diminished only by reducing the explosive charge weight and this decreases the power time duration. By using propellants to simulate a nuclear excursion, it is possible to approximate reasonably the power-time and energy-time curves. [For thermal reactors --- Editor] The total energy released by a propellant can be regulated by varying the type as well as the quantity and the ratio controlled by using slow-burning or fast-burning propellants.

The validity of the model test results depends directly upon the soundness of the scaling laws. The deletion of gravity loads when compared with the transient pressure loads is an important assumption in the response scaling laws. The gravitation forces were not scaled into the results as it was believed that no significant error would be introduced.

Large plastic deformation and fracture of ductile parts of the model may not scale properly.

The concrete damage caused by round 7 imposed a limitation on the test results as the remaining tests were conducted on a weakened structure.

The impulse-momentum theorem is applicable for predicting the initial velocity of massive parts having no horizontal restraints and subjected to high impulses during the excursion.

An analysis of the top and bottom plugs is possible if the elastic limit of the bolts is not exceeded. A similar analysis is possible for the bolts exceeding the elastic limit; however, the prediction of displacement-time histories is difficult. Neglecting the bolts on the top plug permits a free body analysis; namely, a free body subjected to an impulsive upward load and a downward gravitation force.

The analysis of structural parts undergoing large plastic deformations is unjustified. The analytical prediction of the dynamic response of simple structural elements in the plastic region when compared with experiments is usually in poor agreement. This fact was a contributor in the decision to conduct the tests.

A number of changes have been made in the prototype design since the tests were run. The dome containment shell has been extended by inclusion of an ellipsoidal cap set in concrete beneath the structure. This extension undoubtedly strengthens the containment structure considerably and increases its ability to withstand an incident.

The outer shell thickness in the vicinity of the thermal column roll door tracks has been reduced from 36 in. to 24 in. The extension of the domed shell probably compensates for this decrease in concrete near the thermal column.

Had a steel shell been incorporated in the model, it probably would have prevented the rupture of the outer shell on the final round.

It is planned to reduce the depth of the reactor tank, bulk shielding pool and the concrete in the vicinity of these components by nine feet. The water depth reduction over the core is intended to reduce the maximum credible energy release. This design change is believed to have little effect on the damage of the massive parts. The loading on the walls of the bulk shielding facility would be somewhat reduced while the loading on the top plug would be increased somewhat with the analysis used.

It is anticipated that none of the proposed changes in the design impair the reactor ability to contain the maximum malfunctions. The modified design is quite likely safer than the one incorporated in the model because of the steel containment shell.

For the simulated reactor excursion tests, the majority of the energy is discharged into the thermal column and test cells by the immediate failure of the sides of the center core section facing the test cells. The remaining energy is transmitted upward through the core tank and also through the weak graphite window into the bulk shielding pool. The effects of very large excursions in the prototype, if not modified, are as follows:

If the water level in the core tank is near the top plug, the water mass thrown vertically will impinge upon the plug and cause the hold-down bolts to fail. When the underwater shock reaches the water-air interface of the bulk shield tank only a minute amount escapes into the air; however, a sizeable water spray dome follows. A large surface area bridge across the tank can be hurled against the containment vessel if not securely fastened from the blast pressure and water spray dome. The air blast is not intense enough to damage the containment vessel. A slight static pressure rise will occur in the shell following the excursion.

Water-filled piping connected to the core assembly can be expected to undergo considerable disarrangement, but there will be no occurrence of dangerous fragments. The ejection of control rods or beam tube assemblies from the core appears not probable.

The first level floor slabs will move radially outward and concrete parts of the structure near the core tank will be cracked and split. Damage to the substructure will be practically nil. No major concrete fragments will be thrown with sufficient velocity to pierce the containment shell.

Several minor design modifications are believed necessary to enable handling an excursion of 1000 megawatt-seconds.

The major modification should be a design change in the wing wall between the basement and first floor of the reactor core. A gap in the vertical joint between this wall and the outer shell would prevent outward displacement of the wing wall and consequent breaking of the outer shell, and additional ring reinforcement around the core tank in the concrete would reduce this displacement.

By using stronger bolts, or an interrupted-screw hold-downs on the breech mechanisms of many guns, failure of the top plug hold-down bolts could be eliminated.

The thermal column roll door should be securely locked in position to eliminate its impingement on the outer wall. A buffer could be positioned behind the door to retard the motion if locking is not possible.

The moveable bridge above the bulk shield tank should have a position hold-down at all times.

Direct efforts at simulating the time behavior of a prescribed energy release have been made. Zabel⁽⁷⁶⁾ reports that experiments have been performed with a system consisting of a train of high explosives wound in the shape of a helix. The cross section of the train at any point determines the rate of energy release; the length of the helix determines the duration; and the mass determines the total amount of energy. The energy release curve could be fitted by a series of steps so that the area in each step would equal the area under the curve. No final definite simulations have been obtained at the time of reporting although the method is very promising. The plastic explosive C-3 for sizes down to 3/16 in. in diameter worked very well. There is apparently no problem of transmission of the explosion through junctions of different diameter explosives. The problem of cross-ignition or shock damage to undetonated explosives is solved by placing a half inch of solid steel between the coils.

Some high energy explosive experiments and an analysis of the final BORAX-II excursion were performed to correlate the damage from a nuclear energy release with the damage from an explosion. This is reported by Fye and Noonan⁽⁷⁷⁾ US Naval Ordnance Laboratory, White Oak, Maryland, April 30, 1956. Some basic relations of the cube-root scaling law are applied, and it is concluded that a TNT charge of 6 to 17 lbs. could have resulted in the same damage as was experienced. It is of interest to note that this is only 9 to 25% of the equivalent charge which would be estimated on the basis of a direct conversion of the nuclear energy release to TNT based on the detonation energy of approximately 10^3 calories per gram.

The basic relations for the cube root scaling law are given. For a vessel under pressure, the stresses are directly proportional to the thickness, radius and pressure; hence, they should scale linearly. Experiments show that the peak pressure is proportional to some power of the ratio of the cube root of the charge weight to distance from the charge. To produce the same pressure stresses in the model the ratio of the charge weight required for the model to full-scale charge weight must equal the cube root of the ratio of model radius to the full-scale radius, that is, the cube root of the scaling down factor, e.g., if the model is scaled down by $1/8$, the model charge weight should be $1/512$ of the full scale charge weight.

When the loading time is of the order of the natural period of vibration of the structure, other methods of predicting damage are available from studies of the response of structures to dynamic loading. For shells having a thickness to radius ratio less than $1/10$, a good approximation for the response of a spherical shell is given by Baker and Allen.⁽⁷⁸⁾

It is assumed that the spherically symmetric internally applied pressure pulse is of the form

$$\begin{aligned} p(t) &= 0 & t \leq 0 \text{ and } t \geq T \\ p(t) &= P(1 - \frac{t}{T}) & 0 \leq t \leq T \end{aligned}$$

where p is the pressure, P is the peak pressure, t is the time and T is the duration of the pulse. Then good approximation for deflection of a thin shell is

$$\begin{aligned} U_r &= K(1 - \frac{t}{T} - \cos \omega t + \frac{\sin \omega t}{\omega T}), \quad 0 \leq t \leq T \\ U_r &= A \cos \omega(t-T) + B \sin \omega(t-T), \quad t \geq T \end{aligned}$$

where

U_r = radial deformation

$$K = \frac{P}{pd \omega^2}$$

$$\omega = \frac{2E}{(1-\nu)\rho b^2}$$

$$A = K \left(\frac{\sin \omega T}{\omega T} - \cos \omega T \right)$$

$$B = K \left(\sin \omega T + \frac{\cos \omega T}{\omega T} - \frac{1}{\omega T} \right)$$

b = radius of sphere

E = Youngs' Modulus of Elasticity of shell material

ρ = density of shell material

d = shell thickness

ν = Poisson's ratio of shell material

The problem of determining the ability of the components of the primary structure surrounding the core to alter the energy adequately, before the pressure wave reaches the outside of the structure (biological shield, for example) and results in spalling, must include a determination of the energy absorption by a permanent strain of the reactor vessel or container. This is being given experimental and analytical treatment at the Naval Ordnance Laboratories by W. Wise, Jr.⁽⁷⁹⁾ He develops the general expressions for the energy absorption potentials of a reactor pressure vessel. The reactor-vessel containment work being conducted at the Naval Ordnance Laboratory is predicated upon the thesis that for every excursion-energy release and rate of release there exist particular values of reactor-vessel design parameters which constitute optimum containment design. These parameters are taken to be material, configuration, constraint, and size. The purpose of the program is to determine, through an investigation of these basic parameters, the optimum containment design of nuclear reactor vessels for a large range of excursion energy release, fluxes, and flux geometries. The purpose is being achieved through a fundamental investigation of the structural response of model reactor vessels to simulated excursion-type loading produced upon the internal walls of the vessels.

It is postulated that strain-energy absorption, achieved through the medium of large plastic deformation, is a means for substantially improving the containment potential of nuclear reactors. It follows that strain-energy absorption potential is one of the important containment design criteria.

Consistent with the criterion of strain-energy absorption potential, the general strain-energy equation of dynamic equilibrium for a vessel subjected to internal dynamic loading has been derived in closed form. This general equation and its simplified working forms require knowledge only of the transient pressure and strain vector phenomena accompanying hydrostatic and dynamic dilation.

Using an hydraulic pump for hydrostatic loading and solid propellant for dynamic loading, reactor vessel models have been dilated in the laboratory. The internal pressures and external strain vectors corresponding to the dilations were monitored, and comprehensive analyses of structural response data are presented in tabular and graphical form.

The results obtained prove conclusively that the mechanism of strain-energy absorption can be very effective in improving the containment design of nuclear reactors. It is likewise concluded from the general results of the work that the approach to the problem is technically and economically sound and that it will yield the fundamental technology required to achieve optimum containment design of nuclear reactors.

The conditions necessary to produce spallation at the outside of the primary structure (concrete shield face, for example) are generally known. Essentially the pressure wave intensity must be great enough to produce stresses near the surface of the order of the tensile strength of concrete, unless special design provides metal reinforcing and shields adequately tied to the primary structure. Direct experiments on this specific application have not been reported. However, general behavior has been reviewed by Stanford Research Laboratories and is reported (not available) in the following paragraphs.

The breakup of concrete walls 3 ft. thick by 6 ft. high by 9 ft. long, and 3 ft. concrete cubes, by explosives was examined through Fastex films. The walls were usually broken by a number of cracks radiating from the charge position to the outer boundaries. Pieces broken from the top of the wall were projected up and away from the charge. The material on the other side of the wall and directly opposite the charge was usually broken into small pieces which had a velocity probably in the neighborhood of 50 ft. per second. In general the cubes were broken into small pieces the size of large gravel with the far face of the block in larger pieces.

The missile problem of greater concern is that of specific missiles, such as control rods, sections of pipe, or operating equipment which penetrate the primary structure and may be loaded, broken, and ejected during the energy release in the core. Under some condition, these missiles might penetrate the shell. It appears that the velocities of such missiles will be low. Ordnance experience on penetration, mostly in classified papers, deals with velocities generally above 500 ft/sec.

A detailed experimental study reported by Zabel⁽⁷⁶⁾ has been undertaken by Poulter Laboratories of Stanford Research Institute to fill the gap in information below 500 ft./sec. This has taken the form of scaled experiments on the penetration of steel plates by steel rods, with a high ratio of mass to impact area. The principal result obtained from experiments is the following relation between the thickness, T , of mild steel plate penetrated in inches and the kinetic energy, e , per unit diameter of the rod in foot-lb per in.

$$e = 16,000 T^2 + 1,500 T$$

The formula was determined for plates supported by a rigid backing with a 4-inch opening, a thickness between 0.030 and 0.25 in., and for flat-ended projectiles of drill rod steel with a mass between 1 and 10 oz, a diameter of 3/16 to 1/2 in., and a velocity from 0 to 400 ft/sec. The formula has been extended to cover other values of the parameters by further experiments. The effect of parameter variation has been determined to be as follows:

1. The value of e increases by a factor of 3 as the opening in the rigid backing becomes very large.
2. The value of e increases if the missile deforms from having too small a diameter or yield strength.
3. The value of e increases if penetration occurs at other than normal incidence.
4. $\Delta e/e = \Delta \sigma_u / \sigma_u$ for steels other than mild steel where σ_u is the ultimate tensile strength of mild steel.
5. The value of e decreases by one-half for static conditons.
6. The value of e increases for cylindrical rods with conical tips in order to achieve full penetration, but acutely pointed rods can produce small perforations at appreciably lower energy than that required of flat-ended rods for penetration.

Another type of missile problem is typified by possibility of ejection of the large "plugs" usually forming the closure of the upper part of the primary structure above the core. The principal part of the problem is to determine the load-time history for the plug. Since if this is known its behavior can be rather easily predicted. This problem

has been studied specifically for most reactors in which it may exist. It is of interest to report in detail the approaches and treatment of three studies.

The EBR-II plug loading and behavior was analyzed by Sluyter.⁽⁶⁰⁾ In this reactor the plug is separated from the bulk sodium, in which the core is immersed, by a layer of argon gas 18 inches thick. The study led to the following conclusions.

1. Acoustic impedance mismatched at the free surface of the sodium reduces the pressure on the plug by two orders of magnitude.

2. The peak pressure exerted on the bottom of the plugs by the argon gas is not more than about 1.4 atm abs acting for less than 2 milliseconds.

3. The peak pressure exerted on the bottom of the plugs by impingement of sodium is not more than about 3.53 atm abs acting for less than 2 milliseconds.

4. The total thrust applied to the plug is sufficiently small to permit the use of a simple plug-to-structure locking devices to prevent plug ejection.

5. The total energy transferred to the top enclosure by mechanisms or components fastened to the closure and extending through the primary tank argon gas blanket into the bulk sodium is insufficient to appreciably strain the top structure.

The pressure on the argon, P_1 , transmitted by the sodium is determined from the Rankine-Hugoniot equation:

$$\frac{\rho_1}{\rho_0} = \frac{\frac{\gamma+1}{\gamma-1} + \frac{P_0}{P_1}}{1 + \left(\frac{\gamma+1}{\gamma-1}\right) \left(\frac{P_0}{P_1}\right)} \quad \text{and} \quad \left(\frac{u}{c_0}\right)^2 = \frac{\left(\frac{P_1}{P_0} - 1\right) \left(1 - \frac{\rho_1}{\rho_0}\right)}{\gamma}$$

where

P = pressure in the argon
 c = acoustic velocity in argon
 ρ = density in argon
 γ = adiabatic exponent for argon
 subscript 0 = region ahead of shock
 subscript 1 = region behind shock

and

$$u = \frac{2P_m}{\rho_0' C_0'} \text{ is the initial sodium velocity at the surface, where } P_m$$

is the peak sodium interface pressure and the primes indicate sodium properties. P_m is obtained from Cole's expression for pressures developed in water.⁽⁷⁰⁾

Having the argon pressure, the peak pressure loading, P_2 , in the plug is given by:

$$\frac{P_2 - P_0}{P_1 - P_0} = 1 + \frac{1 + \frac{\gamma-1}{\gamma+1}}{\frac{P_0}{P_1} + \frac{\gamma-1}{\gamma+1}}$$

The above approach is used if there is a gas buffer zone between the liquid, in which the energy source is immersed, and the plug.

Another approach is used if the gas space is absent and the liquid is in contact with the plug. In this type of study, underwater explosion data is again employed with adjustments made for the sodium media. The residual pressure after the initial shock is calculated by considering the energy subsequent to the shock available for raising the temperature and pressure of gas formed during the release, and of the other gases and materials initially present.

Peak overpressures are obtained from underwater data using the cube root scaling law. The time is further scaled by the ratios of velocity of sound in water to that in sodium. The velocity of sound in sodium at the initial sodium temperature is given by Marshall Sittig.⁽⁸⁰⁾ Using his data, reflected pressures can be obtained. [The underwater explosion data on pressure and impulse as functions of time and distance, along with methods of estimating reflection coefficients data, can be obtained from Cole⁽⁷⁰⁾ or from Effects of Atomic Weapons⁽⁷¹⁾ --Ed.] The pressure-time data can next be integrated to give the impulse. If the hold-downs fail under these pressure loads, the time-motion relations are easily obtained from the force-mass-acceleration equation.

The treatment of additional or residual loadings in this study is more dependent on the prevalent conditions and on assumptions made with respect to individual systems. If gas (air, for example) is housed somewhere within the plugged container, and the liquid medium (sodium for example) is forced into the gas during expansion of the energy source, the gas may be heated, and its pressure raised, loading the plug.

In this case the time-dependent state conditions of the gas may be calculated on the basis of initial states and heat capacities of liquid and gas using a constant volume process. Heating of the gas due to the passage of a shock wave may be included, but is probably unimportant. Additional heat may be released if the sodium burns on contact with the air. If the residual pressure buildup is greater than the weight of the plug per unit area exposed, the plug will be propelled. Once again with the loading time history known, the motion is easily calculated.

Treatment of this second type of configuration where there is no gas space between plug and liquid is given by Fisher and Wise⁽⁸¹⁾ and separately by Hoenig et al.⁽⁸²⁾

The work in NAVORD-5747⁽⁸¹⁾ is currently being supplemented by experimental determination of plug motion.⁽⁸³⁾ These tests cover both shock loading and residual loading. The scaled tests, although they cannot be extended to full scale as yet, indicate that about one-tenth of one per cent of the total energy released goes to moving the plug.

[It might be considered that for an actual reactor most of the material volatilized to form the gas bubble is condensible, and that pressure reduction through recondensation must make the actual case less severe than the case in which the energy release is simulated by use of an explosive charge --Ed.]

3.4.1.4 External Loading

(1) Partially and Completely Underground

Installations

An underground installation serves perhaps three useful purposes. First, such an installation integrates well with natural defense plans in that it can be made almost invulnerable to attack by bombing. The use of national defense funds to pay for the additional cost of going underground might increase the economic attractiveness. Second, it provides a reasonable method of containing the radioactive material. Third, shielding is provided in the event that the radioactive material escapes from the primary reactor system to the contained gas environment of the plant.

Experience in this country with respect to underground installation is nil. However, a short general proposal was made by Ringo and Inglis⁽⁸⁴⁾ which led to an extensive study by Christian Beck.⁽⁸⁵⁾ Foreign experience is available in the form of a Norwegian Paper by Aamodt⁽⁸⁶⁾ and a Swedish paper by Carlbon, et al.⁽⁸⁷⁾

These references are reviewed here:

The report of Ringo and Inglis presents a rational discussion of the special considerations of site selection and design of underground power reactors. A rough cost analysis for a 150-megawatt reactor indicates the estimated additional cost for underground installation to be the excavation cost only, or about 3.5% the total cost of the plant.

Christian Beck's study⁽⁸⁵⁾ outlines and evaluates the factors, advantages, disadvantage and cost of a nuclear power plant constructed underground relative to aboveground methods. Briefly the advantages and disadvantages appear to be:

1. In the event of a nuclear incident, effective containment is provided by underground rock excavation, especially when leak proof lining is incorporated.
2. Considerable reduction in maintenance work is achieved with regard to the exterior portions of the aboveground reactor plant.
3. Underground building is void of extreme temperature changes, atmospheric disturbances such as tornadoes, blizzards, etc., and relatively invulnerable to earthquake shock.
4. Ventilation requirements are relatively light.
5. Excellent protection is provided against blast and radiation effects due to an enemy attack.
6. The cost for underground construction in rock will generally be greater than a contained aboveground plant and the increase may be estimated to range from 3 to 7% of the total cost of the installation.
7. There is uncertainty of establishing the exact condition of the underlying formation in advance of excavation.
8. The flexibility is limited with respect to access and expansion.
9. The reactor plant must be located near a sufficient water supply or a forced fan cooling tower system must be incorporated.
10. In the event of an accident, there is a small possibility of contaminating the ground water.

The study goes on to give the extensive reference data that are available on underground construction in rock. Most of

the data is related to underground hydroelectric stations and protective structures for an enemy attack; however, some data exists on reactor plants being constructed or constructed underground in Norway and Sweden.

Site possibilities are ample, and surveys by the Bureau of Mines have been conducted. For underground installations, the best geological formations are considered to be limestone, granite, and sandstone as they permit large spans, high ceilings, and horizontal entries.

A detailed cost analysis was made of a hypothetical underground EBWR and compared with the actual EBWR installation costs. The unit cost of rock excavation is expected to go down with increasing volume, thus, the cost on a unit basis in constructing an underground reactor plant improves with increasing size. Based on excavation from rock, the study shows plant cost increase of 3% to 7% are to be expected. If the cost of excavation is eliminated by selection of naturally occurring underground regions, the cost may well be lower than for aboveground installation.

Aamodt's paper⁽⁸⁶⁾ gives a description of the civil engineering design, construction, and costs of the Norwegian boiling heavy water reactor at Halden. The erection costs of the underground reactor building are compared with the possible costs of an ordinary concrete and masonry building, and a steel shell building.

The reactor has a thermal rating of 20 MW. The reactor site is about 1 Km from the center of Halden (population 20,000). Within 0.5 Km from the reactor site, 2,000 to 2,500 are living or working. Within 1.5 Km from the reactor site, 12,000 people are living.

The main requirement of the reactor building was that it should be explosion-safe and gas-tight in order to contain radioactivity released in case of a reactor accident.

A suitable, cheap and reliable solution for Norwegian conditions was found to be an underground location. The following information was obtained:

1. Explosion resistance can be obtained.
2. The rock itself is tight although the tightness properties are unknown. The cracks in the rock can be filled with a material with good filtering and ion-exchange properties.
3. The inside rock surfaces can be lined with a tight concrete surface which seals the cracks in the rock.

4. Leakage from the reactor will be delayed by the 100 to 200 ft of rock cover above the reactor.

5. Excavation of rock is a well-known technique in Norway and is competitive there with aboveground construction.

6. If later found necessary, a thin tight seal, supported by the explosion resistance surfaces, can be built inside the hall.

7. By pressurizing the complete hall, the rate of leakage before starting the reaction can be determined.

The control and service area is above ground level. The reactor plant area is under ground level. The control building contains the control room, subcontrol room, battery room, mechanical and electronic workshops, offices, and wardrobes. The service building contains emergency diesel generators, control of feedwater intake, retention tank, sampling area for drainwater, garage and storage.

A pre-tunnel was built at the tunnel entrance for protection from snow slides or eventual stone falls from the sides of the open cut. The building contains the feedwater makeup and fresh air intake. An underground cable and walking tunnel connects the pre-tunnel, the subcontrol room, and the retention tank area.

A 164-foot entrance tunnel, 15 ft wide and 20 ft high, connects the pre-tunnel and the reactor hall. The tunnel is horizontal with a section angled at above 45 degrees near the reactor hall to establish an explosion pocket. The cross section dimensions of the tunnel were established by the dimensions of the largest piece of equipment to be transported through it.

The reactor hall is 33 ft wide by 98 ft long with a total height of 85 ft of which 38 ft are above the main floor. The rock surfaces of the hall and tunnel are lined with a minimum of 6 in. of concrete. Due to the rough rock surface, the average thickness is 12 in. The reactor tank is mounted below the main floor level in the middle of the hall. The well has a minimum wall thickness of 8 ft of concrete and is lined with a water-filled steel cylinder.

Behind the reactor well are two installation floors and a pump basement, all below the main floor level. In front of the reactor are a fuel storage well and waste well, both concreted and steel lined.

A 50-ton overhead crane serves the entire main floor and by an opening in the main floor, the installation and other floors.

A sprinkler system for washing the walls, ceiling and air inside the reactor hall and an airlock in the entrance tunnel will be installed.

The planning of an underground plant in rock is dependent on the type and quality of the rock, the homogeneity of the rock, extent, and distribution of cracks, water, or gas content and leakages. The building of a plant is dependent on the people experienced in this field.

Tensile stresses in rock should be avoided, which means the roof should be shaped like an arch or dome and be as smooth as possible. The rock limited the span width of the Halden reactor hall to 35 ft. The rock is good but is weakened by cracks. With good rock the span width might be considerably increased.

A total of 11,600 cu yd of rock was excavated of which 1800 cu yd was in the open cut in front of the entrance tunnel, 2500 cu yd was in the tunnel, and 7300 cu yd was in the reactor hall.

The width of the hall was restricted by rock conditions but the length and depth were not restricted. The facilities were arranged symmetrically along the axis of the hall giving a total length of about 100 ft. The dimensions of the components together with necessary transportation and mounting space determined the maximum height of 85 ft.

The hall is equipped with both a fresh air and re-circulating air system. The fresh air system, which is normally in use, can be closed in one to two seconds by pneumatic operated valves in the air lock. The valves are operated either by a monitor in the exhaust duct, the neutron flux, the temperature in the hall, or the fall-out of the main current. Tubes through the air locks are in cases cast into the walls. The cases have flanges for the tightening of gaskets against the tubes. Cables are put through the walls in U-shaped openings which will be sealed with a hardening material. The air lock has concrete walls approximately 3 ft thick with steel doors.

The net volume of the reactor hall inside the air lock is 162,000 cu ft. A sudden rupture of the pressure vessel gives a dynamic pressure of 43 psi. Additional shock waves from subsequent chemical reactions are not considered. The maximum static over-pressure is 21 psi. In this calculation the effect of condensation on the surfaces and an evaporation time of 2.5 sec is taken into account.

The safe construction of the air lock places the burden of containment on the rock, concrete, and paint on the surface of the concrete.

Average construction cost comparisons based on costs in Norway are given as:

Type	Average Cost/cu ft of Gross Volume
Underground Building	\$0.90 to \$1.20
Ordinary Building	\$0.70 to \$0.90
Steel Shell	\$2.10

Excavation costs assumes good rock conditions. Average rock conditions could call for about \$0.07/cu ft more for underground structure.

The report on underground reactor installations by Carlborn *et al*⁽⁸⁷⁾ points out that the Swedish bedrock generally is extremely strong and it is usually possible to excavate large chambers in the rock without much support. Over the past ten years, greatly improved techniques in Sweden have decreased the unit cost of blasting while at the same time increasing the speed of work. The location of a nuclear power plant underground is restricted to rock impermeable to gases. The natures of the ground and surface water are also of great importance to the location.

Of interest is a comparison between the ability of below ground and above-ground construction to meet the following requisites for a reactor building:

1. The building should be able to contain pressures and missiles released during an incident.
2. The building should have a low leak rate.
3. The building should give protection from radiation emitted by fission products in the building after an accident.

Condition (1) can be satisfied by either a steel building or an underground building, but rock can contain higher pressures. For example, 60 ft of overlying rock above a room with a span of 60 ft will hold over 200 psi overpressure. A steel shell lined with concrete can contain missiles as readily as an underground building.

Condition (2) is particularly important with respect to heavy water reactors since long term operation enriches the heavy water with tritium. Tritium radioactivity presents a serious hazard to the ground water. The usual gaseous and airborne activity released in an accident are also present.

In a steel shell, the passage through a hole in the wall is short and the time required to pass through is negligible. However, a rock wall may be over 15 to 50 ft thick, sometimes with an overburden of soil. Here diffusion time will be considerable, allowing for decay of activity. Filtering also will eliminate a great part of all the fission products except gaseous elements.

Condition (3) is difficult to satisfy in a steel shell building while the shielding of 50 ft of rock provide an excellent shield.

The paper makes an assessment for rock reactor enclosures which may be summarized:

- a. Overpressures can be easily held.
- b. Protection against wartime demolition is obtained.
- c. Good shielding for radiation from released activity in case of an incident is obtained.
- d. The overlying rock and soil layers delay and filter the release of activity leaking from the building after an incident.
- e. There is less freedom in site selection.
- f. A thorough control of ground water seems necessary.
- g. Expansion of an underground building is inconvenient.
- h. The dimensions are limited.

Certain special design features of rock containment are discussed:

The reactor and its auxiliaries will be placed in its own chamber while the rest of the power plant will be placed in adjacent chambers or above ground. The rock can be used as radiation shielding within the plant. The rock will withstand expected pressures but special measures should be taken to insure the tightness of the rock.

- a. A gas-tight lining of concrete may be constructed in two possible ways. Either the concrete may be poured directly against the rock or free standing walls and roof can be erected. When the lining is in direct contact with the rock, a drainage system must be provided to control migrating activity and to relieve the lining from outside water pressure. For free standing walls, the interface will serve for drainage purposes.

Concrete can be made gas tight to a certain degree. Experiments indicate, for example, the leakage of air through a 6-inch thick wall is of the order of $10^{-5} \text{ ft}^3/(\text{ft}^2)(\text{sec})$ at a pressure difference of one atmosphere. Application of the leak rate to large surfaces is not practical since local defects such as fine cracks are almost certain to be present. Thus an additional lining of thin steel sheets or plastic will be necessary. The Swedish design, for example, calls for a welded steel liner 5/32-inch thick.

b. High pressure injections of cement emulsions into cracks in the rock can be performed before the chamber is excavated. The method is expensive, time consuming and somewhat unpredictable. Under favorable circumstances it may prove feasible.

c. Some soils such as moraines and clays are quite impermeable to water and gases. If the overburden consists of such soils, the time delay of permeation is increased and the total leakage diminished.

In addition to the information outlined above, there is a detailed discussion of radioactive material problems which must be considered. There is no economic comparison made with above-ground installation, but considerable information on construction costs (in Sweden) is provided.

3.4.2 Radioactive Loading

Information available on Radioactive Loading is presently included in Section 3.2.

3.4.3 References - Area IV

1. R. Brittan and J. Heap, "Edited Proceedings of Containment Conference at Armour Research Foundation, May 1957," (unpublished).
2. R. Brittan, "Some Problems in the Safety of Fast Reactors," ANL-5577, (1956).
3. G. Hansen in "Fast Reactor Control and Safety Meeting - Nov. 10-11, 1954," APDA-105, p. 171.
4. K. Fuchs, "Efficiency for Very Slow Assembly," LASL-596, (1946).
5. G. Hansen, "Burst Characteristics Associated with the Slow Assembly of Fissionable Materials," LA-1441, (1952).
6. W. Stratton, T. Colvin and R. Lazarus, "Analysis of Prompt Excursions in Simple Systems and Idealized Fast Reactors," Paper No. 431, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - Sept. 1958, (United Nations, Geneva, to be published).

7. H. Bethe, "Theoretical Estimate of Maximum Possible Nuclear Explosion," KAPL-294, (January 1950).
8. R. Brittan, et al., "Hazard Evaluation Report on the Fast Reactor Zero Power Experiment (ZPR III)," ANL-FWT-105, (April 1955, decl. February 13, 1957).
9. H. A. Bethe and J. H. Tait, "An Estimate of the Order of Magnitude of the Explosion When the Core of a Fast Reactor Collapses," UKAEA RHM(56)/113 (1956).
10. W. McCarthy, D. Okrent and V. Jankus, "Studies of Nuclear Accidents in Fast Power Reactors," Paper No. 2165, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - September 1958, (United Nations, Geneva, to be published).
11. J. Ulrich, "Results of Recent Analyses of BORAX II Transient Experiments," ANL-5532 (April 1956).
12. O. J. Elgert, "A Proposal for Aluminum Water Reaction Experiments in the MTR," IDO-16141 (December 1953).
13. O. J. Elgert, T. J. Boland and G. L. Smith, "Status Report Aluminum Water Reaction Experiment," PPC-89 (March 1955).
14. O. J. Elgert and A. W. Brown, "In-Pile Molten Metal Water Reaction Experiments," IDO-16257 (June 1956).
15. W. N. Lorentz, "Chemical Reaction of Zirconium-Uranium Alloys in Water at High Temperatures," WAPD-PM-22 (del.) (July 1955).
16. W. F. Witzig, "Short Time Autoclave Tests in the MTR," WAPD-P-513, (September 1954).
17. A. W. Lemmon Jr., "Studies Relating to the Reaction Between Zirconium and Water at High Temperatures," BMI-1154 (January 1957).
18. W. A. Bostrom, "The High Temperature Oxidation of Zircaloy in Water," WAPD-104, (March 1954).
19. H. M. Higgins, "A Study of the Reaction of Metals and Water," AECD-3664 (April 1955).
20. H. M. Higgins, "The Reaction of Molten Uranium and Zirconium" AGC-AE-17 (April 1956).
21. H. M. Higgins and R. D. Schultz, "The Reaction of Metals in Oxidizing Gases at High Temperatures," IDO-28000 (April 1957).
22. G. Long, P. T. Shoup and W. T. Ennor, "Explosions of Aluminum and Water," NP-5471 (August 1950).
23. G. Long, "Explosions of Molten Aluminum in Water, Cause and Prevention," Metal Progress, 71, 107 (1957).

24. A. S. Russell, "Aluminum Water Explosions," Memo, Aluminum Company of America, New Kensington, Pa. (April 1950).
25. J. M. West and J. T. Weills, "Reactor Engineering Division Quarterly Report, September 1, 1950 through November 30, 1950," ANL-4549, (December 1950).
26. J. W. Mausteller, ed., "Progress Report No. 27 for February and March 1955," NP-5601 (April 1955).
27. J. W. Mausteller, ed., "Progress Report No. 28 for April and May 1955," NP-5690 (June 1955).
28. J. W. Mausteller, ed., "Progress Report No. 29 for June and July 1955," NP-5739 (August 1955).
29. W. J. Posey, ed., "Progress Report No. 30 for August and September 1955," NP-5779 (October 1955).
30. W. Milich and E. C. King, "Molten Metal-Water Reactions," NP-5813, Tech. Rpt. No. 44 (November 1955).
31. R. F. Plott, "Reactions Produced by the Electrical Explosion of a Metal Immersed in a Fluid," ANL-5040 (December 1950).
32. J. A. Herickes, P. A. Richardson, M. Weiss, and G. Gelernter, "Zirconium Hazards Research, September 1, 1956 through August 31, 1957," AECU-3642, Summary Report No. 3655, Bureau of Mines (August 1957).
33. M. Markels, Jr., and W. A. Stark, "Fast Transient Heat Transfer in a Clad Element Stagnant Water System," TID-7529 (Part 1) (September 1956).
34. A. J. Bendler, J. K. Roros, and N. H. Wagner, "Fast Transient Heating and Explosion of Metals Under Stagnant Liquids," AECU-3623 (CU-1-58-At-187-ChE) (February 1958).
35. D. C. Layman and H. L. Mars, "Some Qualitative Observations of the Zirconium-Water Reaction at Atmospheric Pressure," KAPL-1534, (April 1956).
36. W. C. Ruebsamen and J. B. Chrisney, "Metal Water Explosions," NAA-SR-Memo-75, (July 1951).
37. W. C. Ruebsamen, F. J. Shon, and J. B. Chrisney, "Chemical Reaction Between Water and Rapidly Heated Metals," NAA-SR-197 (October 1952).
38. V. V. Bondarenko, et al, "Resistance of Metals at High Current Densities," Soviet Physics, JETP, 1, (September 1955).
39. I. F. Kvartskbava, et al, "Electrical Explosions of Metal Wires," Soviet Physics, JETP, 3, (1956).

40. J. A. Anderson, "The Spectrum of Electrically Exploded Wires," *The Astrophysical J.*, 51, 37 (1920).
41. J. A. Anderson and S. Smith, "General Characteristics of Electrically Exploded Wires," *The Astrophysical J.*, 64, 295 (1926).
42. A. E. Barrington, "The Fusing of Wires with Heavy Surge Currents," *Brit. J. App. Phys.*, 7, 408 (1956).
43. B. Lustman, "Zirconium - Water Reaction Data and Application to PWR Loss-of-Coolant Accident," WAPD-SC-543 (May 1957).
44. B. Lustman, "Zirconium-Water Reaction," WAPD-137, (December 1955).
45. H. M. Saltsburg, "Metal-Water Reactions," KAPL-1495 (March 1956).
46. W. W. Allison, "A Preliminary Study of the Pyrophoric Properties of Zirconium Machining," WAPD-TM-14, (September 1956).
47. R. A. Beall, R. W. van Dolah, R. W. Huber, "Report of Explosion Disaster, Titanium Melting Furnace," Mallory-Sharon Titanium Corp., Bureau of Mines.
48. H. M. Higgins and C. A. Congiver, "Progress Report," AGC-AE-28 (Aerojet-Gen'l Corp., November 1956).
49. W. R. Hollander, "An Evaluation of the Zirconium Hazard," HW-44989 (August 1956).
50. C. Anderson and L. H. Belz, *J. Electrochem. Soc.*, 100, 240 (1953).
51. H. Eyring and B. Zwolinski, "The Critical Temperature for Combustion of Metals and Their Alloys," *Rec. Chem. Prog.*, July-October Issue, Wayne University, 96-100 (1947).
52. W. Fassel, *et al*, "Ignition Temperatures of Magnesium and Magnesium Alloys," *J. Metals*, 3, 522-8 (1951).
53. R. M. Rome and T. R. Moffette, "PWR Plant Container Sizing Criteria Studies of Transient Temperature and Pressure in Plant Container Following Primary Coolant System Rupture," WAPD-SC-549 (June 1957).
54. H. B. Ketcham, "A Simple Method for Calculating the Maximum Size of Ductile Rupture in Pressurized Systems," WAPD-TM-56 (July 1957).
55. A. Kolflat and W. Chittenden, "A New Approach to the Design of Containment Shells for Atomic Power Plants," Proc. of the American Power Conference Vol. XIX, p. 651 (1957).
56. H. Isben, J. Moy and A. J. Cruz, "Two-Phase, Steam-Water, Critical Flow," University of Minn., Dept. Chem. Engr., Minneapolis, Minn.
57. H. H. Gott and G. Flook, "Design Conditions for Reactor Containing Sphere," FRDC/P-70 (1954).

58. J. Humphrys, "Sodium-Air Reactions as they Pertain to Reactor Safety and Containment," Paper No. 1893, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - September 1958 (United Nations, Geneva, to be published).
59. "APDA NaK-Water Reaction Tests," Atomic Power Dev. Associates, Inc., Detroit, Mich. (not published).
60. H. Monson and M. Sluyter, "Containment of EBR-II," Paper No. 1892, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - September 1958 (United Nations, Geneva, to be published).
61. W. Siler and L. Zick, "Design Considerations for an Atomic Power Reactor Containment Structure," Proc. of the Second Nuclear Engineering and Sci. Conf., Vol. I, pp. 253-263, J. R. Dunning and B. R. Prentice, eds. (Pergamon Press, N.Y. 1957).
62. E. Hines, A. Gemant and J. Kelley, "How Strong Must Reactor Housings be to Contain Na-Air Reactions?" Nucleonics, 14, No. 10, p. 38 (October 1956).
63. Z. M. Shapiro and T. R. Moffette, "Hydrogen Flammability Data and Application to PWR Loss-of-Coolant Accident," WAPD-SC-545 (September 1957).
64. F. Porzel, "Hydrodynamic Problems of Reactor Containment," Paper No. 434, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - September 1958, (United Nations, Geneva, to be published).
65. F. Porzel, "Design Evaluation of BER (Boiling Experimental Reactor) in Regard to Internal Explosions," ANL-5651 (January 1957).
66. H. Karplus, "The Measurement of the Velocity of Sound in a Liquid Containing Gas Bubbles," ARF Final Rpt. on Proj. A097 (1958).
67. H. Karplus, "Velocity of Sound in Reactor Media," Proj. No. A-114, ARF, Monthly Prog. Rpt. No. 2.
68. S. Katz and R. E. Peterson, "Study of Shock Propagation in Ferrous Metals," NP 6806, Poulter Labs. Tech. Rpt. 010-55 (Stanf. Res. Inst. December 1955).
69. H. D. Mallory, "Propagation of Shock Waves in Aluminum," J. Appl. Phys., 26, No. 5, 555-9 (May 1955).
70. R. Cole, Underwater Explosions, (Princeton Univ. Press, Princeton, N. J., 1948).
71. S. Glasstone, ed., "The Effects of Nuclear Weapons," (U.S.A.E.C., June 1957).
72. R. J. Larson and W. C. Olson, "Measurement of Air Blast Effects from Simulated Nuclear Reactor Core Excursions," WASH-747 (September 1957).

73. A. Hoffman and S. Mills, Jr., "Air Blast Measurements about Explosive Charges at Side-On and Normal Incidence," BRL-988 (July 1956).
74. H. L. Brode, "Numerical Solutions of Spherical Blast Waves," J. Appl. Phys., 26, No. 6, 766-75 (June 1955).
75. W. Baker and J. Patterson, "Blast Effects Test of a Quarter-Scale Model of the Air Force Nuclear Engineering Test Reactor," Rpt. No. 1011 (BRL, Aberdeen Prov. Grnd., March 1957).
76. N. Zabel, "Containment of Fragments from a Runaway Reactor," Tech. Rpts. 1 and 2, SRI Project No. GV-2164, (1958).
77. P. M. Frye and E. C. Noonan, "NOL Reactor Safety Program," NAVORD Rpt. 4286, (April 1956).
78. W. Baker and F. Allen, "The Responses of Elastic Spherical Shells to Spherically Symmetric Blast Loading," WASH-746 (November 1957).
79. W. Wise, Jr., "Optimum Reactor-Vessel Containment Design," NAVORD-5748 (June 1958).
80. M. Sittig, Sodium, (Reinhold Pub. Co., 1956).
81. E. M. Fischer and W. R. Wise, Jr., "Containment Study of the Enrico Fermi Fast Breeder Reactor Plant," NAVORD-5747 (October 1957).
82. S. Hoenig, et al., "Containment Study of the Enrico Fermi Fast Breeder Reactor," ARF Proj. No. D633 (PRDC, Detroit, Mich., 1959).
83. W. R. Wise, Jr., "Monthly Progress Report No. 4. NOL Reactor-Vessel Containment Program," NOL-285 (October 1958).
84. D. R. Inglis and G. R. Ringo, "Underground Construction of Power Reactors," ANL-5652 (January 1957).
85. C. Beck, "Engineering Study on Underground Construction of Nuclear Power Reactors," AECU-3779 (April 1958).
86. N. G. Aamodt, "Underground Location of a Nuclear Reactor," Paper No. 561, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - September 1958, (United Nations, Geneva, to be published).
87. L. Carlbom, et al., "On the Design and Containment of Nuclear Power Stations Located in Rock," Paper No. 172, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - September 1958, (United Nations, Geneva, to be published).
88. J. Heap, "Equilibrium P-V-T Relations for Expanding Liquid-Vapor Systems in a Containment Shell," ANL-5828 (November 1958).

3.5 Area V Load Reduction, Design Innovations, and New Applications

Information on several aspects of load reduction as contained on previous sections. An example is the work on "waste heat" blast shields reported in Section 3.5. Work on this problem and others related to load reduction will appear in Section 3.6.1 in future supplements. Only one item, the "vented cell" is discussed here now.

The treatment of design innovations applying not to load reduction, but rather to economics, will also be included in this Section 3.6.2 in future supplements. The item "New Application" is included to cover extension of consideration of containment to systems other than those considered to date. For example, when information on containment for marine propulsion systems or processing plants is available, it will be presented in this Section 3.6.3 in future supplements.

3.5.1 Load Reduction

3.5.1.1 Vented Systems

It is possible with some reactor systems that, following an accident, chemical energy may be released at such a rate and in such quantity that it is economically unsound to attempt containment with the usual containment pressure vessel. Such an energy release results from burning of liquid metal coolant or melted or vaporized uranium in the air of the enclosure after it has been expelled from the core or cooling system. The pressures and temperatures reached, if confined burning is allowed, can be very high and easily lead to breaching of the housing. If such a situation can exist, the accident can be confined only by providing pressure relief through venting.

This particular problem arose at ANL in designing a critical assemblies facility (Bldg. 315) to house, among others, the ZPR-VI, a fast reactor critical experiment duplicating in many respects the fast reactor critical facility ZPR-III at NRTS. R. Brittan, in an unpublished report on the building design, has proposed a system which allows venting and a satisfactory treatment of the associated problems engendered. This report is reproduced here, in part.

Venting of the system described requires a chain of additional treatment. The radioactive material vented must be later immobilized as by filters. In some instances the filters might require protection from blast effects and against high gas temperatures and velocities. Non-particulate material must be adequately restrained by physico-chemical methods.

A "complete" system, which operates satisfactorily after postulating a fast-slow energy release, has the components shown schematically in Fig. 3-56.

FIG. 3-56
VENTING SCHEME FOR CONTAINMENT CELL

An instance of a fast-slow energy release would be an explosive release of energy in the core, resulting in extremely rapid melting, vaporization, and expansion of the materials of the reactor core to a region between the reactor system and the outer containment structure, followed by burning of the reactor core materials in the air of this bounded region. (It would be a simpler matter of have the contained atmosphere inert with respect to the core materials but if such is not reasonable, a vented system becomes mandatory.) This vented containment scheme, then, could be used directly in connection with a reactor which has a violent enough accidental nuclear and/or chemical energy release to eject the core materials such as uranium, zirconium, or some liquid metal coolant in phase states suitable for rapid oxidation.

The vent cannot be open during the lifetime of the blast pressure wave emanating from the explosion source. The duration of the wave is only a few milliseconds. Hence it is proposed to plug the vent with an inertia block which is free to move out of the vent and is restricted only by its own inertia. It is made massive enough so that, although it begins to move under action of the blast wave, it will not have cleared the vent mouth at the outside of the thick-walled cell before the blast wave has subsided. This feature protects the system beyond the cell during the positive phase of the blast wave. However, the blast pressure has started it moving and, if a pressure build-up occurs as a result of a flash fire, the inertia blocks motion will be enhanced. The time to vent must be estimated in such a way that relief is obtained before the pressure builds up sufficiently to rupture the cell.

Once venting is realized, the hot gases and particulates will rush out of the work cell, at high velocity, sweeping towards the filters. This air motion can carry away many types of filters. If the filters are not carried away, they may be destroyed by the heat in the gas. To circumvent this difficulty, an intermediate cell is proposed which will have a heat sink. The purpose is to remove the energy, both thermal and kinetic,

from the vented gases, cooling them off and slowing them down. The most reasonable form of heat sink seems to be a water spray incorporated in a parallel set of scrubbers. There is an added advantage here in that the scrubbers remove much of the particulate matter.

It is proposed that a sand bed type filter system, made up of graded aggregates of diminishing size be interposed as a course filter between the heat sink cell and the high efficiency filters. This is a rugged stand-by system which can serve as an additional heat sink if the spray system fails. It needs no attention after initial installation. This intermediate filter could be topped by conventional CWS or AEC high efficiency filters to remove the fines.

Noble gases and some volatiles would not be stopped. If they cannot be diluted adequately by use of a stack, a low pressure storage tank can be used on the far side of the filters. By this time the pressure and temperature of the through-gases will be reduced to a point where such storage is possible.

In selecting an inertia block for the plug, it is necessary to determine the motion of the plug and the time required to unplug the vent. During this time after the blast occurs and burning begins, the pressure and temperature increase by a constant volume process. After unplugging, the pressures and temperatures and mass flow behave according to the complex nozzle flow process relations given by Heap.

Initially, and for a very short time, the plug is loaded by the blast pressure wave. This load is characterized as a suddenly applied load diminishing exponentially to ambient in the length of time known as the time of duration of the positive phase (usually a few milliseconds). This loading may overcome friction and accelerate the plug mass. After this early loading has reduced to zero, the friction may bring the plug to a halt (or the plug may have a residual velocity) at the time the pressure begins to increase due to burning. In determining subsequent displacements of the plug, the residual velocity may be added to the problem as an initial velocity.

The analysis of the plug behavior is presented here assuming that the plug motion has stopped after the blast and before pressure due to burning is applied. The symbols used in the analysis are:

- p = pressure in cell at any time
- p_1 = pressure in cell at start of block motion
- p_s = pressure in cell at start of heating
- p_f = pressure in cell as block clears vent

- p_0 = pressure outside of cell
 Δp = pressure rise in cell
 c_v = specific heat of gas in cell at constant volume
 w_s = total weight of gas in cell initially
 T_s = absolute temperature of gas in cell initially
 T_i = absolute temperature of gas in cell at start of motion
 T_f = absolute temperature of gas in cell as block clears vent
 q = heat source strength
 t = time after start of heating
 t_i = time at which block begins to move
 t_f = time at which block clears vent
 g = gravitational acceleration
 A = cross-sectional area of vent
 ℓ = length of vent to be cleared
 μ_s = static coefficient of friction for block
 μ_d = dynamic coefficient of friction for block
 γ = specific weight of block
 $\tau = (t - t_i)$

$$C = \frac{qp_s g}{c_v w_s T_s \ell \gamma}$$
 x = displacement of plug at any time from start of motion

$$F = t_i \left(1 - \frac{\mu_d}{\mu_s} \right)$$

$$a = 3x/C$$

If one has a vent with an inertia block just filling it, the friction force to be overcome before motion can begin is $A\ell\gamma\mu_s$. This requires a net pressure force $(p_i - p_0)_s A$, or a value $(p_i - p_0)_s = \ell\gamma\mu_s$. Immediately after start of motion, the friction force to be balanced becomes $A\ell\gamma\mu_d$, which requires that $(p_i - p_0)_d = \ell\gamma\mu_d$. The net force at start of motion is then $A\ell\gamma(\mu_s - \mu_d)$. Now the internal pressure in the cell after heating starts and until the inertia block clears the vent (that is, travels a distance ℓ) is

$$p = p_s \left(1 + \frac{q}{c_v w_s T_s} t \right)$$
, assuming a constant-volume process. The pressure rise is then $\Delta p = \frac{qp_s t}{c_v w_s T_s} = (p - p_s)$ after $t = 0$. At $t = t_i$, $(p_i - p_s)$ is then $\frac{qp_s}{c_v w_s T_s} t_i$

and equals the static frictional resistance $A\ell\gamma\mu_s$. Immediately, the motion

starts and the frictional resistance reduces to a fraction μ_d/μ_s of the static value or $\frac{qp_s}{c_v w_s T_s} t_1 \frac{\mu_d}{\mu_s}$. The driving force then jumps from zero to $\frac{qp_s A}{c_v w_s T_s} t_1 \left(1 - \frac{\mu_d}{\mu_s}\right)$ at $t = t_1$, and continues to rise at a rate of $\frac{qp_s A}{c_v w_s T_s} / (\text{unit time})$.

The force acting on the inertia block after $t = t_1$ is then:

$$\frac{qp_s A}{c_v w_s T_s} \left[t - \frac{\mu_d}{\mu_s} t_1 \right] = \text{mass} \times \text{accel.} = \frac{A \ell \gamma}{g} \frac{d^2 x}{dt^2}.$$

Then, substituting τ , F , and C ,

$$\frac{d^2 x}{d\tau^2} = C (\tau + F) = \text{acceleration of plug.}$$

Integrating once yields

$$\frac{dx}{d\tau} = C \left(\frac{\tau^2}{2} + F\tau \right) = \text{velocity of plug (plug initially at rest).}$$

A second integration then yields

$$x = C \left(\frac{\tau^3}{6} + \frac{F\tau^2}{2} \right) = \text{displacement of plug,}$$

from which can be obtained, by algebraic solution of the cubic equation,

$$\tau = \left[a - F^3 + \sqrt{a^2 - 2aF^3} \right]^{1/3} + \left[a - F^3 - \sqrt{a^2 - 2aF^3} \right]^{1/3} - F.$$

For $p_s = p_0$, Δp (from start of heating to start of motion) = $\ell \gamma \mu_s = \frac{qp_s t_1}{c_v w_s T_s}$

and $t_1 = \frac{\ell \gamma \mu_s c_v w_s T_s}{qp_s}$, so that $t_f = t_1 + \tau_f$,

where $\tau_f = \tau$ at $x = \ell$.

A brief numerical example follows. If

$q = 3.97 \times 10^4 \text{ Btu/sec}$, $p_s = p_0 = 14.7 \text{ psi}$, $g = 386 \text{ in./sec}^2$,

$c_v = 0.1715 \text{ Btu/(lb)(}^\circ\text{R)}$, $\ell = 48 \text{ in.}$, $A = 576 \text{ in.}^2$, $\gamma = 0.0868 \text{ lb/cu in.}$,

$T_s = 520^\circ\text{R}$, $w_s = 4590 \text{ lb}$, $\mu_s = 1/3$ and $\mu_d = 1/2 \mu_s$:

$t_1 = 0.972 \text{ sec}$ (start of heating to start of motion)

$$C = \frac{qp_s g}{c_v \ell \gamma w_s T_s} = \frac{g \mu_s}{t_1} = 132.2 \text{ in./sec}^3$$

$$F = t_1 \left(1 - \frac{\mu_d}{\mu_s} \right) = \frac{1}{2} t_1 = 0.486 \text{ sec}, \quad a = \frac{3x}{C} = \frac{3 \times 48}{132.2} = 1.088 \text{ sec}^3.$$

Substituting these values for a and F in the algebraic solution of the cubic equation above, $\tau_f = 0.952$ sec. Then

$$t_f = t_1 + \tau_f = 0.972 + 0.952 = 1.924 \text{ sec.}$$

$$\text{Velocity of block as vent opens} = 132.2 \left[\frac{0.952^2}{2} + (0.486)(0.952) \right] / 12 = 10.1 \text{ ft/sec.}$$

$$\text{Acceleration of block as vent opens} = 132.2(0.952 + 0.486) / 12 = 15.9 \text{ ft/sec}^2.$$

$$\Delta p @ t_1 = \ell \gamma \mu_s = 1.39 \text{ psi}$$

$$\Delta p @ t_f = (\Delta p)_1 \frac{t_f}{t_1} = 1.39 \frac{1.924}{0.972} = 2.75 \text{ psi.}$$

Peak pressure as vent opens is $14.7 + 2.8 = 17.5$ psi,

$$T_1 = T_s \frac{p_1}{p_s} = 520 \frac{16.1}{14.7} = 570^\circ\text{R},$$

$$T_f = T_s \frac{p_f}{p_s} = 520 \frac{17.5}{14.7} = 620^\circ\text{R},$$

and

$$\text{Kinetic energy of plug} = \frac{MV^2}{2} = \left(\frac{4 \times 4 \times 150}{2 \times 32.2} \right) (10.1)^2 = 3800 \text{ ft-lb.}$$

From this point on, the interdependence of performance of the basic components or elements of such a system makes it necessary to know the time dependent thermodynamic relations of pressure, temperature, and density, along with molten parameters of acceleration and velocity everywhere in the system.

A detailed procedure has been organized by J. C. Heap which relates these quantities for a vented cell with heat source or sink. His treatment was published in ANL-5987.

The problem must now be handled using his treatment to determine the mass flow rates and velocities into the heat sink, the rate of heat removal required, and the mass flow and velocities into the filter system.

Heap has found that for diminutive vents it is possible for the pressure rise to continue even after venting occurs. However, for realistic heat sources, a vent of a few square feet will always lead to a pressure decrease immediately on venting.

4. U. S. EXPERIENCE WITH CONTAINMENT OF REACTOR SYSTEMS

In this section an attempt is made to summarize the experience in design, fabrication, and testing of reactor containment systems in the United States.

The summary is principally contained in tables on the next few pages. The tables do not pretend to be complete either with respect to presenting all reactors or all available information about the reactors listed. It is hoped that the table may be completed in future supplements. Aspects of containment which are awkward to tabulate are discussed in the text following the table.

This summary is an extension of that given by Brittan and Heap, using their sources of information and others which have become available. These reference sources are given for each reactor in a listing following the text.

The reactors are indicated symbolically. These reactor symbols are identified in the Symbolic Listing presented at the beginning of the report.

4.1 U. S. Reactor Containment Data

SYMBOL	NAME	LOCATION	NUCLEAR THERMAL POWER kw	TYPE	CONTAINMENT TYPE (SEE FIG. 4-1)
EBWR	EXPERIMENTAL BOILING WATER REACTOR	LEMONT, ILLINOIS	20,000	BOILING WATER - POWER	II
VBWR	VALLECITOS BOILING WATER REACTOR	PLEASANTON, CALIFORNIA	30,000	BOILING WATER - POWER	II
DRESDEN	DRESDEN NUCLEAR POWER STATION	MORRIS, ILLINOIS	626,000	BOILING WATER - POWER	I
APPR-1	ARMY PACKAGE POWER REACTOR NO. 1	FORT BELVOIR, VIRGINIA	10,000	PRESSURIZED WATER - POWER	III
PWR	PRESSURIZED WATER REACTOR (SHIPPINGPORT ATOMIC POWER STATION)	SHIPPINGPORT, PENNSYLVANIA	231,000	PRESSURIZED WATER - POWER	IV
YANKEE	YANKEE ATOMIC ELECTRIC CO. REACTOR	ROWE, MASSACHUSETTS	392,000	PRESSURIZED WATER - POWER	I
INDIAN POINT	CONSOLIDATED EDISON THORIUM REACTOR	INDIAN POINT, NEW YORK	500,000	PRESSURIZED WATER - POWER	I
GETR	GENERAL ELECTRIC TEST REACTOR	PLEASANTON, CALIFORNIA	33,000	WATER MODERATED - TESTING	II
ENRICO FERMI	ENRICO FERMI ATOMIC POWER PLANT	LAGOONA BEACH, MICHIGAN	300,000	FAST BREEDER - POWER	II
EBR-II	EXPERIMENTAL BREEDER REACTOR NO. 2	NRTS, IDAHO	62,500	FAST BREEDER - POWER	II
HRT	HOMOGENEOUS REACTOR TEST	OAK RIDGE, TENNESSEE	5,220	AQUEOUS HOMOGENEOUS - POWER	IV
WEST MILTON	GENERAL ELECTRIC CO. KAPL REACTOR TEST FACILITY	WEST MILTON, NEW YORK		SODIUM COOLED INTERMEDIATE -POWER	I
MITR	MASSACHUSETTS INSTITUTE OF TECHNOLOGY REACTOR	CAMBRIDGE, MASSACHUSETTS	1,000	HEAVY WATER - RESEARCH	V
PRTR	PLUTONIUM RECYCLE TEST REACTOR	HANFORD, WASHINGTON	70,000	HEAVY WATER - TESTING	II
CP-5	ARGONNE RESEARCH REACTOR	LEMONT, ILLINOIS	2,000	HEAVY WATER - RESEARCH	V
LPTR	LIVERMORE POOL TYPE REACTOR	LIVERMORE, CALIFORNIA	1,000	TANK - RESEARCH	V
NASA	NATIONAL AERONAUTICS SPACE ADMINISTRATION REACTOR	SANDUSKY, OHIO	60,000	TANK - TESTING	V
AFNETR	AIR FORCE NUCLEAR ENGINEERING TEST REACTOR	DAYTON, OHIO	10,000	TANK - TESTING	II

FIG. 4-1

REPRESENTATIVE TYPES OF CONTAINMENT SYSTEMS

4.1 U. S. Reactor Containment Data (Contd.)

	DIAMETER (FT)	LENGTH OR HEIGHT (FT)	DISTANCE GRADE TO BOTTOM (FT)	PLATE THICKNESS (IN.)			MATERIAL	ASME CODE STAMP	GROSS VOLUME (CU FT)	FREE VOLUME (CU FT)
				MAIN	TOP	BOTTOM				
EBWR	80	119	-56	0.625 ^a	0.375 ^c	0.625 ^d	A201-A300	NO	497,500	400,000
VBWR	48	110	-35	0.875 ^a	0.436 ^c	0.436 ^c	SA212		160,000	
DRESDEN	190		-39	1.25 ^b upper 1.40 ^b lower			A201-A300	YES	3,600,000	2,880,000
APPR-1	32 (ID) 36 (OD)	52 64	-10 -20	0.125 ^a 0.875 ^a	0.375 ^c 0.500 ^c	0.375 ^c 0.500 ^c	SA212		52,700	37,000
PWR	2-50 ^e 1-50 ^e 38 ^b	97 147	-52	1.25 ^a 1.25 ^a 0.75 & 1.00 ^b			A201-A300	YES	635,000	473,000
YANKEE	125		+24	0.875 ^b upper 1.250 ^b lower			A201-A300	YES	1,020,000	840,000
INDIAN POINT	160		-75.5	0.89 ^b upper 1.03 ^b lower			A201-A300	YES	2,140,000	
GETR	66	105						NO	322,000	230,000
ENRICO FERMI	72	120	-45	1.03 ^a	0.52 ^c	1.03 ^d 1.25 knuckle line	A201-A300	YES	415,800	
EBR-11	80	139	-47.5	1.00 ^a	0.50 ^c	1.00 ^d	A201	NO	595,000	450,000
HRT	54 x 30.5 x 19 PARALLELEPIPED						A285 (partly clad)	NO	28,100	25,000
WEST MILTON	225		-38	0.9 ^b upper 1.1 ^b lower			A201-A300	NO	6,000,000	
MITR	74	50	-20				A283	NO	258,700	
PRTR	80	121.5	-46.5	0.375 ^a	0.625 ^c	^d	A201-A300		504,500	375,000
CP-5	70	48	0				CONCRETE	NO		
LPTR	80	52	0	0.250 ^a	0.3125 ^c			NO	257,500	
NASA	100	109	-56	0.750 ^a	0.750 ^d	0.750 irregular	A285-54T	NO	520,500	
AFMETR	80	160	-48	0.563 ^a	^c	^d	A201-A300	NO		

- (a) CYLINDRICAL
(b) SPHERICAL
(c) HEMISPHERICAL
(d) HEMI ELLIPSOIDAL

- (e) AXIS OF CYLINDER IS HORIZONTAL
(f) HYDROSTATIC

4.1 U. S. Reactor Containment Data (Contd.)

COOLANT PROPERTIES ASSUMED FOR CALCULATION OF STORED ENERGY				ENERGY RELEASES IN ACCIDENT YIELDING DESIGN PRESSURE ON CONTAINER, 10 ⁶ BTU			POSITIVE DESIGN PRESSURE	NEGATIVE DESIGN PRESSURE	STRENGTH TEST PRESSURE (PNEUMATIC)	
POUNDS AND TYPE	INITIAL PRESSURE, psia	INITIAL TEMPERATURE °F	STORED	CHEMICAL	NUCLEAR AND OTHER ADDED NET DURING ACCIDENT	psig	psig	psig		
EBWR	16,800 H ₂ O	600	488	8	3.2	NONE	15	0.5	18.75	
VBWR	35,000 H ₂ O	1000	545	20	NONE	NONE	45		56.25	
DRESDEN	376,000 H ₂ O	1000	545	200	NONE	NONE	29.5	1	37	
APPR-1	12,000 H ₂ O	1500	600	5	NONE	2.4	63.3		30	
PWR	145,000 H ₂ O	2000	523	78	NONE	4.0	52.8		70	
YANKEE	186,000 H ₂ O	2150	518	94	NONE	NONE	34.5		40	
INDIAN POINT	163,000 H ₂ O	1500	500	80	NONE	NONE	27.5	1	34.4	
GETR							5	0.2	6.35	
ENRICO FERMI	SODIUM - NON PRESSURIZED			DESIGNED ON SODIUM-AIR REACTION PLUS SHUTDOWN NUCLEAR HEATING			32	2	40	
EBR-II	SODIUM - NON PRESSURIZED			NONE	18	NONE	24	1	30	
HRT	3910 D ₂ O	2000	572	0.35	NONE	NONE	30		32 (f)	
WEST MILTON	SODIUM - NON PRESSURIZED						19.75	0.8	25	
MITR	NON PRESSURIZED H ₂ O				2.2	0.13	2	0.1		
PRTR	18,100 H ₂ O OR D ₂ O	1050	504	{ 9	7	3.5	TWO BASES	15	0.58	18.75
CP-5	NON PRESSURIZED D ₂ O				DESIGNED ON BAROMETRIC PRESSURE DIFFERENCES					
LPTR	NON PRESSURIZED H ₂ O			DESIGNED ON BAROMETRIC PRESSURE DIFFERENCES			2	2" H ₂ O		
NASA	NON PRESSURIZED H ₂ O				1.4	0.13	5			
AFNETR	NON PRESSURIZED H ₂ O						13		16	

(f) hydrostatic test

4.1 U. S. Reactor Containment Data (Contd.)

	OPERATION ASSUMED BEFORE INCIDENT	% RELEASE OF FISSION PRODUCTS TO CONTAINER	GAMMA SOURCE STRENGTH HISTORY (ASSUMED) FOR CONTAINED FISSION PRODUCTS, CURIES (t in seconds)	SPECIFIED (g) LEAK RATE	OBSERVED LEAK RATE	LEAK TESTS USED
EBWR			$1.74 \times 10^8 t^{-0.21}$	0.25	0.05	h, i
VBWR				1.00	<0.02	j
DRESDEN				0.5	0.016	h, j
APPR-1	9120 MWD					k (FOR WELDS)
PWR	33,750 MWD	10%		0.1	0.25	h, j, k
YANKEE	392 MW ∞	100%	$3.26 \times 10^9 t^{-0.21}$	0.2		h, k, i
INDIAN POINT				1.0		
GETR				2.0	<2.0	
ENRICO FERMI	300 MW ∞	100%	$3 \times 10^9 t^{-0.21}$	0.06		h, j
EBR-II	8100 MWD	50% of f.p. 50% of Pu 239	$5.4 \times 10^8 t^{-0.21}$	0.125		h, i
HRT	5 MW ∞	100%	$6.6 \times 10^6 t^{-0.2}$	0.2	<0.2	k
WEST MILTON				0.5	<0.1	h, i
MITR		20% GASEOUS OF WHICH 35% UNSTABLE	$0.5 \times 10^6 t^{-1}$ (t)	2.0	1.25	
PRTR	14,000 MWD	25%	$0.17 \times 10^8 t^{-0.22}$ (<10 ⁴ min)	0.27		
CP-5	18 MWD ∞	10% HALOGENS 100% NOBLE GASES	$5.5 \times 10^2 t^{-0.21}$			
LPTR		100%		3.0		
NASA	60 MW ∞	100%	$5.46 \times 10^8 t^{-0.21}$	0.3		
AFNETR				0.1	<0.1	

(g) % CONTAINED VOLUME PER 24 HOURS AT DESIGN PRESSURE

(h) SOAP SOLUTION

(i) DIRECT PRESSURE TEMPERATURE MEASUREMENT

(j) REFERENCE SYSTEM

(k) FREON OR HELIUM INJECTION

MWD = MEGAWATT DAYS

MW ∞ = MEGAWATT-INFINITE OPERATION

4.2 Untabulated Experience

Most of the work on containment problems has been done by individual reactor builders in connection with design, construction and testing of radioactive barriers for specific reactor systems. References for the information on these specific systems are listed by systems, following the tables.

The earliest work for what might be considered large power reactor systems was done in connection with the KAPL facility at West Milton, N. Y. The loadings were determined by considering a disruptive core explosion from nuclear energy release, followed by Na-water and air reactions leading to pressure rises and missile production. The attack on the problem and resulting large spherical shell design, construction and testing set the pattern for most future work in the U.S. The next facility was the EBWR at ANL, followed by the Vallecitos reactor of G.E. and the PWR at Shippingport, Pa. Analyses for these included possible metal-water reactions as a chemical energy source. The leak testing of the Vallecitos system first used the reference method. The shells for the Enrico Fermi Reactor near Detroit, and the EBR-II of ANL at NRTS, both representing fast reactors, have been built and tested. The containment structures for a number of reactors in the testing and research class, typified by AFNETR, MITR, LPTR, and GETR are now in service.

Most types of reactor outer containment systems are represented in Figure 4-1. The treatment in size and detail varies with each reactor, and a quantitative compilation of features has been presented in the tables at the beginning of Section 4.

The original containment vessel at West Milton was of the Type I. Three others have followed this course, Dresden, Indian Point and Yankee. Principal variation is with the Yankee Plant, which is supported entirely above ground. The others support the sphere as shown, with ground contact used for support of ballast, shielding, and the plant components. The sphere is a natural first choice for designers, being the most efficient structure. Depending on the ratio of cylindrical height to diameter, ϵ , it can be shown that the weight of steel required for a spherical vessel is as low as 68% of that required for a cylindrical vessel with hemispherical ends for equal volumes (and pressures):

$$\frac{W_s}{W_c} = \frac{\text{steel in spherical shell}}{\text{steel in cylindrical shell}} = \frac{1.5 \epsilon + 1}{\sqrt{5} \epsilon + 1}$$

ϵ	$\frac{W_s}{W_c}$
0.1	0.94
0.5	0.83
1.0	0.77
2.0	0.73
10.0	0.69

This advantage does not always make the sphere the best choice. Some fabrication charges may be higher. Also, since the free volume is normally determined by the expanding coolant, it is easier at times to standardize on a cylinder diameter and vary the length, especially in early design stages. The cylinder looks more attractive when considering the shape of the reactor, and the operating requirements such as rod motions and fuel handling. Furthermore, the space in a cylindrical vessel can often be used more easily for plant component arrangements than is possible with the sphere. The ratio of free volume to gross volume can be larger and free volume is the important requirement. These and other special considerations may have led to selection of the cylindrical form, Type II.

The EBWR, VBWR, EBR-II, Enrico Fermi, NASA-TR, PRTR, and AFNETR are all examples of systems for which the cylindrical vessel having hemispherical or hemiellipsoidal heads (Type II) has been selected.

Some special problems arise when the reactor housing is close to controlled plant boundaries. In such instances, a plain steel shell affords no protection from direct gamma radiation even though it may physically contain the fission products after an accidental release from the reactor. Most of the large power stations have a considerable distance between container and controlled boundaries. Then, attenuation by distance alone can protect the public. If tight boundary requirements must be met, the container can be modified to Type III.

This has been done in the case of APPR-1. Here a concrete thickness of two feet is used as lining between an inner cylindrical shell with a hemispherical end, a spherical zone end, and an outer cylindrical shell with hemispherical ends. In addition, an exterior shielding consisting of three feet of concrete against the outer containment shell has been placed and extends to the upper bend line.

The Nuclear Generating Station at Indian Point has an external cylindrical radiation shield around the containment shell. This shielding material is ordinary concrete. The construction consists of a circular wall six feet thick up to sixty-five feet above grade, a completely self-supported trussed roof about three feet thick, and a bottom layer following the contour of the spherical containment shell to a considerable extent and in turn supporting the sphere and internal equipment.

The Massachusetts Institute of Technology Reactor is located on a small site in an industrial area of Cambridge, Massachusetts, near the MIT campus. The minimum exclusion distance from the containment structure to the lot line is only 20 ft. Thus, in case of an incident where radioactive material fills the containment structure, shadow shielding is imperative. The shadow shield consists of a 2-foot thick concrete wall, 33 ft high completely around the cylindrical portion of the steel containment shell. This shield effectively attenuates the direct radiation dose following an incident but does not effectively reduce the scattered radiation dose.

The complex containment and shielding problem has been solved in another way in the case of the PWR, noted as Type IV in Fig. 4-1. Here a partial underground installation coupled with a division of free space requirements into several segments has been used. Shielding surrounds the containment vessels to protect after accidents in which radioactive materials are released from the reactor to the confined area. The shielding is provided by free standing enclosures located below grade with only the top portion projecting above grade. The minimum distance between the containment shells and the inner concrete surfaces is one foot. These surrounding air spaces allowed meeting code requirements where the containment shells must receive an annual inspection.

HRT also utilizes supplemental shielding around the reinforced rectangular parallelepiped containment tank which functions as the lining of a pit. This reactor shield pit is located in the center high bay of the building formerly occupied by the Homogeneous Reactor Experiment. The top surface of the roof shield slabs when installed coincides with the floor surface.

The walls of the HRT tank are reinforced by vertical columns embedded in a concrete pad three feet thick and heavily reinforced with steel. The concrete pad consists of original concrete in the upper portion. The wall between the reactor pit and the control room is constructed of two separated parallel steel plates having the intervening region filled with a shielding material such as high density gravel and water.

The roof of the HRT consists of two tiers of removable slabs made of high density concrete and having an overall thickness of five feet. A completely welded steel sheet is sandwiched between these tiers and extends across the top to form a gas tight lid. A slot and key arrangement anchor the roof slabs to the girders and supporting columns.

The primary design load basis for all these power reactor systems except EBR-II, Enrico Fermi, and West Milton, is expansion of the coolant on release from the reactor system by accident. These latter three reactors have liquid metal cooling which would result in little flashing to vapor. However, the hot coolant is incompatible with the environment and metal-air or metal-concrete reactions lead to liberation of heat, providing the

pressure load. Enrico Fermi has considered addition of the shut-down heating of the fission products to the coolant reaction heat in determining the loading. Several of the water reactors have considered the further possibility of exothermic metal-water reactions and have increased the design loads for the outer container accordingly.

In the initial design stages of the water-cooled power reactor containment structure, it is customary to use a conservative approach by neglecting heat transfer to the internal structural components and circumventing enclosure for a postulated nuclear incident where the coolant is liberated. The air occupying the free volume of the containment structure is heated by the coolant as it expands; thus, causing the air pressure to increase. This increase in air pressure is normally considered in the initial phase; however, for a rapid estimate this effort has been omitted.

After the initial design of the reactor containment structure and internal components has been fairly well established, an analysis relating pressure and temperature to time can be made. Here heat transfer to the intermediate components is considered. Difficulties encountered in making this more exacting analysis are the discharge rate of the coolant, the exact distribution of the coolant in the air at any instant and the uncertainty coefficients of heat transfer.

The PWR, Yankee, and Dresden have given consideration to the heat transfer. The peak transient pressure falls below the conservative method as one would expect; however, for PWR the difference between peak transient pressure and the conservative pressure was not too significant. For PWR, the postulated coolant discharge rate and heat transfer rates, the structural arrangement and the enclosing concrete supplemental shielding played predominant parts in having the computed transient peak pressure valve approach the conservative value.

To reduce the pressure of the mixtures of vapor and gas, many of the power reactor systems contain a built-in heat sink. A sprinkler system is often used as a heat sink to reduce the pressure and temperature in the containment system following an incident. Sprinkler systems have been installed in EBWR and Dresden for this purpose. The EBWR water sprinkler system sprays all areas within the containment shell. The sprinkler system can be operated manually from the control room outside the containment shell and is intended to hold down or reduce pressure, temperature and contamination of the atmosphere in the containment shell. A 15,000 gallon storage tank is installed under the top dome of the containment shell insuring infallibility of water supply. The PRTR Plant has a manually operated water-fog spray system as a heat sink. The discharge rate of the system being 500 gpm.

It must be remembered that the decaying fission products release heat which will tend to increase the pressure and temperature. After the flow has subsided to a considerable extent, a thermodynamic heat transfer study is necessary to determine if the pressure will eventually rise above the initial design pressure value. The operational period of the sprinkler system is restricted by the available water supply and the combined hydrodynamic and vapor-gas pressure acting on the containment shell.

Missiles of various masses, shapes, and energies may be created during a reactor incident. These missiles may be formed not only from the core of the reactor but from any part of the system operating at a high pressure or speed of rotation. Protecting the containment shell from penetration by these missiles is very important, as any sizeable penetration in effect destroys all containment.

The usual procedure in providing missile protection for the containment shell is simply to design the system so that missiles cannot be formed. This philosophy is followed in the design of most reactor plants. For example, in PWR all equipment and piping handling high pressure primary coolant were designed for conservative stresses to avoid extensive installation of concrete structure for missile protection.

If, on the other hand, a possibility of missile formation exists, available structure such as biological shielding is used for the dual purpose of missile shielding. In APPR-I, for example, the auxiliary biological shielding lining the inside of the containment shell also provides missile shielding for the shell. Prevention of missiles is first attempted by anchoring the reactor, steam generator, and pressurizer to the concrete floor in the base of the containment vessel.

Some reactor systems have special missile shielding provided. In EBWR special shielding is necessary since the steam turbine is located inside the shell and mechanical disintegration of a turbine is always possible. The cylindrical portion of the EBWR steel containment shell below the main floor is lined with a reinforced concrete liner 2 feet thick. The bottom head is lined with two conical concrete sections which approximate the ellipsoidal shape of the head down to the basement floor. Above the main floor, for a height of approximately 26 feet, the shell has a 1 ft thick reinforced concrete liner. The ceiling slab, 1-ft thick concrete with 3/8 in. thick steel plate on the under side, completes the missile stopping envelope. Wherever openings occur in this concrete envelope, additional shadow walls of concrete or steel are provided.

In EBR-II the missile shielding consists of 1/2 in. of Flexcel on the inside of the shell, covering the cylindrical and hemispherical portions above the grade level. An additional 4 inches of concrete is used on the inside of the dome and 12 inches of concrete used on the inside of the shell from the operation floor level to the dome bend line.

Internal missile protection for PRTR will consist of a one foot thick concrete cylindrical wall on the inside of the containment vessel. The wall will reach approximately 33 ft above the reactor hall floor and also support the rails for the overhead crane. The top reactor shields and access hole plugs will be anchored to the reactor hall floor.

The reactors mentioned so far have been designed in general accord with Section VIII of the ASME Boiler and Pressure Vessel Code.⁽¹³⁾ There will be or has been 100% radiographic inspection of welds in all cases except EBWR and EBR-II, where about 10% of the welds have been so inspected. Note that the cost of X-ray inspection is over \$100/ft. All the power reactors have followed ASTM Spec. A-300 providing for heat treated steel to improve ductility at low temperatures. All have used steel of Specification SA-201 to obtain good weldability and improved ductility. Application of the code results in a nominal factor of safety of 4, which is probably excessive, and in at least one case, the APPR-1, this was not met. The provision of pressure relief valves is another code requirement which cannot be met. There are several others, for which permission to deviate has been received, as evidenced by the inquiries to the ASME Boiler Code Committee which have resulted in special case rulings. These are shown in the Section 3.3 on codes and criteria.

For reactors other than power reactors, the containment structure follows the pattern of Type V of Fig. 4-1. The reactors of the class using Type V are research and test reactors which may have accidents, but not of the kind leading to large pressure loads.

The Type V was chosen for the MITR, the LPTR, and the GETR. The design pressures in the case of these research reactors are small (only a few psig) because of the large free space requirements for experimental work and because there is little available stored energy in the coolant. Meteorological loads are an important consideration in these designs. The shells are well anchored to prevent lifting into the air when design internal pressure is applied. Note that these designs do not come within the ASME code so that American Petroleum Institute standards for low pressure storage vessels (API Standard 620-See Section 3.3) are used. A variation is found in the ANL research reactor, CP-5, where a concrete rather than a steel shell is used.

4.3 Reactor Containment Data Information Sources

General

Nuclear Reactor Plant Data, Volume One, Power Reactors,
(The American Society of Mechanical Engineers, New York,
1958).

Brittan, R. O., and J. C. Heap, "Reactor Containment,"
Proc. of the Second UN Intl. Conf. on the Peaceful Uses of
Atomic Energy, United Nations, Geneva(1958), Vol. 11, p. 66.

A. Kolflat, "Review of Present American Design of Contain-
ment Vessels for Nuclear Power Plants, (Sargent and Lundy,
Chicago, Illinois, June 1958).

Chave, C. T. and Balestracci, O. P., Paper No. 1879, Proc. of
the Second United Nations International Conference on the
Peaceful Uses of Atomic Energy - September 1958, (United
Nations, Geneva, to be published).

✓ EBWR

J. M. West, J. R. Dietrich, A. S. Jameson, G. A. Anderson,
J. M. Harrer, and H. F. Brush, "Hazard Summary Report on
the Experimental Boiling Water Reactor (EBWR)," ANL-5781.

Heineman, A., and L. Fromm, "Containment for the EBWR,"
Paper No. 1891, Proc. of the Second United Nations Interna-
tional Conference on the Peaceful Uses of Atomic Energy -
September 1958, (United Nations, Geneva, to be published).

✓ VBWR

E. Beckjord, D. Fischer, M. A. Mead, L. K. Holland, L.
Kornblith, Jr., S. Untermeyer, and L. Welsh, "Operation of
a High Performance Light Water Boiling Reactor," Paper
No. 1923, Proc. of the Second United Nations International
Conference on the Peaceful Uses of Atomic Energy -
September 1958, (United Nations, Geneva, to be published).

✓ DRESDEN

V.A. Elliott, R. D. Maxon, V. D. Nixon, and J. W. Merryman,
"The Dresden Nuclear Power Station," Paper No. 2372, Proc.
of the Second United Nations International Conference on the
Peaceful Uses of Atomic Energy - September 1958, (United
Nations, Geneva, to be published).

J. E. Love, S. Darrow, and B. H. Randolph, "Civil Engineering Aspects of the Dresden Nuclear Power Station," J. Power Div. Am. Soc. of Civ. Engrs., Paper 1600, (April, 1958).

L. P. Zick, J. T. Dunn, and J. B. Maker, "Spherical Containment Shell of the Dresden Station," J. Power Div. Am. Soc. of Div. Engrs., Paper 1601, April 1958.

INDIAN POINT

G. R. Milne, S. M. Stroller, and F. R. Ward, "The Consolidated Edison Company of New York Nuclear Electric Generating Station," Paper No. 1885, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - September 1958, (United Nations, Geneva, to be published).

GETR

R. K. Andersen, W. M. Cegelski, E. Janssen, M. J. Larocco, T. J. Poshos, W. A. Pitt, W. S. Prime, and B. Wolfe, "Design and Safeguards Considerations of the Thirty Megawatt General Electric Test Reactor," Paper No. 415, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - September 1958, (United Nations, Geneva, to be published).

ENRICO FERMI

C. Burg and J. C. Feldes, "Civil Engineering Aspects of the Fermi Atomic Power Station, J. Power Div., Proc. Am. Soc. Div. Engrs., Paper 1602, (April, 1958).

A. Amorosi and J. G. Gevick, "An Appraisal of the Enrico Fermi Research Reactor," Paper No. 2427, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - September 1958, (United Nations, Geneva, to be published).

EBR-II

H. Monson and M. Sluyter, "Containment of EBR-II," Paper No. 1892, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - September 1958, (United Nations, Geneva, to be published).

HRT

Oak Ridge National Laboratory, Members and Staff of the Homogeneous Reactor Project, "The Homogeneous Reactor Experiment No. 2, Paper No. 449, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - September 1958, (United Nations, Geneva, to be published).

S. E. Beall, "Containment Problems in Aqueous Homogeneous Reactor Systems," ORNL-2091, (August 1956).

MITR

T. J. Thompson, M. Benedict, T. Cantwell, and R. A. Axford, "Final Hazards Summary Report to the Advisory Committee on Reactor Safeguards on a Research Reactor for the Massachusetts Institute of Technology," MIT-5007, (January 1956).

T. J. Thompson, "MIT Research Reactor," Paper No. 417, Proc. of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy - September 1958, (United Nations, Geneva, to be published).

PRTR

N. G. Wittenbock, ed., "Plutonium Recycle Test Reactor Preliminary Safeguards Analysis," HW-48800 Rev., (June 5, 1958).

LPTR

R. L. Detterman, ed., "Safeguard Report - Livermore Pool Type Reactor," AECD-4258; FW-56-006 (1956).

NASA

T. M. Hallman and B. Lubarsky, eds., NASA Reactor Facility Hazards Summary, NASA - Memo 12-8-58E. (October 1956).

YANKEE

Yankee Atomic Electric Company, Boston, Massachusetts, "Preliminary Hazards Summary Report, Part B, License Application," YAEC-60, (April 1957).

APPR-1

Alco Products, Inc. Schenectady, New York, Hazards Summary Report for the Army Package Power Reactor, APAE-2,
July 27, 1955.

PWR

Western Electric Corp., Bettis Plant, Pittsburgh, PWR Hazards Summary Report WAPD-SC-541, September 1957.

Supplementary reports issued as WAPD-SC-542 through WAPD-SC-549; WAPD-PWR-970 through WAPD-PWR-974.