5 ACCIDENTS AND HAZARDS

500 GENERAL

It is generally recognized that pressurized water reactors exhibit a high degree of inherent stability due primarily to the large negative temperature coefficient of reactivity associated with the change in density of the coolant moderator with temperature, superimposed on the somewhat smaller negative coefficient of the fuel itself. In this reactor, the instrumentation and control system is designed to take full advantage of the inherent stability and, furthermore, the full power, 392 mw, operating condition is chosen in such a manner as to maximize this effect. At full power, the mixed mean temperature of the coolant leaving the reactor is 524 F and that leaving the hottest channel is 599 F. Since saturation temperature at 2,000 psia is 636 F, even the water leaving the hottest channel is about 37 deg below saturation temperature, with the result that there is no bulk boiling and only a negligible amount of nucleate boiling in a small portion of the core at the hottest channels. Temperature changes, coupled with the negative temperature coefficient of the reactor, act to limit smaller transients, while bulk boiling in the hot channels operates to reduce and limit reactivity in larger transients where the power increases slowly.

The Doppler coefficient, approximately 10^{-5} Δ k/k per deg F, serves to minimize short, fast transients. Even with this small Doppler coefficient, about 5 per cent in negative reactivity is theoretically available starting from the cold condition before the fuel temperature has risen to its melting point of 5,000 F. Because the temperature of some of the centrally located fuel rises to this value before such temperatures are reached in the balance of the core, the effect of the Doppler coefficient is less than 5 per cent. Approximately one-fifth of this amount, a 1 per cent change in reactivity, is available at full power to limit a fast rising transient. Under these conditions, the average temperature of the fuel rises from the normal value of 3,000 F to about 4,000 F.

Twenty-four mechanical control rods are provided for regulating the power level of the reactor, compensating for fission product buildup, counteracting the effects of fuel burnup, and for scramming the reactor either manually or automatically. The control rods are capable of shutting down the hot clean reactor to about 3 per cent subcritical and of shutting down the hot reactor to about 8 per cent subcritical after operation at power has continued long enough to reach equilibrium xenon and samarium concentration. Additional control to bring the reactor to cold shutdown is provided by the chemical shutdown system through which negative reactivity can be introduced at a rate of 0.6 per cent per minute. Very long-time transients, such as xenon oscillations, are easily taken care of by the control rods.

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A scram by control rods alone and with equilibrium poison present brings the reactor to approximately 8 per cent subcritical at operating temperatures. The process of bringing the scrammed reactor from operating temperature to the cold condition by introducing chemical neutron absorber into the primary system is aided by the buildup of transient xenon which provides increments of negative reactivity for a period of about 8 hr. Some of the control rods can be left in the out position during the cooling-down period and are thus available for safety at all times and at all temperatures.

Reactor accidents are of three general types; those associated with reactivity insertion, those associated with release of chemical energy in the core, and those associated with mechanical failures of the main coolant system.

Two types of reactivity accident are considered; a low power level accident which might occur at start-up, and a cold water accident which could result either from the addition of cold water to the hot reactor at power or from the addition of pure water to a reactor which is operating with highly borated water. Accidents of this type are minimized by slow operating valves, by having high neutron flux and short period automatic scrams available, and by interlocks arranged so that the difference in water temperatures across a closed stop valve must be 50 F or less before that valve can be opened.

The possibility of a chemical reaction between water and the various constituents of the reactor core has been considered. Experimental results to date indicate that there is no problem of this nature.

One type of mechanical accident is the failure of coolant pumps which will reduce the flow through the core and may cause harmful or dangerous heating of the core. For the present, a procedure is proposed of scramming the reactor, operating at or near full power, after the failure of three or more pumps.

Another mechanical accident is one which involves a rupture of the main coolant system and the loss of large quantities of water. Without proper functioning of the safety injection system, partial or complete meltdown of the core may follow with resultant release of gaseous and volatile fission products from the fuel. Other accidents, such as clad failures, may release fission products, but only in the case of a rupture of the main coolant system can these fission products escape into the vapor container. This accident, therefore, has been used as a basis for vapor container design.

The results of a partial or complete core meltdown, no matter how improbable, are analyzed to determine possible criticality of the fuel at the bottom of the reactor vessel. Finally, the subsequent buildup of pressure and radioactivity within the vapor container is calculated to establish conditions therein, following an accident of this type.

501 REACTIVITY ACCIDENTS

Start-up Accident

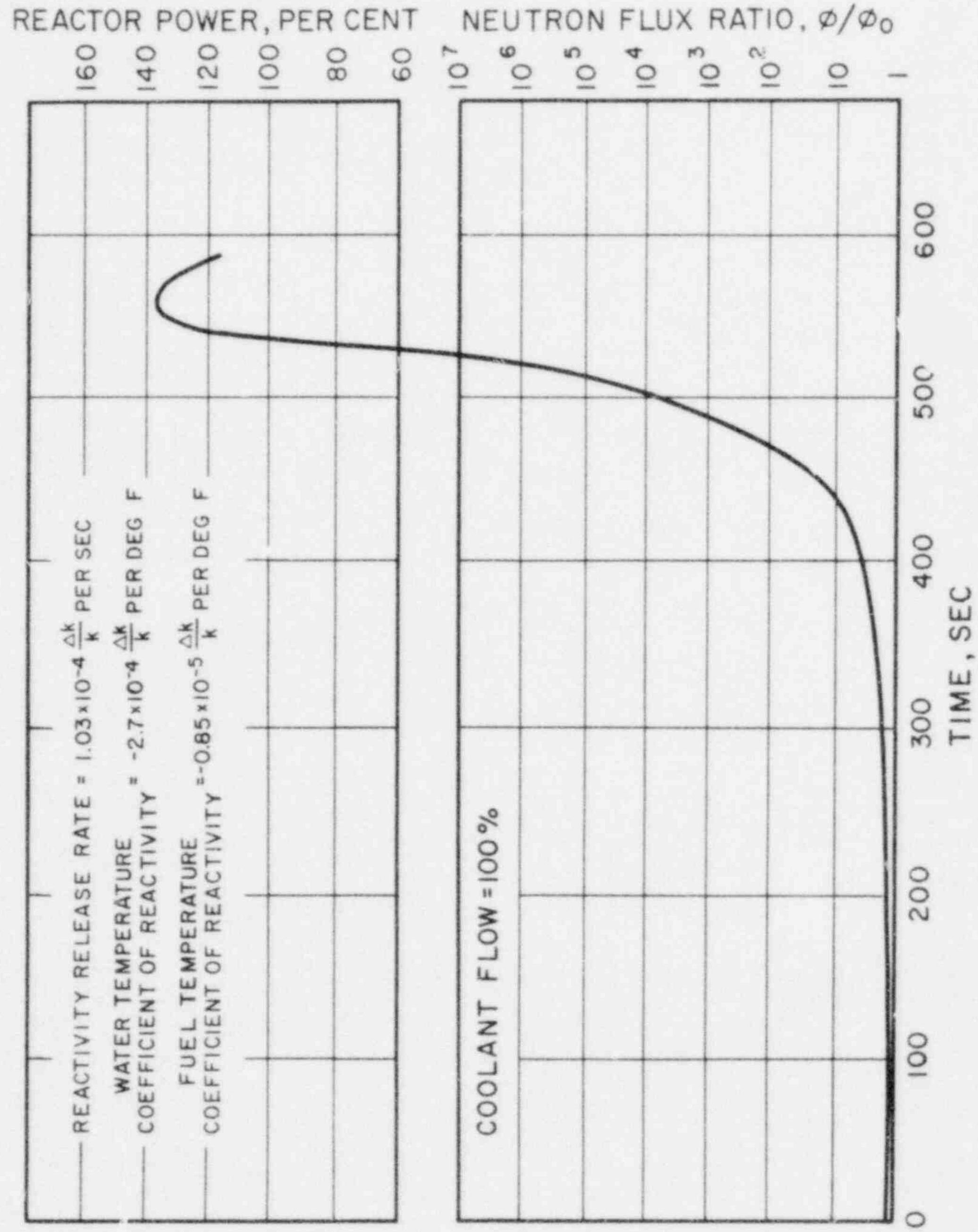
In one type of start-up accident, it is assumed that the control rods are withdrawn at the maximum design rate up to, and beyond, criticality. For such an accident to occur, there must be a series of multiple failures in the nuclear instrumentation and scram systems and, at the same time, there must be errors or negligence on the part of the reactor operators. Ordinarily, the reactor would be scrammed automatically on either a short reactor period or on high neutron flux level. In addition, both flux level and reactor period are displayed on instruments located on the operating console and the operator can take corrective measures.

Because of the importance and relative frequency of start-up operations, start-up accidents have been studied in considerable detail for all research and power reactors. In the case of the pressurized water type reactor, extensive analytical work has been done for this type of accident using analogue computers. The pattern of the accident is, therefore, well understood.

The large negative temperature coefficient in pressurized water reactors results in a large reactivity change from cold shutdown to operating temperature. In spite of this, the negative temperature coefficient of reactivity of a pressurized water-reactor is not effective in the start-up operation until significant power levels are reached.

Fig. 27 shows results of a typical start-up accident involving control rod withdrawal at maximum design rate. Neutron flux level, $\emptyset$, relative to flux level at the beginning of the rod withdrawal, $\emptyset_o$, is plotted on a logarithmic scale up to $\emptyset/\emptyset_o = 2 \times 10^7$. Above this value, power is plotted on a linear scale in per cent of the designed thermal output, 392 mw. The abscissa is time in seconds after initiation of control rod withdrawal. The reactor design limits the rate of reactivity addition by control rod withdrawal to 1.03×10^{-4} $\Delta k/k$ per sec. This rate of reactivity addition is assumed to continue through criticality until the negative temperature coefficient limits the initial power rise of the reactor. In the case analyzed, the maximum value reached in the initial transient is 135 per cent of full power, or 530 mw, and the time available for the operator to take corrective action is several minutes.

The reactivity increase rate assumed for the curve of Fig. 27 is the present design value. This value, 1.03×10^{-4} $\Delta k/k$ per sec, is well within safe limits since it corresponds to 74 sec to go from delayed critical to prompt critical. The maximum time to return to criticality from shutdown following a scram from full



START-UP ACCIDENT
WITH
CONTINUOUS ROD WITHDRAWAL

power would, under the most unfavorable circumstances, be less than 20 min. Both of these times are acceptable from the standpoint of plant operation.

Another start-up accident which involves step changes in reactivity produces results illustrated in Fig. 28. The reactor is assumed to be operating at a temperature and at a power level, 10 mw, which is high enough to make the temperature coefficient effective in limiting power. This figure indicates that, even with no control rod movement, step changes as great as 2 per cent can easily be handled without reaching even 30 per cent of design power rating.

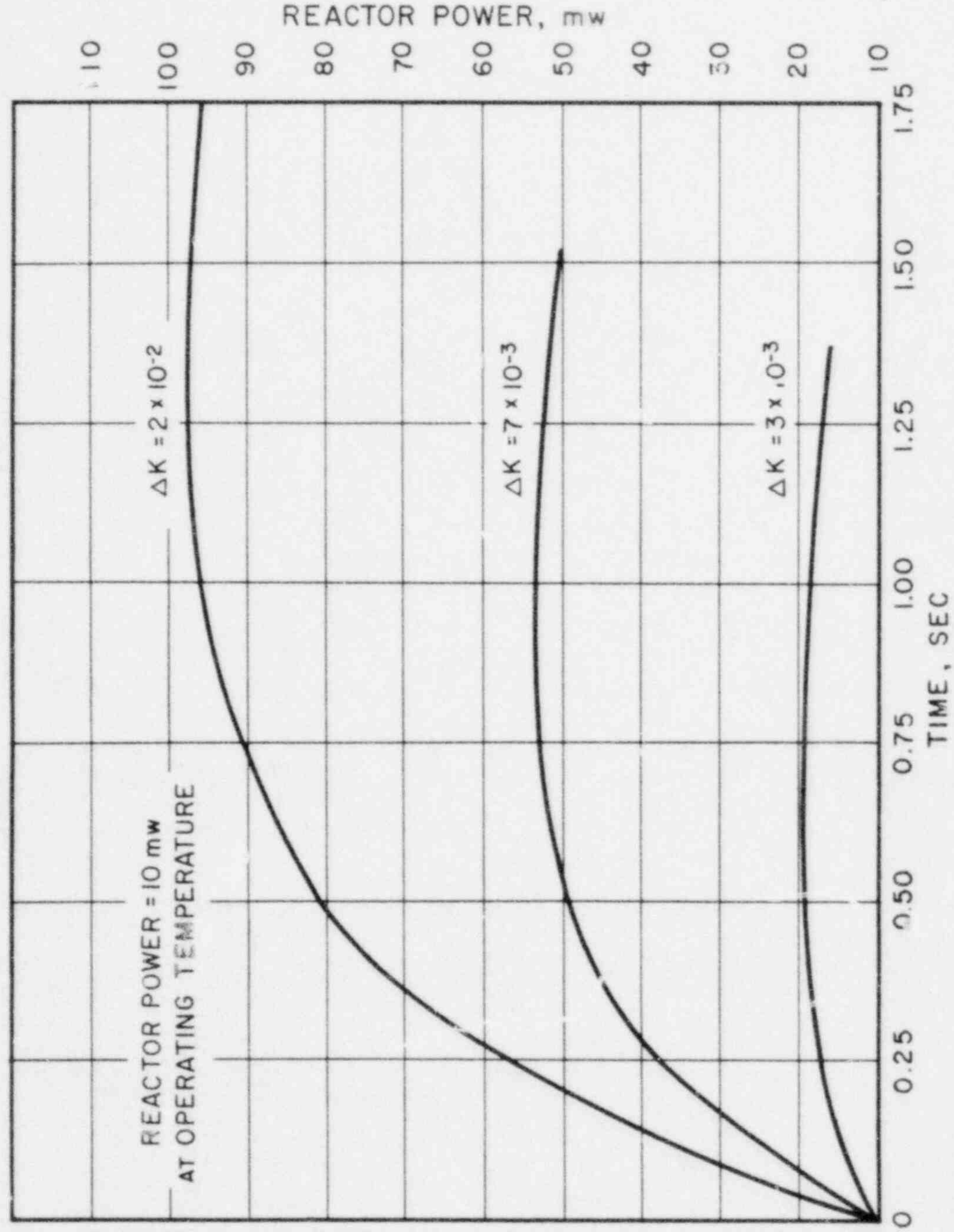
It is also significant that the power level stabilizes after approximately 1 sec and, thus, no corrective action is necessary. This leveling off in power is typical of the reactor for all significant partial power levels up to the temperature where bulk boiling occurs. From a practical standpoint, it is difficult to give an example of how 2 per cent in reactivity can be added instantaneously. Hence, the curves are presented without any explanation of how this might occur.

Cold Water and Boron Concentration Accident

Since the reactor has a relatively large negative temperature coefficient of reactivity, a lowering of temperature represents an addition of reactivity. Such a downward temperature change might come about through opening valves which previously had isolated a coolant loop containing water at a temperature below that of the water in the reactor core. A single isolated main coolant loop section contains approximately 400 cu ft of water, while the remainder of the primary system contains 2,600 cu ft.

The reactor may at some time be operated hot with small quantities of boron in the water which are nevertheless significant in terms of reactivity. If a cold water accident occurred under such conditions, it might be aggravated if the water in the blocked-off loop, in addition to being at a lower temperature, contained a concentration of boron lower than that of the water in the reactor. Also, an accident similar to a cold water accident could be initiated by opening stop valves when a difference in boron concentration exists between the water in a previously isolated loop and that in the reactor, even though both are at the same temperature.

A cold water accident is prevented by incorporating in the design a system of interlocks for the main loop stop valves. These interlocks prevent opening of the valves if there is a difference of more than 50 F in water temperature between the loop isolated by the valves and the reactor itself.



REACTOR POWER
VS
TIME AFTER STEP CHANGES
IN REACTIVITY

Analyses of cold water accidents based on present design concepts have been made. They indicate that, even with a failure of the protective interlocks, the rate of opening of the stop valves will be slow enough so that on opening, with the maximum possible temperature difference between the reactor and the blocked-off loop, an accident of any consequence can not occur. The reactivity effect is limited by the mixing of the small volume of water in an isolated loop with the larger volume of water remaining in the interconnected primary system. In the present design, the time required for the water to complete its travel through a loop is approximately 13 sec, whereas the time required to open stop valves to substantially full flow is greater by a factor of 4. Thus, the cooler water enters the core over a finite time interval during which adequate mixing takes place. Analysis shows that the possible rate of increase of reactivity is less than in the case of the continuous rod withdrawal accident during start-up which has been described. The course of the transient is similar, and the maximum power level reached following a continuous rod withdrawal will not be exceeded.

The possibility of an accident due to a difference in boron concentration is minimized by an operating procedure which calls for sampling and chemical analysis before cutting in a previously isolated loop. Conductivity cells and neutron meters are also possibilities for this purpose. Concentration in the isolated loop is matched to that in the rest of the system on the basis of this analysis before opening the stop valves.

If, through operating error, the concentration of boron in the loop to be cut in is lower than in the rest of the system, the excursion is limited by the slow opening stop valves. In the worst case, if the loop to be cut in contains no boron, there is sufficient reactivity in withdrawn control rods to shut down the reactor.

<u>Loss of Chemical Neutron Absorber</u>

Another type of reactivity accident is an unscheduled decrease in the concentration of the chemical neutron absorber. This could result from the introduction of pure water into the primary system to compensate for loss through a small leak, or by the removal of the dissolved neutron absorber by an ion exchanger. A change in reactivity of less than .00755 $\Delta k/k$, equivalent to going from delayed to prompt critical in 30 sec, presents no serious operational problems. To increase reactivity at this rate would require replacement of 10 per cent of the system volume per minute, about 2,500 gpm, with unborated water. This dilution rate applies to the cold reactor with an initial boron neutron absorber concentration of about 1.6 g per liter. At higher temperatures, the required boron concentration is lower and the dilution rate for the same reactivity change is greater. Since both the maximum pure water make-up rate and the flow through the ion exchanger are limited to approximately 100 gpm, an unscheduled cleanup of the neutron absorber in the primary system cannot cause a significant increase in reactivity.

Another accident is the loss of chemical neutron absorber caused by leakage from the main coolant system slightly greater than charging pump capacity at a reactor temperature and with a core condition that requires dissolved chemical neutron absorber for reactivity control. A leak greater than 100 gpm results in depressurization of the plant and will necessitate shutdown. This is accomplished by injecting water containing chemical neutron absorber from the safety injection system.

Another possible accident might be called a "boron hideout" accident in which a deposit of chemical absorber which has precipitated within the core is suddenly dislodged and swept out. This would cause an increase in reactivity equal to the amount tied up in the absorber. This is somewhat similar to the reactivity tied up in the voids of a boiling reactor. A deposit of absorber would be essentially black to thermal neutrons and would thus have the same reactivity effect as an equal surface area of control rod. A comparison calculation with control rod worths shows that only 0.5% $\Delta k/k$ effect would result from losing a deposit corresponding to a 4 sq ft neutron absorbing surface from the center of the reactor. A deposit of this magnitude that could be instantaneously released is felt to be impossible; and since it requires at least this much deposit to effect prompt criticality, it is concluded that no hazard exists.

Continuous Rod Withdrawal at Power

Another type of reactivity accident is continuous rod withdrawal at power. In this case, the reactor is initially operating at or near full power, and a continuous withdrawal of control rods at design speed occurs. It is conceivable, though highly improbable, that such an accident could occur through a combination of equipment and personnel failures.

If a continuous withdrawal of rods occurs, power level increases and reactor temperatures rise as a result of the reactivity addition. With the design reactivity addition rate of 1.03×10^{-4} $\Delta k/k$ per sec and minimum temperature coefficient of reactivity, with a chemical neutron absorber in the system, of -1.6×10^{4} $\Delta k/k$ per deg F, the temperature rises at the rate of 0.38 F per sec. At these slow rates, even if overtemperature control rod insertion devices and high neutron flux level scrams fail to function, the operator still has ample time to shut down the reactor before any damage results. The scram circuitry, including that of the manual scram, is independent of the circuitry which normally programs the rods and, hence, is not affected by failures of the rod programming system.

If the automatic controls fail and if, in addition, the reactor operator does not promptly initiate a manual scram, bulk boiling occurs in the reactor core, thereby compensating for further reactivity additions after the temperature of the water has exceeded saturation. With forced circulation, the boiling is expected to be steady up to 1 per cent reactivity in the voids.

The void volume corresponding to 1 per cent reactivity is approximately 3 per cent, and boiling occurs only in a relatively small portion of the core. With the reactor initially at full power, bulk boiling begins in approximately 100 sec. The condition of smooth boiling is expected to persist for 100 sec or up to approximately 200 sec after the continuous rod withdrawal is initiated. This allows sufficient time for the operator to halt rod withdrawal or take other corrective action. Even without such corrective action, it is believed that the bulk boiling effect will limit the transient and terminate the accident at safe reactor temperatures.

502 CHEMICAL ACCIDENT

Even in the event of core meltdown, no release of chemical energy by the reaction of water with the stainless steel cladding of the fuel is expected. While a reaction between components of stainless steel and water is thermodynamically possible, an energetic reaction is not to be expected on the basis of experience in industrial plants where molten stainless steel is mixed with water to produce a fine mesh powdered material.

In the production of fine mesh stainless steel powder by the water granulation method, molten stainless steel is poured in a stream about 1 1/2 in. in diameter. The stream of metal is hit with a high velocity water stream. A whole spectrum of particle sizes is produced in this way, varying from 400 mesh powder to globules 1 1/2 in. or so in diameter. The particles are so slightly oxidized that most of the powder is sold without further processing. No energetic chemical reaction between the steel and the water has been observed.

503 MECHANICAL ACCIDENTS

Loss of Coolant Flow Accident

If flow of coolant decreases due to an accident, it is of the utmost importance that thermal damage to the reactor core be prevented. However, in order to be certain that the plant design assures the integrity and continued serviceability of the core, various types of accidents involving a decrease in coolant flow have been analyzed. It is highly improbable that all forced flow will be lost since the four pump motors are divided, 2 and 2, between two sources of electrical supply; a transformer fed from the main generator leads and a transformer connected to a separate incoming 115 kv line. These are considered to be essentially independent sources.

In any such accident, decrease of flow with time is determined by the initial conditions, number of pumps failing, the inertia of the system, and the loop design. Fig. 29 shows the relationship of decreasing flow with respect to time for one, two, and four pumps. As flow decreases, coolant and fuel temperatures rise. The negative temperature coefficient acts to decrease reactivity, and the reactor power level drops.

The cases of one, two and four pump failures have been analyzed by means of an analogue computer, with the following assumptions:

Reactor initially at full power

Pump electrical supply instantaneously lost

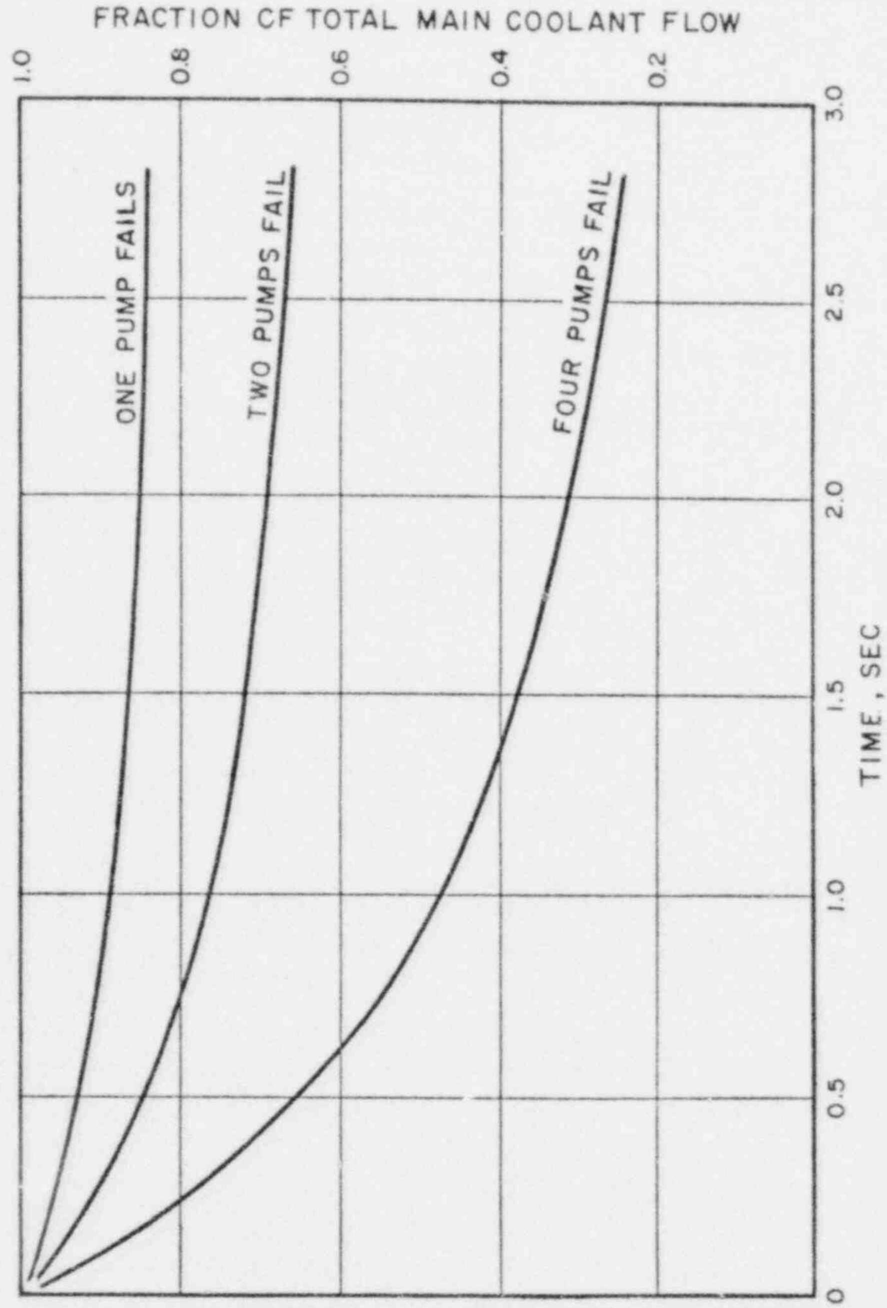
Reactor not scrammed

No steam voids present in the core

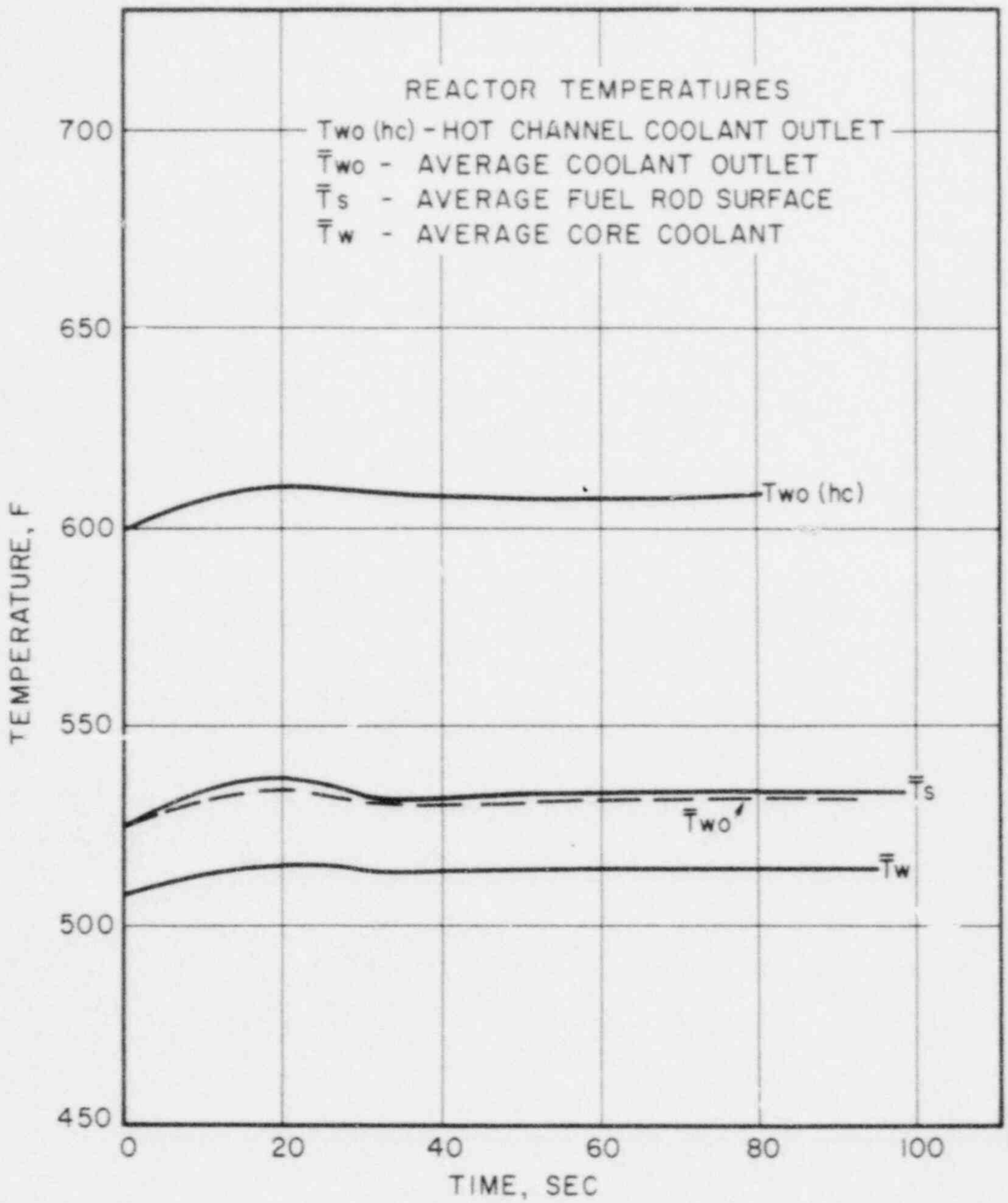
Reactor and steam generator divided into four sections (analogue representation)

Flow transit time through piping considered a first order lag

In this analysis, temperature-time relationships for average core coolant, coolant at the outlet of the hot channel, average coolant at the outlet of the core, and the average surface temperature of the fuel rod have been determined. Figs. 30, 31 and 32 show temperature-time relationships for the one, two and four pump cases, respectively. Excessive temperatures do not occur with the one and two pump failures. In the four pump case, however, bulk boiling begins at the outlet of the hottest channel in 9 sec.

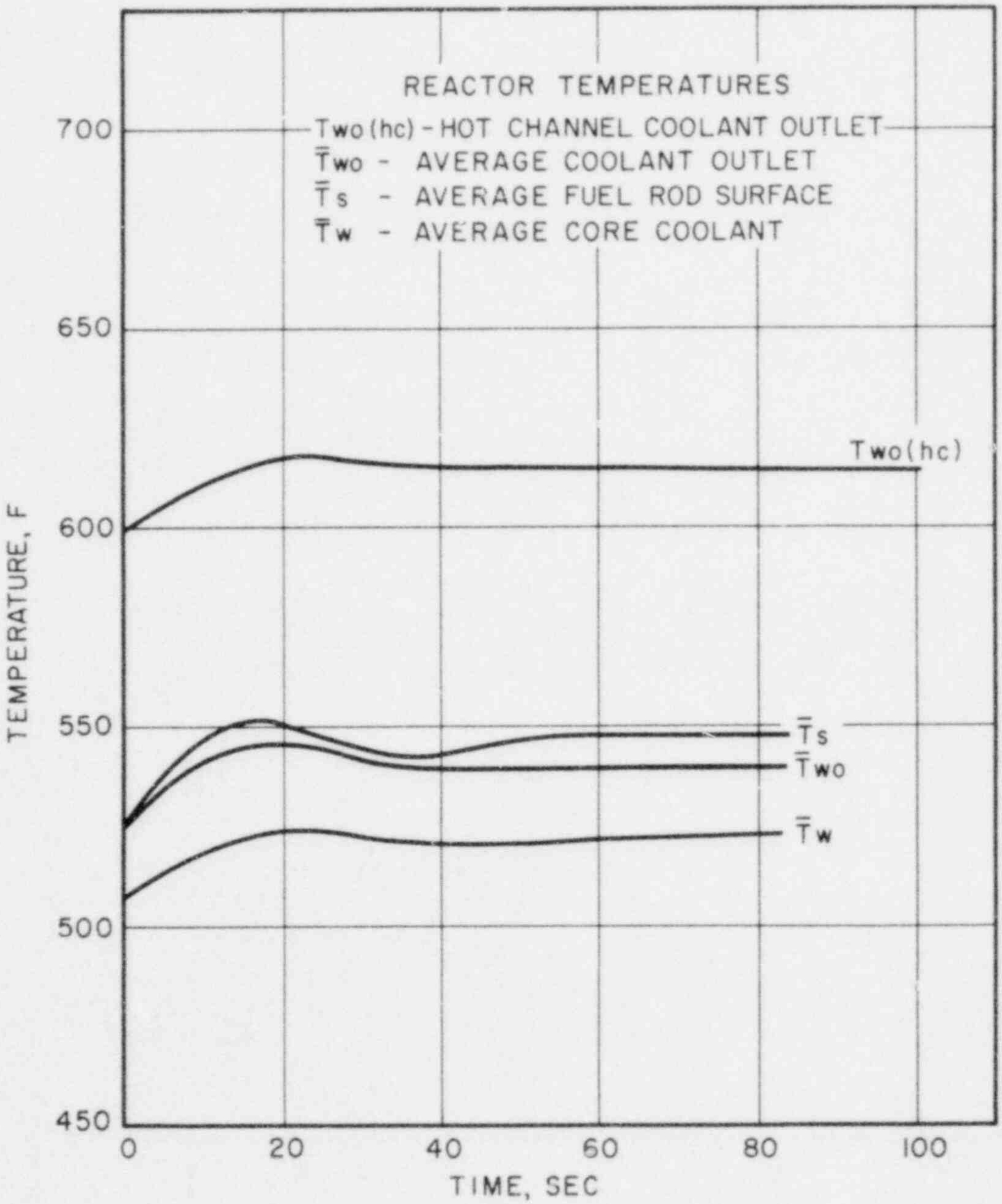


MAIN COOLANT FLOW VS TIME

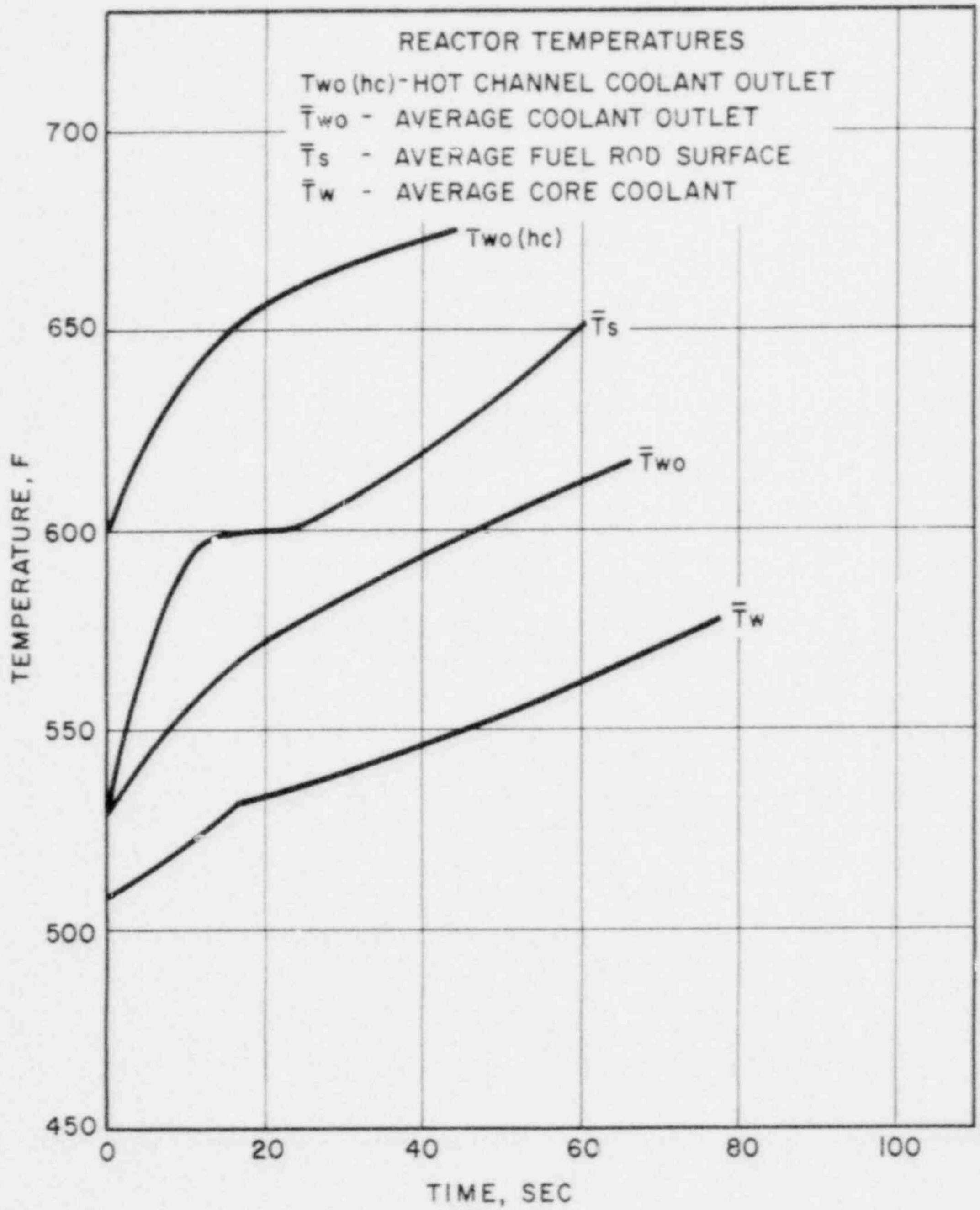


REACTOR TEMPERATURES VS TIME AFTER LOSS OF POWER TO ONE PUMP

FIG. 31



REACTOR TEMPERATURES vs TIME AFTER LOSS OF POWER TO TWO PUMPS



REACTOR TEMPERATURES VS TIME AFTER
LOSS OF POWER TO FOUR PUMPS

Since only the four pump failure without scram exceeds temperature limitations, a four pump failure with scram has been analyzed by means of an analog computer, with the following additional assumptions.

All control rods 60 per cent withdrawn at the beginning of the transient

The control rods fall 5 ft in 0.58 sec

The reactivity decrease due to scram is .05 per cent $\Delta k/k$

Equilibrium decay heat present

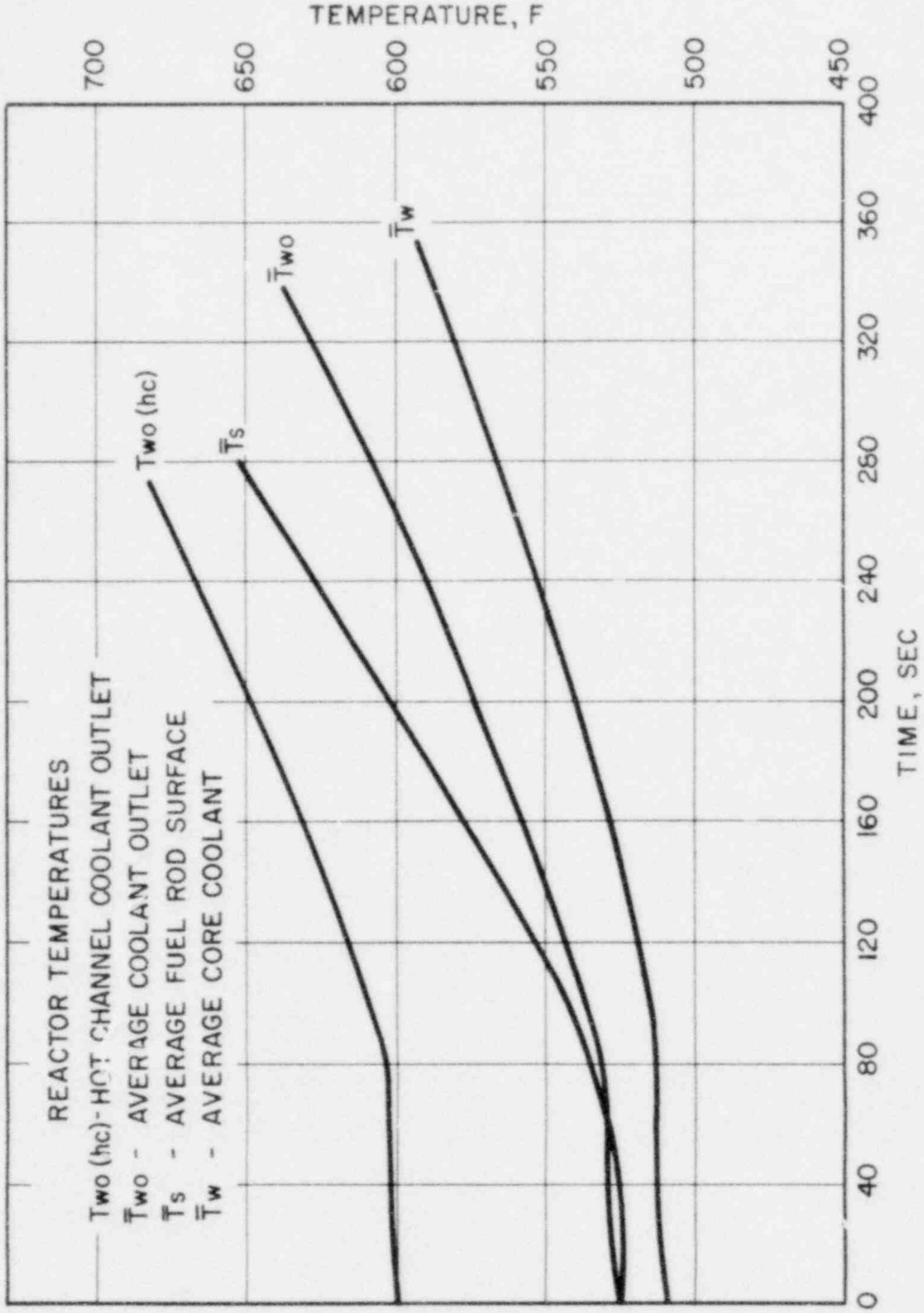
The temperature-time relationships are similar to those given for the previous cases and are shown in Fig. 33. There is no serious effect in this accident within the first few seconds. Assuming no heat transport to the steam generators as the pumps coast down, boiling occurs at the outlet of the hottest channel in approximately 180 sec. Thereafter, temperatures rise slowly as decay heat continues to be generated, until heat transport conditions are established. Heat transport by thermo-siphon circulation through the main coolant loops to the shell side of the steam generators is the subject of a study now in progress. It can be shown, however, on an overall conservation of energy basis, that it takes approximately 4.3 hr for decay heat to evaporate all of the water on the shell side of the steam generators and approximately 7.1 hr to evaporate all of the water in the steam generators plus all the water in the main coolant system down to the level where the core would be partially uncovered. During the first 4.3 hr after the loss of all the main coolant pumps, the evaporated water is discharged as steam to the atmosphere through the steam generator safety relief valves and the plant stack. During the period from 4.3 hr to 7.1 hr, steam escapes from the safety relief valves in the pressure control and relief system and is discharged to the low pressure surge tank. Initially, this steam is quenched by the cool water in the low pressure surge tank. Eventually, this water is heated to saturation pressure and temperature and steam is discharged through the 150 psi gage safety relief valves on the low pressure surge tank into the vapor container. Since there are three essentially independent sources of station service power and two of them are not affected by reactor scram, a total interruption of power to all four pumps is highly improbable. If such an interruption did occur, however, partial service, sufficient to start at least one pump, could be restored in a matter of minutes.

Loss of Water Accidents

General

The effects of loss of water accidents without any insertion of borated water from the safety injection system, but including release of contaminated vapor from the flashing of fluid in the primary coolant system, are considered from the following points of view:

Core again becoming critical
Core melting down when uncovered
Resultant pressure rise in the vapor container



REACTOR TEMPERATURES vs TIME
AFTER LOSS OF POWER TO FOUR PUMPS
WITH SIMULTANEOUS SCRAM

In any case involving loss of primary system pressure, automatic scram is effected by the control system. To investigate the possibility of a return to criticality, a series of breaks of increasing size is assumed, a small break equivalent to a 1/4 in. diam opening, a medium break equivalent to a 1/4 in. to a 4 in. diam opening, and a large break equivalent to the complete severance of a 20 in. OD main coolant pipe.

With the condition of the core melting down when uncovered, the largest break results in the most rapid uncovering of the core and the possibility of meltdown.

The maximum pressure in the vapor container results from a large break which releases substantially all the main coolant fluid and allows it to flash almost instantaneously into the vapor container. Pressure builds up within the vapor container before conduction through the sphere, absorption of the heat by interior concrete and other mechanisms for dissipating heat become effective. In order to select for design purposes a size of break that has physical reality, it is assumed that the large break is a complete rupture of a main coolant pipe, plus the simultaneous rupture of one secondary steam line. For this purpose, it is immaterial whether a pipe line or a vessel of the main coolant system ruptures, as long as the break is large and the loss of coolant is rapid and complete in a few seconds.

Small Break

The reactor is equipped with a high pressure charging system having a capacity of 100 gpm. The system has three pumps, one of which is always in service. If there is a leak so small that the loss of fluid is less than, or equal to, the capacity of the charging system, there is no loss of system pressure. For example, it has been calculated that a 1/4 in. diam opening will discharge approximately 25 gpm at 2,000 psia. Accordingly, a leak of this size, or smaller, anywhere in the primary system does not affect reactor operation.

Medium Size Break

If a break larger than 1/4 in. diam opening occurs, a single charging pump can not maintain system pressure, and the primary system pressure will drop.

A medium size break is defined as one equivalent in size to an opening between 1/4 in. and 4 in. in diameter. The 4 in. diam corresponds to an opening area of approximately 1/10 sq ft. Entrainment of water in the steam escaping from such an opening is not an important factor in removing water from the system. A break of these proportions will expel water in three more or less distinct stages:

Solid discharge of subcooled water caused by the pressure

Flash-flow of steam entraining some water

Flow of steam only, once the level of water in the reactor is below the outlet and inlet nozzles

Flow calculations for a medium size break show that the time required to complete solid water discharge is 30 sec with 67 per cent of the weight of water remaining at the end of this time. Two-phase flow is completed in the next 52 sec, with 33 per cent of the original weight of water remaining. Steam production from flashing of water and from decay heat causes the top of the core to uncover in approximately 30 sec more. The core is completely uncovered in an additional period of 110 to 600 sec.

Because so little water is entrained during a blowdown through a 1/10 sq ft or smaller opening, the temperature of the water drops in an orderly fashion, thereby increasing reactivity because of the negative temperature coefficient.

Large Break

A large break is defined as one ranging in size from an opening 4 in. in diameter to a 20 in. pipe severance, 16 in. ID, with two open ends. The corresponding areas are 1/10 sq ft and 3 sq ft.

For a 1 sq ft break, in the middle range of large breaks, pressure blowdown requires 3 sec and will eject one-third of the water from the vessel. The water level drops to the outlet nozzles of the reactor in 5 more seconds. The flashing mixture entrains such large quantities of water that the core is uncovered in 17 more seconds, or 25 sec after rupture.

With a full 20 in. pipe severance, sufficient water is ejected to uncover the core in 12 sec and essentially no water remains after 18 sec.

Calculations show that complete scram of the reactor does occur despite the flow of water and steam upward through the reactor core. A pressure differential greater than 77 psi through the core is required to exceed the gravitational force on the control rods. This pressure drop can not be achieved even under this extreme condition.

Criticality of the Core During Blowdown

A return to criticality may occur during blowdown if there is insufficient control in the rods to hold a new core subcritical at temperatures reached during the transient without a chemical neutron absorber present. However, although this can occur theoretically, an event of this type is highly improbable.

Reactivity increases during a blowdown transient because of the decrease in temperature. Two factors present in this reactor, void production and uncovering of the core, tend to counterbalance the reactivity increase.

The multiplication factor k_{eff} as a function of temperature is shown in Figure 10. k_{eff} as a function of void volume is shown in Figure 9. k_{eff} as a function of height of water in the core is shown in Figure 34.

The change in these variables with respect to time has been calculated as a function of size of opening. Breaks smaller than 1/4 in. constitute no problem. In a 1/10 sq ft, medium-size break, the k_{eff} goes below unity owing to void production and reactor scram. The k_{eff} with a clean core drops to a value of less than 0.96 approximately 1 min after the rupture, as the temperature falls. After this, the reactor is held subcritical by voids and control rods as core uncovering proceeds. With large openings, 3 sq ft, the rapidity with which the water is blown out of the reactor causes the water level to drop abruptly. The void coefficient, approximately 0.03% $\Delta k/k$ per vol % steam, is the controlling factor, and there is no return to criticality.

Decay Heat

Following reactor scram, which should be completed within 2 sec after a drop of system pressure, decay heat will be given off at the rate indicated in Figure 35. As long as the core remains covered, the decay heat will be extracted by boiling water. The rate of heat loss to the boiling water is such that the core will be cooled and the temperature of the fuel will drop.

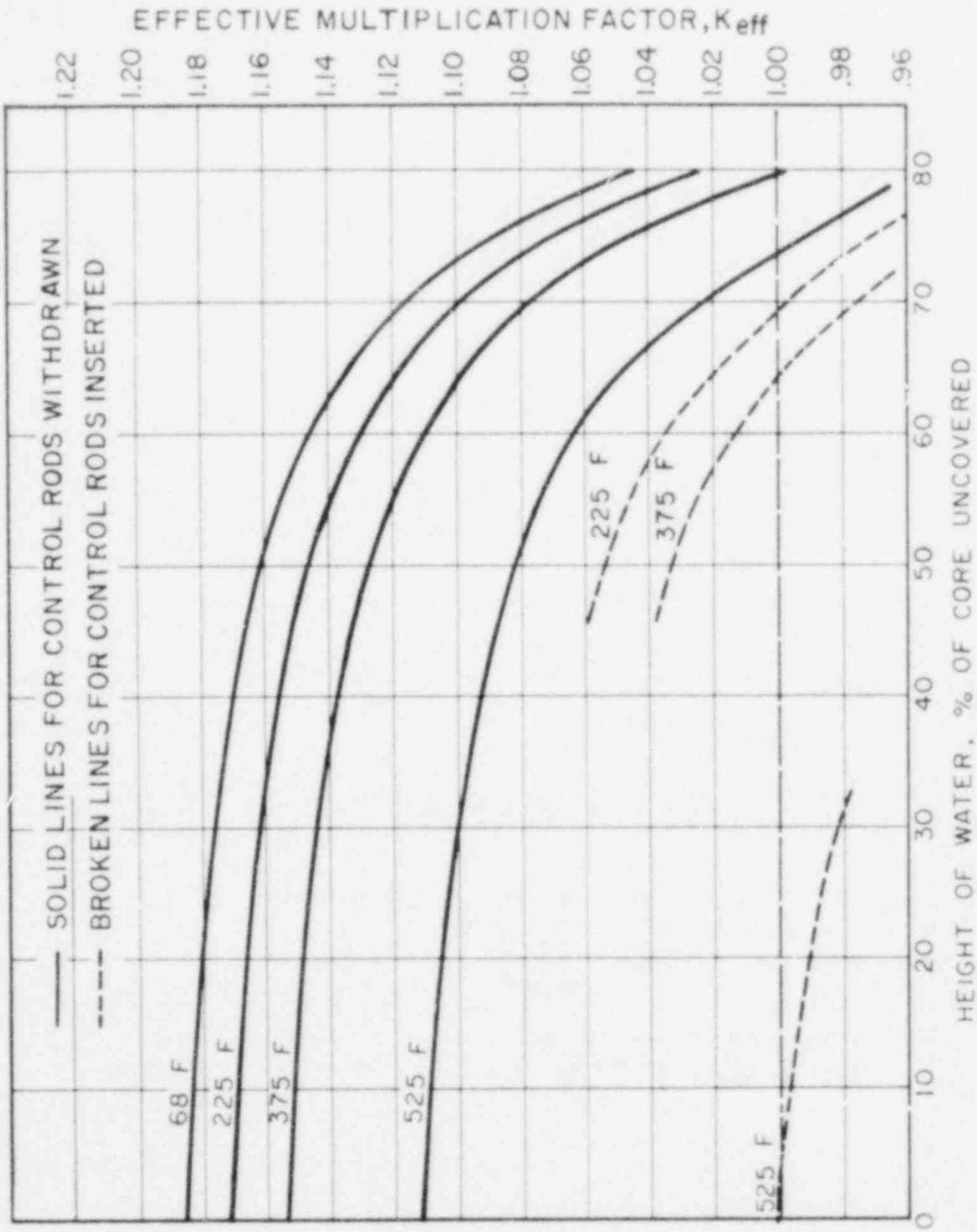
Boiling will take place in the core and fuel temperatures will decrease as the water approaches saturation temperature during the pressure discharge. As the water level falls in the core, the temperature of the fuel tends to rise but can be held within safe limits by use of the safety injection system.

Vapor Containment

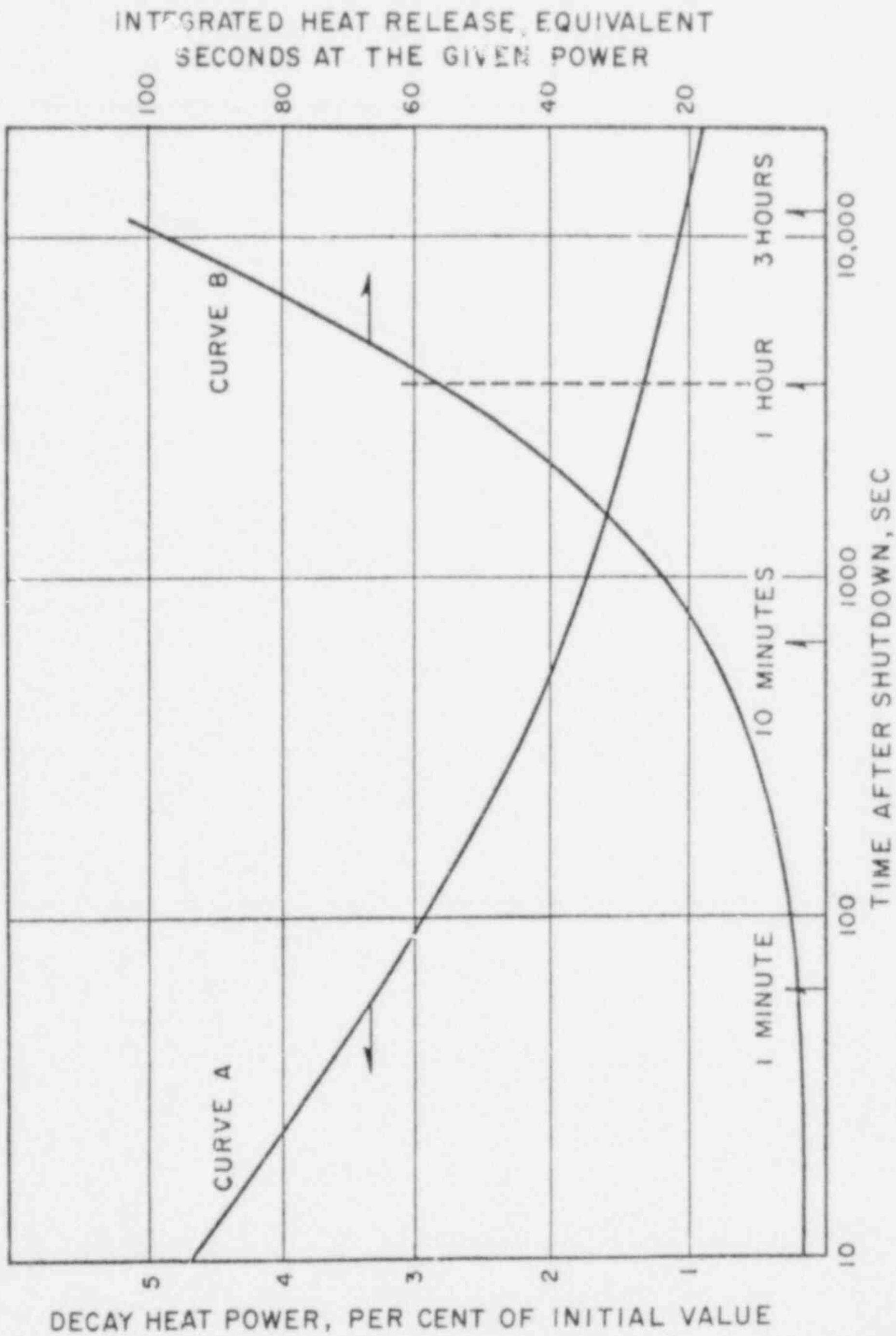
The vapor container is designed to retain all vapors, gases, liquids and solid materials released as a result of a loss of coolant accident. The maximum loss of coolant accident employed in the vapor container design consists of:

Complete severance of one 20 in. main coolant line, with two open pipe ends

Simultaneous rupture of one secondary main steam line inside the vapor container. The placement of each main coolant loop in a separate concrete shielded compartment and the installation of a nonreturn valve in the main steam line from each steam generator limit this part of the accident to the rupturing of a single secondary main steam line



REACTIVITY vs HEIGHT OF WATER



TOTAL DECAY HEAT GENERATION
(BETA AND GAMMA)
AFTER INFINITE TIME OF OPERATION
AT A GIVEN POWER

Detachment of an object or metal fragment from the pressurized system in such a way that it acquires kinetic energy, which, unless restrained or stopped by a barrier, might perforate the steel shell of the containment vessel, thus releasing contaminated vapor following the loss of water accident.

Figure 36 shows the initial pressure transient following the release of 186,000 lb of fluid from the main coolant system and one secondary coolant circuit into the net volume of the vapor container of 840,000 cu ft. The maximum differential pressure between the concrete compartment and the vapor container is 6 psi, and this pressure is reached in 0.2 sec. A port area of 400 sq ft in any one loop shield compartment is provided to limit the pressure differential across the concrete walls to this value. The concrete walls are designed for a maximum differential pressure of 8 psi. All coolant is released from the main coolant system within approximately 18 sec and equilibrium is attained inside the vapor container at a maximum pressure of 34.5 psi gage, or 49.2 psia. The corresponding vapor temperature is 249 F and the energy released is 94×10^6 Btu.

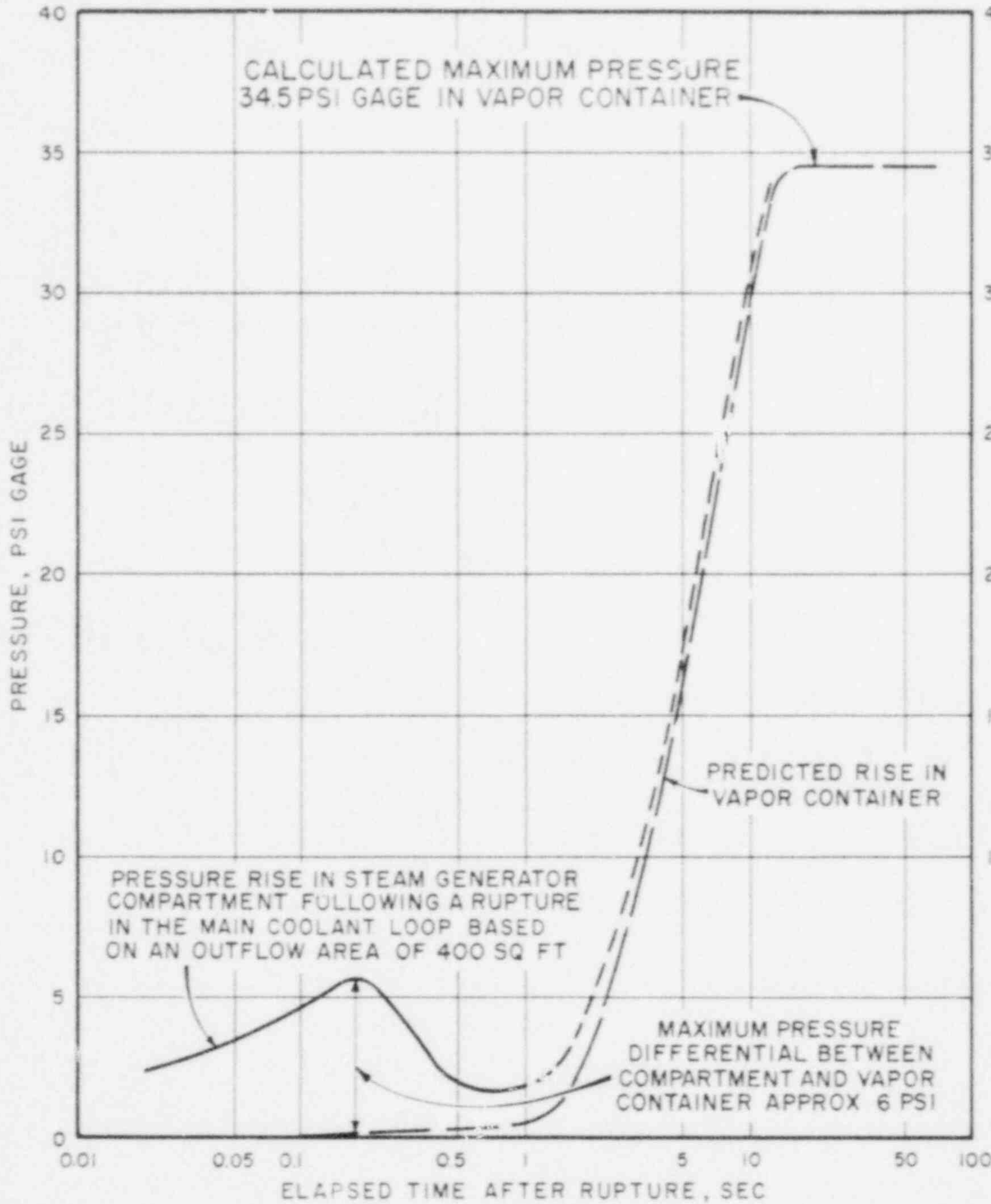
Figure 37 shows the long-time effect after the release of vapor and initial pressure rise to 34.5 psi gage. During the first 2 hr, there is a marked decrease in pressure due to thermal radiation and convection from the uninsulated vapor container shell and due to the diffusion of heat into the inner concrete structure. Subsequently, there is a gradual decrease in pressure with a small secondary rise, peaking in 4 hr at 15 psi gage, due to the continued release of decay heat from the reactor core.

The air-vapor mixture pressure within the vapor container after the maximum loss of coolant accident is based on the assumption that the total internal energy of the fluid remains the same before and after the rupture. This is based on the conservation of energy relation:

$$Q = AW + \Delta E$$

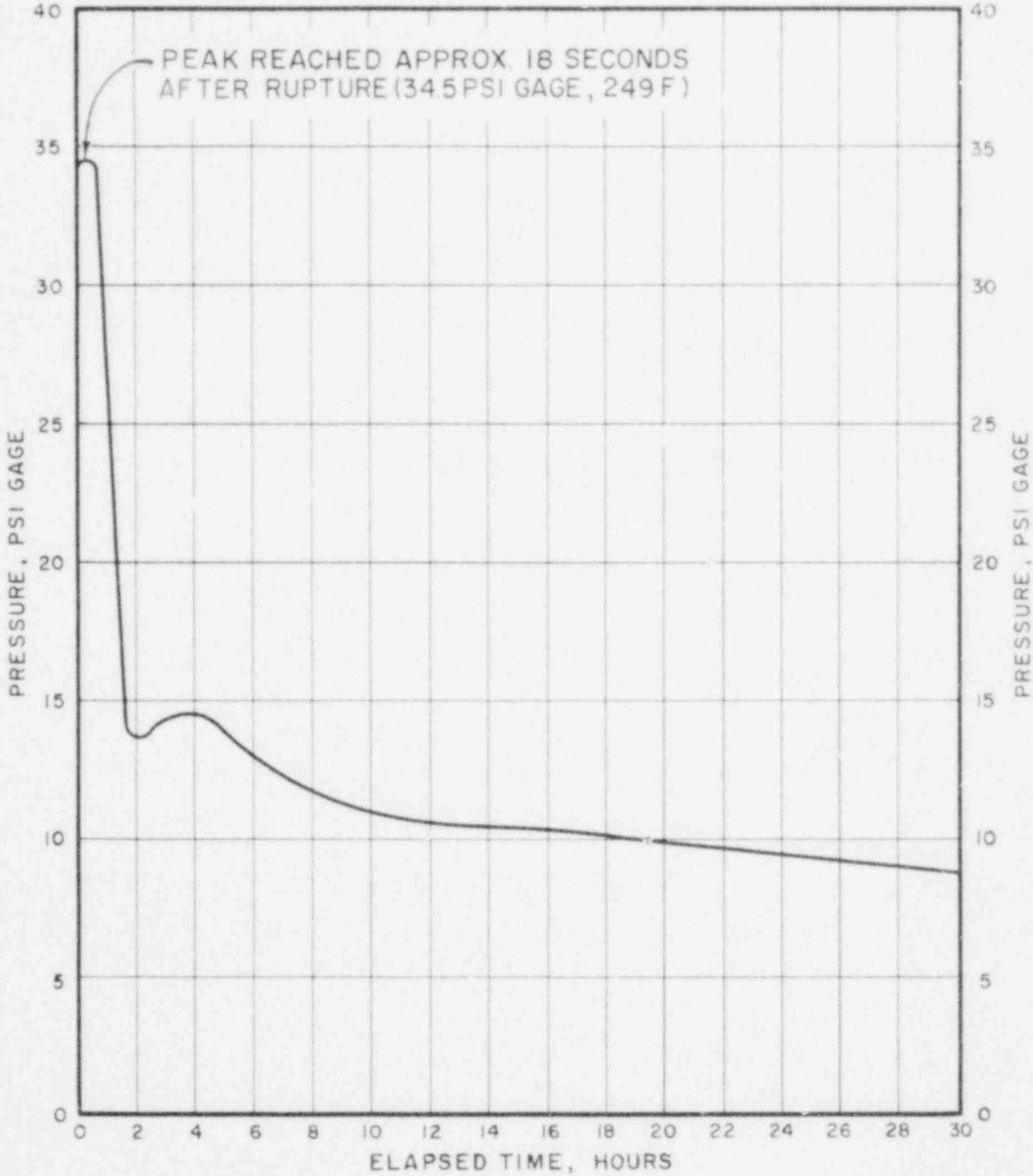
Where Q = Net heat release, Btu
A = Reciprocal of mechanical equivalent
W = Mechanical work performed, ft-lb
ΔE = Change of internal energy, Btu

During the brief interval after the initial burst, it is assumed that there is no heat loss, or $Q = 0$. There is no work done, since the fluid begins and ends in a state of rest, or $AW = 0$. Therefore, the internal energy before the accident is the same as that after the accident, or $\Delta E = 0$.



PRESSURE RISE IN VAPOR CONTAINER
VS
ELAPSED TIME AFTER RUPTURE

REV. 7-15-57



PRESSURE IN VAPOR CONTAINER
FOLLOWING A NUCLEAR ACCIDENT
NO INSULATION ON VAPOR CONTAINER SHELL

REV. 7-15-57

The summary of the principal data for the major loss of water accident is as follows:

Main coolant pressure, upper operating limit, psia	2,150
Average temperature main coolant, upper operating limit, F	518
Total volume of water in main coolant system, cu ft	
Reactor	1,600
Pressurizer	150
Steam generators	800
Piping	548
Pumps	20
Miscellaneous	52
Total	3,170
Total volume of steam in main coolant system, cu ft	110
Total volume of water in one secondary loop, cu ft	570
Total volume of steam in one secondary loop, cu ft	590
Gross volume of vapor container, cu ft	1,020,000
Net effective volume of vapor container, cu ft	840,000
Weight of fluids in main coolant system and one secondary circuit, lb	186,000
Internal energy of released fluids, Btu	94,000,000
Vapor flashed from main collant, per cent	32
Final pressure, psia	
Vapor	29.2
Air	20.0
Total	49.2
Total, psi gage	34.5
Final temperature, F	249

Subsequent to the initial pressure release, the following heat transfer effects proceed simultaneously, with the net integrated effect of these on the vapor container pressure shown in Figure 37. These effects are as follows:

Decay heat is released from the reactor core in accordance with the following relation:

$$\frac{P}{P_o} = .076 \quad \theta^{-.2}$$

Where P = rate of heat release after time θ, mw
P_o = initial rate of heat release, 482 mw
θ = time after reactor shutdown, sec

The rate of release of decay heat is dependent upon the number of hours which the core has operated; the longer the operating period, the greater the rate of release of decay heat.

This relationship is based on an infinite time of operation, which corresponds substantially to the rate of heat release after many hours of operation, and is thus conservative.

Heat is lost by radiation and convection through the spherical shell. This rate of heat release depends on the ambient temperature within the vapor container, the outside ambient temperature and a radiation and convection coefficient which available data indicate for large spheres is 2.2 Btu per sq ft, per hr, per degree F. The outside ambient temperature is taken as 70 F, the average of a summer day. The initial ambient temperature within the vapor container prior to the accident is 120 F.

Heat is absorbed by the vapor container metal. This rate of absorption is proportional to the ambient temperature within the vapor container. The weight of the containment vessel is approximately 2,500,000 lb and the specific heat is 0.12 Btu per lb, per degree F.

Heat is slowly released from metal parts which have been operating at normal temperature. This rate of heat release is proportional to the ambient temperature within the vapor container. The insulated metal parts weigh approximately 1,500,000 lb, and have a specific heat of 0.12 Btu per lb per degree F. Insulation is provided by 4 in. of Foamglas with an average thermal conductivity of 0.55 Btu per sq ft, per degree F, per hr, per in. of thickness.

Heat is absorbed by the internal concrete structures. The rate of diffusion of heat into the concrete with time is dependent on the ambient temperature within the vapor container. The temperature-time relationship for the concrete was determined by use of the Schmidt method, using a specific heat of 0.22 Btu per lb, per degree F, a thermal conductivity of 0.5 Btu per sq ft, per degree F, per hr, per ft of thickness, and a density of 150 lb per cu ft.

The ambient temperature is an independent variable in all of these factors, except decay heat, contributing to the redistribution of heat. This permits the determination of the new vapor temperature and total pressure of the air-vapor mixture after any elapsed time. Only the first two and last items in this list contribute importantly to the redistribution of heat. The results of the calculation are shown in Figure 37. The conservative assumption has been made that there is no condensation during the first few minutes after the initial rupture to reduce the calculated initial pressure.

Missile Protection

Although it is believed that no plausible missile could be released by the main coolant system, protection is provided by the inner concrete structure. It is considered that the ductile austenitic stainless materials of construction of the main coolant system piping will not fail in a brittle manner when in contact with the hot compressed coolant. It is likewise considered that the stainless clad, carbon steel reactor vessel, fabricated and tested according to the best techniques and under the proper codes and in contact with the hot compressed fluid, will not fail, and is not a feasible source of missiles.

The internal reinforced concrete structure serves as a secondary biological shield, as structural support for equipment and, in addition, provides a missile barrier. For biological radiation dose considerations, the walls and bottom of this structure consist of 4.5 to 6 ft of reinforced concrete, and the top or upper floor level consists of a minimum of 3 ft.

Loss of Load Accident

If flow of steam from the steam generators is accidentally stopped by malfunction and closing of the steam throttle valves or by turbine trip-out, it is important that thermal damage to the primary system be prevented. A study of this problem is currently in progress, however, preliminary information has been developed. Present design is predicated upon no control interlocking of turbine trip-out to effect automatic control and run-in (or reactor scram), and only audible and visible alarm signals are energized. However, if this design becomes unreasonable, interlocking will be provided. Preliminary figures indicate a steam dump bypass flow of 7 per cent of full steam flow as adequate for safe maximum temperatures in the primary and secondary systems utilizing only the negative temperature coefficient for shutdown.

504 CORE MELTDOWN

General

Partial or complete melting of the reactor core following rupture and the loss of all water from the main coolant system would result in release of fission products into the vapor container. To prevent such an occurrence a safety injection system with ample storage of borated water, a highly reliable means of introducing this borated water into the reactor vessel, and a dependable power supply is provided.

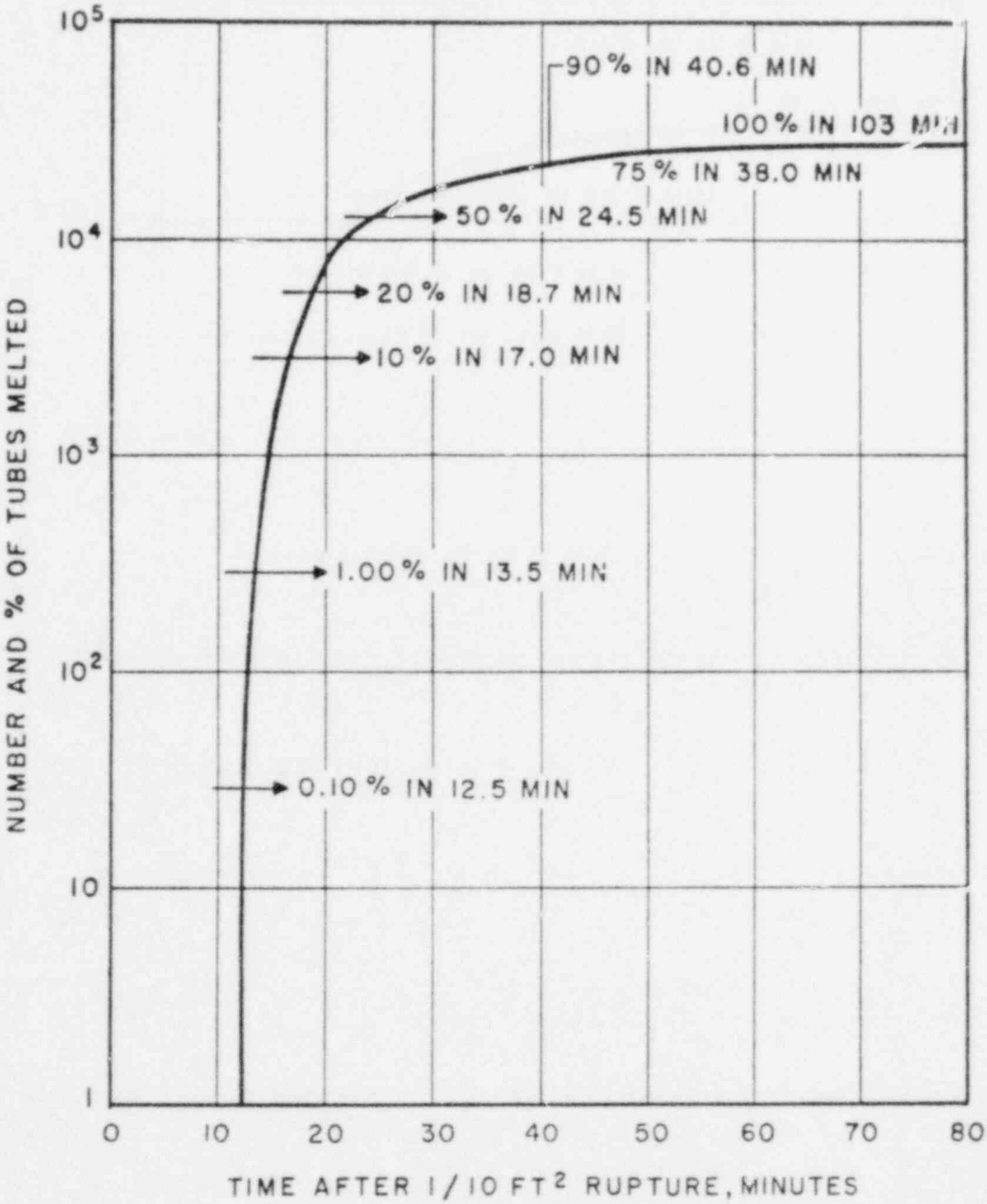
Whether melting of the core can be prevented may depend on the time available for the operator to introduce the borated water for cooling. The time interval between the initiation of the accident and the start of core melting is available to start injection of borated water into the reactor vessel. An analysis has been made to ascertain the length of this time interval. The core meltdown event has been carried to its conclusion, assuming no injection of borated water, to determine the rate of melting, fission product release and possible criticality of the melted fuel.

Mechanism of Core Meltdown

Under steady state full power conditions, 392 mw thermal, the average center fuel temperature is 1,362 F; the corresponding fuel cladding surface temperature is 570 F. The maximum fuel temperature at the center of the hottest pellet in the reactor is 4,500 F, while the corresponding cladding temperature is 642 F. The steep gradient in temperature between the center of the pellet and the fuel cladding surface is the result of the high heat flux which prevails and the low thermal conductivity of sintered UO_2. A significant decrease in the rate of heat removal from the surface of the fuel cladding will cause the temperature gradient to decrease and the cladding temperature to approach the fuel center temperature. The melting point of stainless steel is 2,800 F, while that of uranium dioxide is approximately 5,000 F. Therefore, stainless steel cladding will melt before the fuel melts if heat generation continues within the fuel while the rate of heat removal from the cladding surface decreases, as is the case if the fuel cladding tube is surrounded by steam. As soon as any point on a fuel rod reaches the melting point of 2,800 F, the cladding will rupture, allowing some of the gaseous and volatile fission products to escape. When a sufficient portion of the structural material in the fuel assembly has reached the melting point of stainless steel, the assembly fails and fuel pellets fall to the bottom of the reactor vessel.

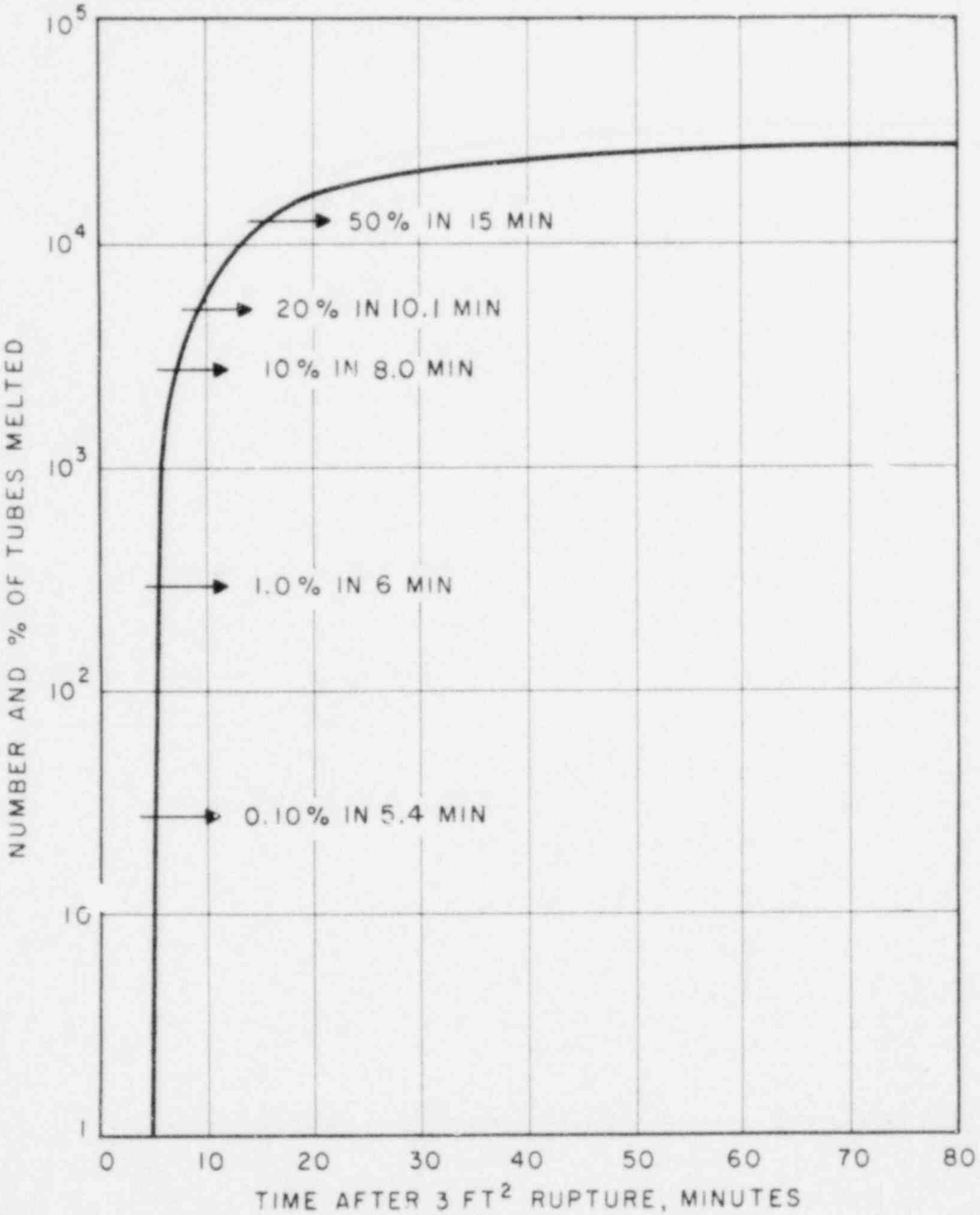
Rate of Core Melting

Calculations have been made to determine the rate at which the core melts and the data are presented in Figs. 38 and 39.



CORE MELTDOWN vs TIME AFTER RUPTURE

1/10 FT² RUPTURE



CORE MELTDOWN vs TIME AFTER RUPTURE

3 FT² RUPTURE

The fractions of the core melted for 1/10 sq ft and 3 sq ft breaks are shown as a function of time. The calculations are based on the steady state power distribution within the core from which the variation in decay heat generation rate follows.

Assuming no injection of borated water following a 1/10 sq ft rupture which uncovers the core, the temperature of the dry core rises in accordance with the decay heat, as shown in Fig. 35, and heat capacity of the system. The first tube melts in 12.5 min and many other tubes melt shortly thereafter. As shown in Fig. 38, half the fuel cladding tubes melt in 24.5 min. It is important to note that a time interval of 12.5 min is available before any melting occurs. This period of time is considered sufficient for the operator to take corrective action and inject borated water into the reactor vessel to cover a substantial portion of the core.

A 3 sq ft break without use of borated water injection follows the same pattern on a shortened time scale. Cooling occurs by convective boiling for 12 sec, after which there is no further heat removal. At the end of the cooling period, the average temperature of the pellets is approximately 510 F. The first tube starts to melt 5.4 min after the accident, as indicated in Fig. 39, and this time is available for the operator to take corrective action. The first pellets are released and fall to the bottom 7 min after the accident. Twelve minutes later, 40 per cent of the pellets are in the bottom of the vessel.

Criticality Consideration Following Melting of the Core

Calculations have been made to determine whether a criticality problem exists after the fuel pellets have fallen to the bottom of the reactor vessel. If all the fuel pellets are stacked in the bottom of the vessel, the maximum k_{eff} is 1.16 at 300 F, and the maximum k_{eff}, 1.04 at 380 F. It is assumed that the principle fission product poison, xenon, has been released from the fuel. The corresponding figures with equilibrium poison present are approximately 1.11 and 0.95, respectively. A period exists in the operational life of the core during which fission product buildup is sufficient to generate the heat required for melting the core but is insufficient for any significant reactivity reduction; thus, the figures of 1.16 at 300 F and 1.04 at 380 F are taken as being a possible condition. They are based on there being no stainless steel in the melted fuel assembly and on a water-to-equivalent uranium metal ratio of 0.82 corresponding to a random packing of the pellets. This low water-to-metal ratio is responsible for the strong negative temperature coefficient of the critical assembly of 0.0015 $\Delta k/k$ per deg F which may be computed from the reactivity figures given.

Criticality is attainable with water present if some quantity less than all the fuel drops to the bottom of the pressure vessel. At 300 F, however, at least 40 per cent of the fuel must fall to be critical; at 380 F, 62 per cent is required.

The timetable for fuel pellets to fall from the core into the bottom of the vessel is not the same as that for melting of the core. The fraction of pellets in the vessel bottom at a given time is less than the fraction of the core melted at that time. For example, following a 3 sq ft rupture, 40 per cent of the core will melt within 12.5 min but 40 per cent of the pellets will not fall to the bottom until a total time lapse of 19.5 min after the accident.

The criticality hazard at the bottom of the vessel depends on the following factors:

Presence of water

Temperature of fuel

Amount of fuel stacked in the bottom

Presence of equilibrium poisons in the fuel

Presence of stainless steel in the fuel at the bottom

Configuration of vessel bottom

If it is assumed that water is present to a level just below the core, a minimum of 40 per cent of the fuel pellets must be in the bottom of the vessel before $k_{eff} = 1.0$. This requires a minimum of 19.5 min after the 3 sq ft rupture or 35 min for the 1/10 sq ft rupture. If it is further assumed that all of the poisons escape from the fuel pellets and no stainless steel is present, the mass of pellets would still not go critical if a grid of neutron absorbing material were located at the bottom of the vessel. Without such a grid and in the unlikely event that all other factors mentioned occur in such a way as to produce maximum reactivity, a nuclear excursion with a total energy release of 312 mw-sec would occur. Furthermore, the power release would be distributed fairly uniformly over a time interval of 100 sec so that the rate of energy production would be only 3 mw, a rate smaller than the decay heat generation 3 hr after shutdown.

Fission Product Activity at the End of Core Cycle

At the end of the core cycle, a variety of fission products is present in the fuel material. An analysis of the gross gamma activities of the gaseous and volatile fission products has been made, since fission products of this type can be released if core melting is not prevented. It is assumed that the core has been operated at full power of 392 mw for an infinite length of time. This is a conservative assumption which yields somewhat higher activities than those which might actually be present.

The noble gases and the halogens are considered to constitute the gaseous and volatile fission products. Altogether, 12 isotopes of bromine, krypton, iodine and xenon are adjudged to possess significant gamma activity to be considered. Five elements are volatile at, or below, 450 C in combination with oxygen and/or one of the halogens. These elements are arsenic, molybdenum, antimony, tin, and tellurium. Of these elements, the isotopes of arsenic and tin have a low yield in the fission process. Fifteen isotopes of the other three elements have been examined and significant activities have been included in the totals. Gross fission product gamma activities are given in Table 7 at three different times after reactor shutdown from full power, 392 mw.

Table 7

Fission Product Gamma Activity Following Shutdown

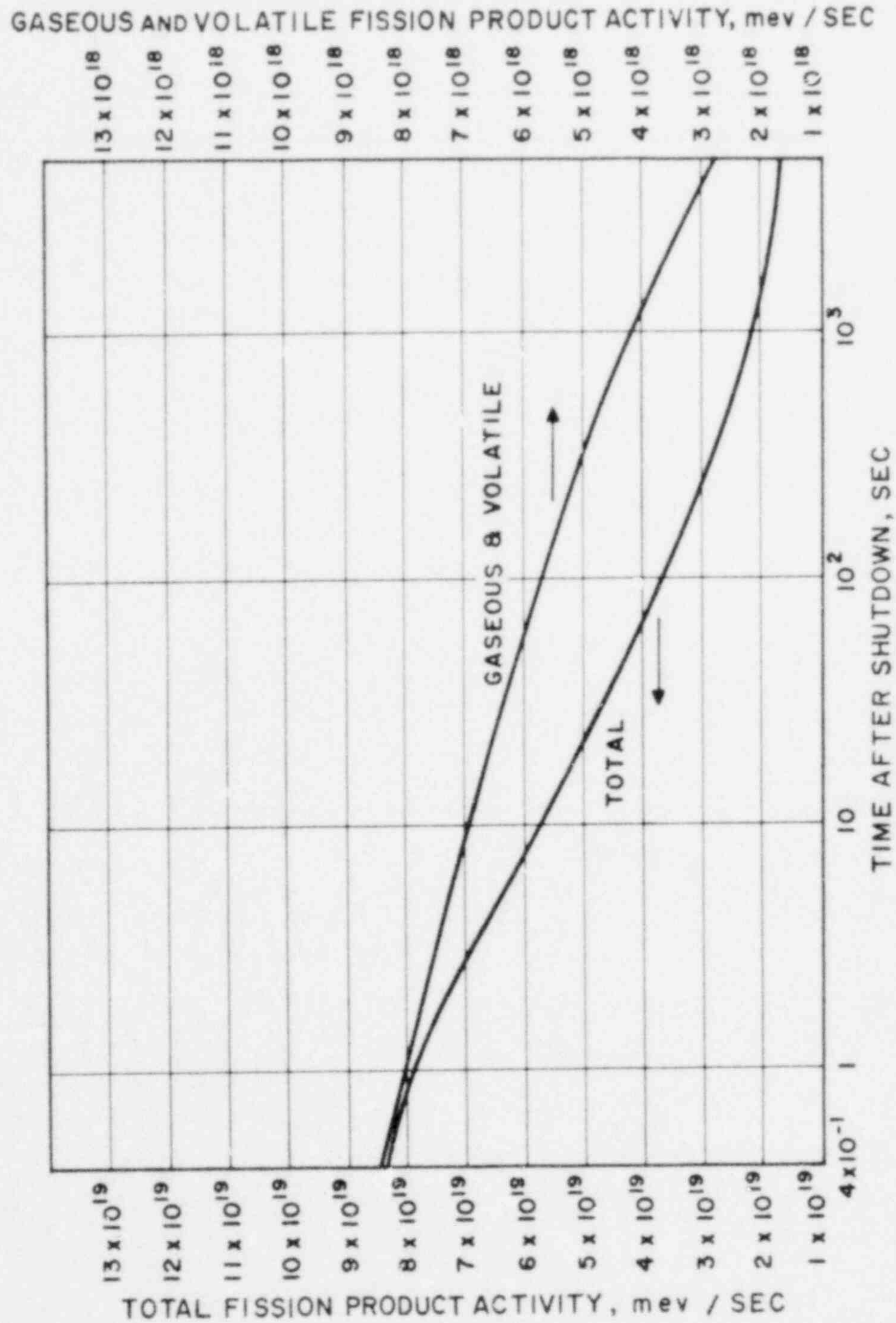
(In units of 10^{18} mev/sec)

Time after shutdown	0	5 min	1 hr
Gases	7.64	4.31	2.66
Elements volatile as compounds	.81	.73	.44
Total	8.45	5.04	3.10

Fission product gamma activity is plotted graphically as a function of time up to 1 hr after shutdown in Fig. 40. The lower curve scale of 10^{19} gives the gamma activity of all the fission products following shutdown; the upper curve scale of 10^{18} includes only the gaseous and volatile fission products and may be compared with the data presented in Table 7.

Fission Product Release to Vapor Container

The fission product gamma activity which is present in the core after infinite time of operation at 392 mw is shown in Fig. 40. Following an accident in which no use is made of the safety injection system and complete core melting results, the activity attributable to gaseous and volatile fission products could be released into the vapor container. If the safety injection system is used but is not fully effective, partial melting occurs and the activities released to the vapor container are the product of the values shown in Fig. 40 and the fraction of the core melted. For a 1/10 sq ft break in the main coolant system, the fraction of the core melted as a function of time is given in Fig. 38. Similar fractions for a large break of 3 sq ft are given in Fig. 39.



FISSION PRODUCT GAMMA ACTIVITY AFTER INFINITE TIME OF OPERATION AT 392 mw POWER

505 HAZARDS FROM REACTOR ACCIDENTS

Maximum Credible Accident

In foregoing sections analyses have been made of a number of accidents the most serious of which is the large loss-of-water accident. Such an accident might occur through rupture or severance of a 20 inch main coolant line resulting in depressurization and virtually complete loss of water from the primary system. The danger here, of course, is the possibility of core meltdown and release of fission products to the vapor container, thereby causing a radiation hazard to the public. To prevent such an occurrence a safety injection system which is described in Section 219 is provided. It is believed that this system is designed and can be administratively controlled to assure instant availability in case of need and cannot in any credible way be disabled or rendered inoperative by the primary effects of the accident. In the opinion of Yankee Atomic Electric Company and its technical advisors, the maximum credible accident can be defined as this large loss-of-water accident, with core melting prevented by the safety injection system, no release of fission products from the core, and therefore no hazard to the public.

Hypothetical Accident

The unique danger from a nuclear reactor installation is the accidental release of fission products from the plant and the creation thereby of external radiation hazards. The present state of reactor technology demands that all reasonable measures be taken to guard against even the most unlikely event by incorporating effective safety features in the plant design. Accordingly, even though the maximum credible accident does not result in the release of fission products and cause external radiation hazards, a hypothetical accident in which such a release does occur has been postulated and analyzed in order to evaluate the effectiveness of containment and other safety features which are incorporated in the plant design.

The hypothetical accident is based on the following assumed conditions and sequence of events:

A 20 inch pipe severance occurs in the primary system resulting in depressurization and virtually complete loss of water from the primary system.

Blowdown of the primary system results in an initial pressure rise in the vapor container to 34.5 psi gage, decreasing to approximately 15 psi gage after 2 hours.

For unexplained reasons, partial core meltdown occurs.

No criticality of the melted down fuel occurs.

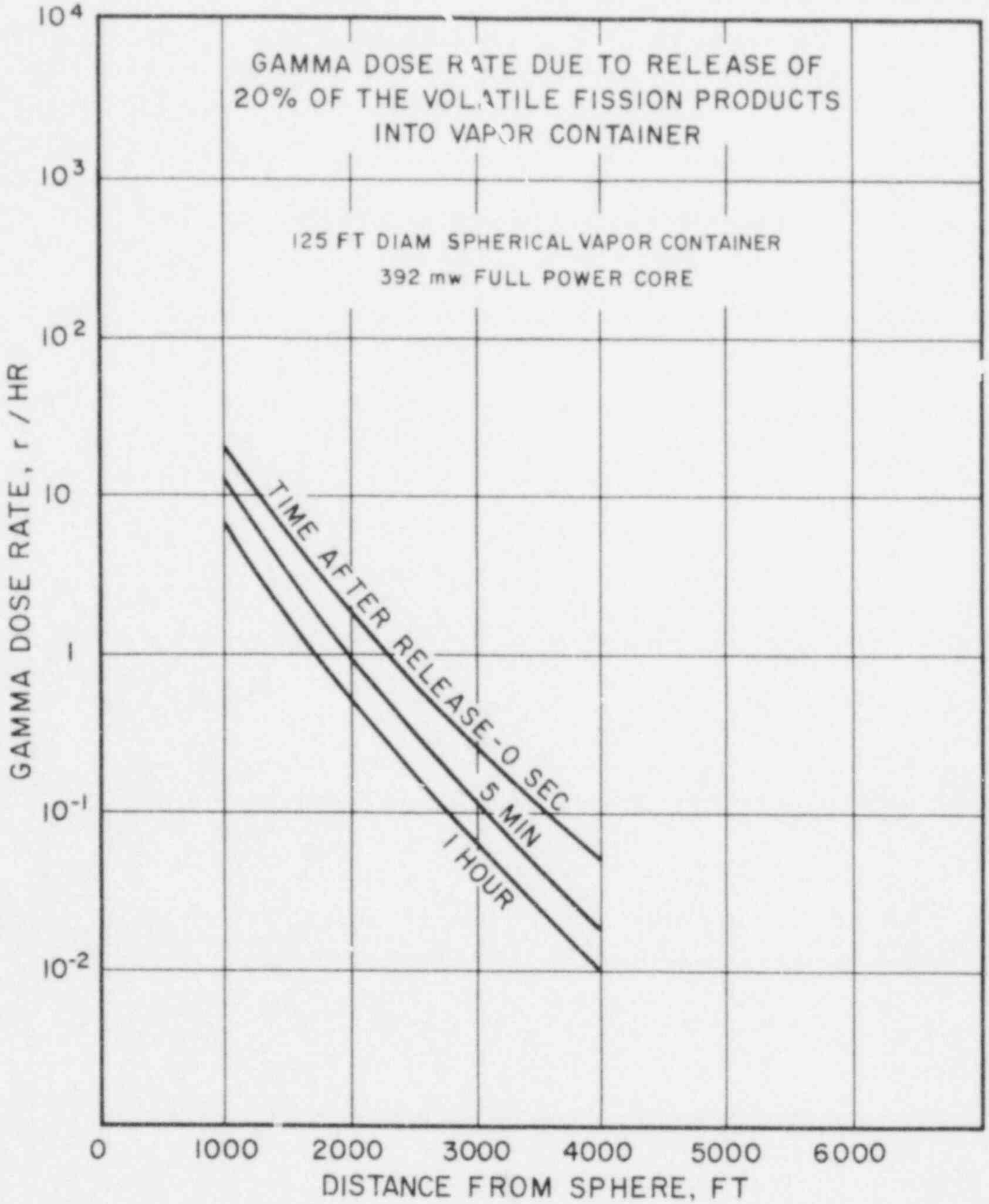
Twenty percent of the gaseous and volatile fission products present in the core after 10,000 hours of reactor operation at 392 mw are released to the vapor container and dispersed homogeneously therein.

Gaseous and volatile fission products are released instantaneously from the fuel into the vapor container, although the release actually would occur over a finite time interval which would begin several minutes after the start of the accident.

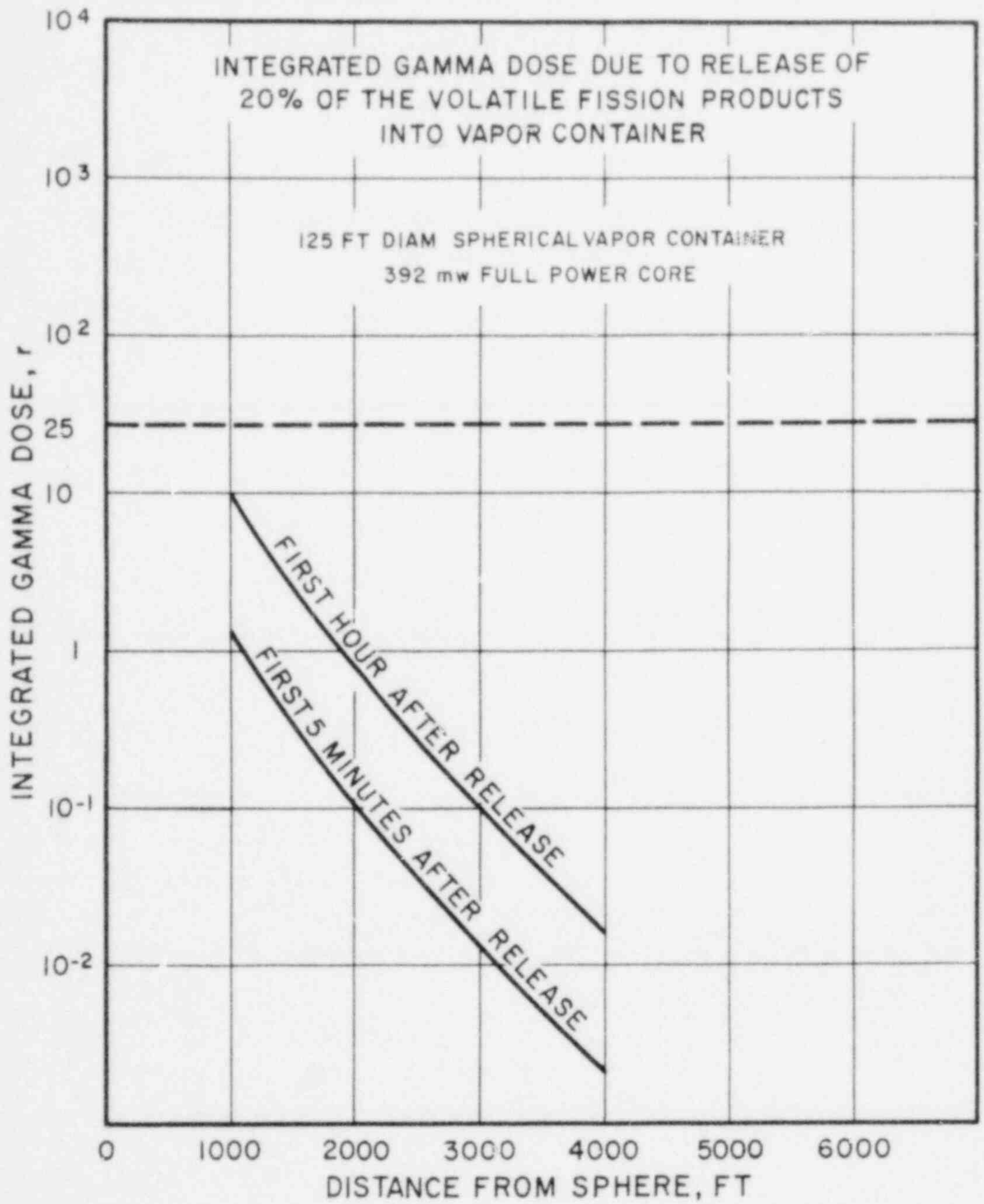
The vapor container has a leak rate of 70 cu ft per hr with the vapor container internal pressure at 15 psi above atmosphere, and this leak rate and driving head are assumed to continue indefinitely.

Direct Radiation On-site

In the hypothetical accident, fission products are released into the vapor container, and a radiation source exists at the site. Calculated radiation levels at a point external to the vapor container are based only on that activity which is not attenuated by the internal secondary shield. To obtain radiation levels, no credit is taken for any delay time due to slow melting of the core before release of the fission products begins. Figures 41 and 42 show gamma dose rates and integrated doses as a function of distance



GAMMA DOSE RATE FOR OUTSIDE EXPOSURE



INTEGRATED GAMMA DOSE
FOR
OUTSIDE EXPOSURE

from the sphere. The control room, which has additional shielding, provides a place where essential plant operating personnel can gather for protection from radiation. The total direct dose received under these conditions is less than 200 mr in the first hour and less than 2 r in the first 24 hours following the release. All plant personnel other than essential personnel will proceed to the guardhouse on the sounding of a clearly audible signal. Since a period of several minutes is available before a significant portion of the core melts, plant personnel can take stations or evacuate the plant without receiving harmful direct radiation doses.

Direct Radiation Off-site

Dose rate due to direct radiation emanating from the vapor container in the case of a large primary system rupture followed by partial core meltdown is shown in Fig. 41 as a function of distance from the source. Integrated doses for 5 minutes and for 1 hour after the instantaneous release as a function of distance are shown in Fig. 42.

The dose at the public road across from Sherman Pond, 1,300 ft from the vapor container, is 5 r during the first hour after release. Hence, several hours are available to remove persons and vehicles which might be on the road at the time to a safe distance from the site. This is based on the once-in-a-lifetime direct radiation dose limit of 25 r indicated in National Bureau of Standards Handbook 59.

Because the power plant is located at the bottom of a deep narrow valley, direct radiation from the vapor container does not reach inhabited buildings, except for one dwelling approximately 4,000 ft distant in the down-river direction. Since the integrated dose for the first hour is 20 mr, the occupants of this house could remain there for several weeks without serious exposure. In the up-river direction, there are no buildings within 9,000 ft, and beyond that point all buildings are shielded by hills.

All buildings east of the river and within one mile of the site are owned in fee by Yankee Atomic Electric Company or New England Power Company and are considered to be under administrative control of these two companies.

Vapor Container Leakage and Air-borne Radiation

In the hypothetical accident, 20 per cent of the gaseous and volatile fission products are assumed to be homogeneously dispersed in the vapor container. Leakage from the vapor container at the assumed leak rate will release these fission products to the atmosphere and, under certain meteorological conditions, they can be carried to populated areas where they may be inhaled or ingested.

Of the volatile and non-volatile fission products in the core, radio-iodine and radio-strontium provide the controlling activities with respect to the inhalation dose, with iodine being selectively absorbed by the thyroid and strontium by the bone. For the purpose of this report, it has been conservatively assumed that 20 per cent of all the iodine and strontium are released from the core even though the release of strontium has been reported more nearly 1 to 5 per cent. The total activity of iodine and strontium assumed to be in the core is:

Activity	Curies
Iodine - 131	1.0×10^7
Iodine - 132	1.6×10^7
Iodine - 133	2.3×10^7
Iodine - 134	2.7×10^7
Iodine - 135	2.1×10^7
Strontium - 89	1.3×10^7
Strontium - 90	2.4×10^5

Based on KAPL - 1178, it has been shown that the integrated 60-day dose to the thyroid from the inhalation of iodine - 131 is approximately a factor of 10 greater than the dose to the bone from the inhalation of a curie-equivalent of strontium - 89. Since the radio-iodine activity as iodine - 131 equivalent is approximately 1.8×10^7 curies as compared to the strontium - 89 activity of

1.3×10^7 curies, and since the dose to tne bone due to strontium - 89 is comparable to the dose due to strontium - 90, the iodine - 131 dose to thyroid was selected as the controlling dose.

The total radio-iodine activity emanating from the vapor container is assumed to have a concentration of 2.8×10^7 microcuries per cu ft. Based on the Sutton Continuous Point Source Equation and using in-valley meteorological conditions presented in Professor Austin's report the concentrations of radio-iodine, as iodine-131 equivalent, 100 ft below the center of a radioactive cloud, which is over the nearest inhabited area 4,000 ft away, are as follows:

Meteorological Condition	Wind Velocity, Fps	Radio-Iodine Concentration, Microcuries/ml
Inversion	3.3	1.6×10^{-6}
Moderate Lapse	19.7	6.9×10^{-9}
Unstable	16.4	2.3×10^{-9}

This tabulation and additional meteorological information indicate that the highest concentration of activity would occur under an inversion condition with a low velocity down-valley air movement. If the accident occurs under these conditions, the leading edge of the radioactive cloud reaches the nearest inhabited area approximately 20 minutes after release of fission products from the vapor container begins.

The once-in-a-lifetime off-site dose for ingestion and inhalation of air-borne radioactivity has not yet been established by the AEC, and there exists some difference of opinion on the subject. Lacking a definitive allowable dose, values suggested by K. Z. Morgan, W. S. Snyder, and Mary R. Ford in their paper, Maximum Permissible Concentration of Radioisotopes in Air and Water for Short Period Exposure, presented in 1955 at the Geneva Conference on the Peaceful Uses of Atomic Energy, have been adopted. These are:

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Dose Criterion Following Exposure	Maximum Permissible Radio-Iodine Concentration for 8 Hr Exposure, Microcuries/ml
0.3 rem in week	7.0×10^{-8}
15.7 rem in year	1.7×10^{-6}
150 rem in 70 yr	1.7×10^{-5}

Dr. Shields Warren has stated that he believes a dose of 50 rem to the thyroid may show clinically detectable effects, while a dose of 15.7 rem would probably provide no clinical indication. On this basis, 15.7 rem in the year following exposure has been taken as the off-site, once-in-a-lifetime internal dose.

Based on the assumption that a person is 4,000 ft from the plant and 100 ft below the center of the radioactive cloud, and taking no credit for radioactive decay, the doses received under various meteorological conditions are as follows:

Meteorological Condition	Thyroid Dose, rem in Year Following Exposure
Inversion	15.
Moderate Lapse	0.064
Unstable	0.021

Comparison of this tabulation with the 15.7 rem dose limit adopted shows that in all cases the dose received in 8 hours is below the limit. Thus, from this analysis it is clear that, even under the worst meteorological conditions, the hypothetical accident does not result in excessive concentrations of radioactivity in the nearest inhabited area.

506 CONCLUSIONS

This pressurized water reactor possesses inherent stability because of its negative temperature and Doppler coefficients. Since it is normally operated with the coolant moderator near its saturation temperature, further stability results from the formation of steam voids in any extensive rising power excursion. In addition to this inherent stability, mechanical control rods, capable of making the hot reactor subcritical, are provided to regulate power level and to control reactivity throughout the power production lifetime of the core. A supplementary chemical control system is provided to bring the reactor to cold shutdown.

The plant design and the selection of materials provide four sequential barriers to the escape of fission products. These barriers, in order, are:

Oxide Fuel - The noncorrosive UO_2 fuel acts as a first barrier to contain large percentages of the fission products within its matrix.

Stainless Steel Cladding - The fuel rod cladding with only two end welds per full length tube acts as a second barrier to escape.

Main Coolant System - The third barrier to escape is the high integrity main coolant system.

Vapor Container - This barrier acts as a fourth line of defense in the event of fission product release from the main coolant system.

An additional geographical barrier is inherent in the site selected for the plant. The nearest privately owned and occupied dwelling is approximately 4,000 ft away, and the population density within the first five-mile radius is 25 people per square mile.

Several reactor plant accidents have been investigated and analyzed. Accidents involving reactivity additions during start-up and at full power result in transients but give every indication of leveling off at power levels that are neither harmful nor dangerous. Accidents involving release of energy through chemical reaction between the water and the metallic constituents of the core are considered impossible because of the materials employed. Mechanical accidents in the form of pump failures with ensuing decrease or loss of coolant flow can be handled without dangerous power or reactivity excursions. In failures of one or two pumps, stability is regained with practically no increase in temperature level, even without scram. In failures of three or four pumps, however, a low flow scram occurs and the reactor is kept under control by this means.

Among the mechanical accidents that have been analyzed is one caused by a break in a 20 inch main coolant line at the worst possible location and involving loss of all water from the main coolant system. This is considered to be the maximum credible accident.

In none of these accidents is there any melting of the core, any release of gaseous and volatile fission products to the vapor container, nor any hazard to the public.

However, an analysis has been made of a hypothetical accident in which core melting and fission product release are assumed. An accident has been examined in which it is assumed that a large break occurs in the main coolant system; virtually all water is lost from the system; partial core meltdown occurs; and 20 per cent of the gaseous and volatile fission products are released to the vapor container. The analysis shows that there would be no hazard to the general public because of direct radiation from the vapor container. Since the vapor container has a finite leak rate, some of the fission products may escape to the atmosphere and, under certain meteorological conditions, the escaping fission products may be carried to nearby inhabited areas. At the nearest community, however, an 8 hour exposure to the indicated concentration of radioactivity, under the most unfavorable meteorological conditions, would result in less than tolerable once-in-a-lifetime inhalation and ingestion doses.

Yankee Atomic Electric Company, therefore, concludes that this reactor can be operated without undue hazard to the public health and safety.