CHAPTER 19
SAFETY ANALYSIS III - CONTAINMENT

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ABSTRACT

This chapter introduces the CANDU concept of containment. A review of containment definition, purpose, design and analysis is given. Containment practice together with examples of postulated accidents and thermohydraulic phenomena occuring within the containment system are described. This is followed by a brief description of how the containment response to a loss of coolant accident is simulated mathematically.

19.1 Background

19.1.1 Containment Definition

A wide variety of radioactive materials are generated within a reactor during the course of its operation. Most of these are formed within the fuel. Some are generated in the moderator, coolant, and other reactor parts. A major function of reactor system design is to prevent these materials, or radiation from them, from harming reactor systems, operating personnel, the public, or the general environment.

This "containment" function is achieved in CANDU reactors in several stages.

a) The fuel is a ceramic material which is formed into small fuel pellets. This material contains most of the radioactive materials during normal operation of the reactor.

b) The fuel pellets are enclosed within zircaloy tubes which contain any material which evolves from the pellets.

c) A number of these tubes are, in turn, assembled into the fuel bundles which are placed inside the primary coolant system pressure tubes. This closed system provides yet another container for the radioactive material of the fuel and the primary coolant itself.

d) Radioactive materials (primarily tritium) within the moderator are contained by that closed system.

e) The reactor assembly is finally surrounded by a massive structure of steel, concrete and water to absorb the radiation from the materials of the reactor.

f) The entire reactor assembly is enclosed within a final structure, the <u>containment</u>. This structure, during the course of normal day-to-day operation serves to control <u>minor</u> releases of radioactive materials from the reactor assembly. Its primary purpose is to contain the larger amounts of radioactive materials which might be released should the reactor components ever fail.

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We finally define the containment, herein, as the structure and the supporting systems which provide the final barrier to limit the release of radioactive material to the environment to easily tolerable levels.

19.1.2 Review of Containment Designs and Analysis

19.1.2.1 Containment Concepts

Containment system designs have generally evolved on the basis that they must be able to contain all of the steam and/or water discharged following a reactor system piping failure. The volume and mass of radioactive material which might be released is negligible in comparison. The primary element of most containment systems is the leak-tight building which covers and encloses the reactor systems. Piping or ventilation systems which might convey radioactive materials and penetrate the containment boundary are closed shortly after an abnormal condition is detected. Sub-systems to help reduce the pressure in the building may also feature in the design. Containment design types which have achieved enough prominence to acquire a name include:

a) Pressure Containment - Figure 19.1

A structure is provided to contain all of the energy contained in the steam and water released by failure of the primary coolant system. If the containment atmosphere is sub-atmospheric during reactor operation leakage may be into containment for some time following an accident. The leak-free period can extend for months.

b) Pressure - Suppression Containment - Figure 19.2

This type of containment depends on condensation of the vapor released by a loss of coolant to help reduce the design pressure of the containment. One common variation of this category is designed to pass the steam released into a pool of water where it is condensed. Another condenses the steam on ice.

c) Pressure-Release Containment - Figure 19.3a,b

This type of containment allows the venting of some or all of the released steam to the outside atmosphere. Radioactive materials are contained by closing the vents before significant amounts are released into the atmosphere and/orfiltering and scrubbing radioactive materials from the steam before it is released.

d) Multiple Containment - Figure 19.4a,b

Two (or more) containment systems, in series, may be provided. Since each can contain most of the fission products released into it, a very high degree of containment can be achieved. The containment shown in Figure 19.4b could, in theory, eliminate the release of radioative materials as leakage is pumped from the secondary containment back into the primary containment. The multiple reactor containment systems designed and built in Canada are also sometimes called multiple containment. The concept is quite different. Figure 19.6 illustrates the Canadian concept of a single containment shared by several reactors.

More history and details on variants of containment are given in Reference (1).

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19.1.2.2 CANDU Containment Systems

CANDU containment systems have evolved in a manner similar to containment practice elsewhere. Some unique features have developed from the Canadian experience.

The first CANDU power reactor, Nuclear Power Demonstration, NPD, at Rolphton, Ontario, is fitted with a pressure release containment. The next generation of CANDU reactors, Douglas Point and Gentilly-1, are fitted with a variant of pressure suppression containment (Figure 19.5) which condenses the discharged steam by a gravity fed water spray or "dousing" system. The dousing water is supplied from a storage tank located near the roof of the containment building. Subsequent single reactor CANDU designs have followed the same philosophy.

A decision to build four reactors at Pickering near Toronto has lead to the development of a unique containment concept. The four reactors of the original station are housed in buildings which are connected by large ducts to form a single containment system (Figure 19.6) shared by all four reactors. The large volume of the four buildings is able to accommodate a given discharge with a considerably smaller building pressure increase than that which would be experienced by a single building. The containment system includes a large volume building which is normally kept at nearly full vacuum. It is normally isolated from the remainder of the system by a number of large pressure relief valves. These valves are opened directly by rising pressure following loss of coolant in any one of the reactors and allow the escaping steam to vent into the vacuum building. The rising pressure in the vacuum building, in turn, forces a high flow of dousing water in the vacuum building to condense the inflowing steam. A very important feature of this containment system is the overall sub-atmospheric pressure of the system which ensues a few minutes after accident initiation. This ensures zero leakage to the environment for a few days after the accident. The Pickering installation has been expanded so that eight reactors now share a single vacuum building and containment system. The multiple reactor containment concept has subequently been applied to three additional four reactor complexes (Bruce A, Bruce B and Darlington).

19.2 Containment Analysis Practice

19.2.1 Goals

The primary purpose of containment analysis is to establish that any reactor failures which lead to a release of radioactive material pose an acceptable level of risk to the environment.

A detailed discussion of the risks posed by, and accepted for, power plant operation and the general philosophy of accident analysis has already been given in Chapter 2. The design and regulatory process outlined there leads to a requirement that the containment system response to a large number and variety of postulated system failures be considered. The estimated dose, which is a function of the quantity, type, and dispersion of released radioactive material, for the various events can then be compared with the established dose target for the probability of occurrence category within which the event falls.

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19.2.2 Postulated Accidents

19.2.2.1 Regulatory and Design Guidelines

The Canadian safety approach recognizes that the hazard people will accept is dependent on the frequency of the event which puts them at risk. Consequently the basic safety criteria regulations specify a permissible radiation dose to a member of the public resulting from three types of plant conditions, which are listed in order of decreasing probability:

i) normal operation
ii) following the failure of one of the process systems, and
iii) following the failure of one of the process systems coupled with the unavailability of one of the special safety systems incorporated to cater for the process system failure.

The latter two conditions are referred to as the single and dual failure conditions. Different radiation dose guidelines for each condition are specified, with a higher dose level permitted for the dual failure because of its lower frequency of occurrence. The maximum probability of failure of defined systems is shown in Figures 19.7. The two points defined by ii) and iii) above, together with the specified doses are shown in Figure 19.8.

Not all postulated accident scenarios fall neatly into the single/dual failure concept outlined. This risk concept can be logically extended in order to cater for low probability situations. The dotted lines of Figure 19.8 show the continuous guidelines derived from the points defined by the single-dual criteria. Chapter 2 discusses regulatory and safety philosophy in greater detail.

19.2.2.2 Examples

Postulated single failure events which figure heavily in containment design include primary and secondary piping failures of various sizes. Large pipe breaks can discharge steam at a high rate and thus usually define the maximum pressue for which containment must be designed.

Dual failure events requiring containment analysis include piping failures combined with safety system failures such as loss of the emergency coolant system <u>or</u> piping failures combined with containment failures in the dousing sub-system, the ventilation isolation sub-system valves, or airlock door seals.

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19.2.3 Phenomenology of Containment Thermohydraulic Analysis

19.2.3.1 Behaviour of Radioactive Materials

Although this lecture is primarily devoted to thermohydraulic phenomena the behaviour of radioactive materials needs to be kept in mind as a guide to the modelling of thermohydraulic phenomena.

A very large number of nuclides (several hundred) could conceivably be released to the containment following certain types of accidents. Each could be present in many different chemical forms. Fortunately, only a small number need to be considered in routine containment analysis.

Many of them will decay very rapidly (half lives of a fraction of a second). Their containment is not of great interest as they would disappear before they could spread very far in the environment.

Iodines and noble gases generally receive the most attention in containment analysis. They are present in sufficient quantities and have characteristics such that they create the bulk of the predicted risk to the public. The noble gases, since they do not condensate at normal conditions and are non-reactive will remain in the atmosphere and can potentially leak through the containment walls. The "hydraulics" of their leakage is thus quite important.

Iodine 131 with a half life of 8 days, can be contained long enough to allow the material to decay to negligible quantities (e.g., Iodine-131, rleased to the TMI containment about 3 years ago, is by now reduced by a factor of $\sim 10^{40}$. There is strong evidence that it was mostly contained within water in the containment. Its containment thus reduces to keeping the water within the structure. Some iodine, as a result of its reactive nature, evidently combines with materials in containment to form gases which may remain in the containment atmosphere. A substantial fraction of these can be removed by filters. Some may leak through the walls.

A great deal of knowledge with respect to the exact nature of the chemical and physical behaviour of radioactive materials is not necessary. Estimates of activity release into the containment for the postulated accidents can be deliverately set on the high side. The containment system is designed subsequently to accommodate the overestimate. Additional information on the behaviour of radioactive materials is actively sought by the nuclear industry. This knowledge is expected to lead to lower cost containment systems.

19.2.3.2 Thermohydraulic Phenomena

A large number of thermohydraulic phenomena are important to the design and safety analysis of containment. These are revealed as we follow the steam discharged from a broken pipe through typical CANDU containment systems.

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During the early part of a typical postulated discharge of coolant high pressure and temperature steam will be discharged to the low pressure air atmosphere of the containment. The high pressure water will tend to be mixed with the containment building atmosphere by the jet of steam. The containment atmosphere pressure will begin to rise as a result of this addition of mass and energy. The air/steam/water mixture is displaced from the reactor vault raising the pressure throughout the containment atmosphere.

In the case of the Douglas Point single unit containment the displaced steam is passed through a tunnel-like opening which is fitted with a dousing spray fed from an overhead tank. The flow of water is thus ensured by the supremely reliable system known as gravity. The flow velocity of steam is measured. The flow of water is proportionally controlled by valves to condense the steam flow. This dousing design, which directs the steam and dousing water into intimate contact, is quite effective in suppressing the rise in pressure following loss of coolant. The design pressure is quite low ($\sim$ 240 kPa (6 psig)) as a result.

The dousing system of the CANDU standard 600 MWe single units is somewhat simpler.

The rising pressure in the building is measured. When it reaches an upper building "setpoint", two dousing systems which are supplied with water from an overhead tank and controlled by valves are turned on. Should the water condense sufficient steam to drop the building pressure below a lower "setpoint" the valves are closed to conserve dousing water. The water is sprayed from high in the building to cover most of the building volume. The pressure in the containment of the reactors may remain somewhat above atmospheric following a loss of coolant. There is thus some potential for leakage from the building. The containment is, as a result, designed to a high standard of leak tightness. The design target leak rate is 0.1%/day at design pressure. Containment analysis is based on 0.5%/day at design pressure. Should we imagine all the leakage concentrated in a single orifice this is equivalent to an opening about 6 mm in diameter. Leakage rates of 0.15%/day have been achieved in practice.

The response of the multiple reactor containment system is somewhat more complex and presents additional interesting opportunities for thermo-hydraulic modelling of wave propagation, dynamic valve response, and dousing water flows.

The increasing pressure in the ducts generates hydraulic forces which act to open the relief valves allowing a discharge into the vacuum building. The resultant rising pressure in the vacuum building, in turn, generates hydraulic forces on the fluid surfaces of the water in the dousing system. This causes the water to flow from the dousing system in a spray which condenses the incoming steam to maintain a relatively low (general sub-atmospheric) pressure therein for a few days. The entire system is quite simple and reliable since the rising pressure is all that is needed to actuate the valves and dousing spray.

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The discharge of steam continues at a decreasing rate in all the reactor systems discussed. The emergency coolant systems establish an alternate source of cooling. The water spilled into containment is cooled and circulated back through the reactor core to remove decay heat for an indefinite period of time.

Throughout this period the building air coolers have also been acting to remove energy from containment and reject it to the outside atmosphere. Cool surfaces within the building serve to temporarily store energy absorbed from the hot steam and reduce somewhat the maximum pressure, experienced by the containment envelope.

The remaining section is devoted to a discussion of the mathematical simulation of some of these important containment phenomena.

19.3 Mathematical Simulation of CANDU Containment Response Following A Postulated Loss of Coolant Accident (LOCA)

19.3.1 Event Sequence

Following a loss of coolant accident (in which steam or two-phase water escapes from a pipe rupture into containment) the "breakroom" pressure begins to rise. If the pressure rise is sufficient a pressure sensor opens valves connecting the dousing tank (located in the roof of the reactor building) to the sprayheaders (for single unit stations - Fig. 19.9a) causing water from the dousing tank to flow through the sprayheaders and finally through the spray nozzles. The escaping spray of relatively cold water from the surrounding relatively hot fluid (coolant) during its decent to the bottom of the reactor building. This dousing spray, as it is commonly referred to, acts as en energy heat sink and thus as a pressure suppression mechanism. This pressure suppression dousing system must be designed so that the design pressure of the whole containment structure is not exceeded for the largest anticipated break.

Multiple reactor stations such as Bruce and Pickering are equipped with a separate vacuum building in which the dousing tank is located (Fig. 19.17). Following a pipe rupture in a reactor building the increase in pressure causes the resulting mixture of air + water (vapour + droplets) to move down the pressure relief duct into the manifold (the region of containment in which the pressure relief valves are located) and the other reactor buildings. When the pressure in the manifold is sufficient to lift the valve pistons the pressure relief valves will be opened permitting the fluid to flow into the vacuum building (which is initially at a pressure substantially below atmospheric (i.e. $\sim$ 10 kPa). The air and steam mixture entering the vacuum building causes the vacuum building pressure to rise thus forcing water in the dousing tank to flow up the riser pipes, over the weir, down the downcomer ducts and finally through the sprayplates in the sprayheaders (Fig. 19.18). For illustrative purposes Figures 19.10 and 19.11 show a typical short term pressure transient in the breakcomer and vacuum building respectively for a multiple reactor containment. Figure 19.11a shows how the containment pressure transient develops over a period of several days.

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19.3.2 Physical Phenomena

A steam discharge from a pipe in containment initiates a fluid flow transient throughout containment. In addition to the dousing heat sink mechanism previously described, there exists other potential heat sinks within containment. These include coolers, the containment walls and internal structures within the containment such as airways, railings, piping components, etc. These heat sinks become the prime heat sink in the case of small breaks in which the pressure rise is insufficient to activate the dousing heat sink mechanism and/or following the completion of dousing.

In order to predict pressure and temperature transients within containment as well as the amount of radioactive material which may escape to the environment, we must set up an integrated mathematical model which adequately describes the many physical response processes involved.

This involves setting up individual mathematical models for the fluid dynamics of the system, valve dynamics, dousing flow, coolers, wall and internal structure heat conduction systems as well as a fission transport model.

Having set up the individual models we must then resort to a numerical method which will provide use with a reasonbly accurate solution of the differential equations describing the physical system.

The code PRESCON2 is the latest of several computer models that have been developed at AECL for containment analysis. We will now proceed with a brief description of the individual mathematical models contained within the code followed by a description of the numerical algorithm employed.

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19.3.3 Mathematical Models

i) Global Fluid Flow Modelling:

The partial differential equations governing the flow of the steam-air mixture are those of mass, energy and momentum conservation for one-dimensional compressible isentropic flow. Spatial discretization of these basic equations leads to the commonly called node-link structure in which the physical geometry being considered is represented by a network consisting of a set of nodes connected by links. A typical non-critical link (a link which has an initial and a terminal node) is shown in Figure 19.12.

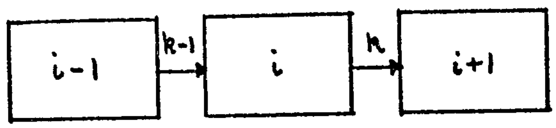

Figure 19.12: Non-critical Links

Link k is assigned node i as its initial node (upstream node) and node i + 1 as its terminal node (downstream node). This assignment defines the direction of positive flow in the link as being from node i to node i + 1. Link k is also referred to as a downstream link of node i and as an upstream link of node i + 1.

Figure 19.13 shows a particular partitioning of a four station containment structure into a node-link network.

Associated with each node in the network is a total mass, total air mass and a total mixture energy equation reflecting conservation of these quantities within the node.

These equations (discretized forms of the governing partial differential equations) take the following forms.

Total Mass Conservation

$$\frac{dTM_i}{dt} = \sum_{k\varepsilon n} S_k W_k + \dot{Q}_{TM} \qquad (1)$$

where

TM_i = total mass (air + water) in node i (kg)

S_k = Sign of link connection (+1 for a downstream link, -1 for an upstream link)

W_k = link flow rate (kg/sec)

$k\varepsilon n$ = sum over all links connected to node i

$\dot{Q}_{TM}$ = total mass addition/removal rate within node i (i.e., any other source/sink term in addition to the link flow rate)

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Air Mass Conservation

$$\frac{dAM_i}{dt} = \sum_{k \varepsilon n} S_k \left(\frac{AM}{TM}\right)_d W_k + \dot{Q}_{AM} \qquad (2)$$

where AM_i = air mass in node i (kg)

$\left(\frac{AM}{TM}\right)_d$ = air mass fraction of donor node connected to node i via link k

$\dot{Q}_{AM}$ = air mass addition/removal rate within node i (i.e., any other source/sink term in addition to the link flow rate).

Energy Conservation:

$$\frac{dET_i}{dt} = \sum_{k \varepsilon n} S_k \left(h + \frac{V^2}{2}\right)_d W_k + \dot{Q} \qquad (3)$$

where ET_i = total energy (internal + kinetic) in node i

h = specific enthalpy of donor node connected to node i via link k = $\left(\frac{ET + PV}{TM}\right)_d$ P = pressure, V = volume

$\frac{V^2}{2}$ = kinetic energy per unit mass of donor node

$\dot{Q}$ = heat generation/removal rate in node i

Associated with each link k in the network is a spatial finite difference form of the one-dimensional momentum equation.

This equation may be written in general form as

$$\frac{dW_k}{dt} = W_k \left(P_i, P_{i+1}, P_{i-1}, W_k, W_{k-1}, W_{k+1} \ldots\right) \qquad (4)$$

where W_k = mass flow rate kg/sec

P = Pressure

Equations (1) - (4) are coupled through the pressure P which is usually given by an equation of state of the form

$$P_i = P_i \left(ET_i, TM_i, AM_i, W_k, W_{k-1} \ldots\right) \qquad (5)$$

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Assuming that there are N nodes and K links in the network, the set of equations (1) - (4) for all nodes and links constitute a set of 3N+K ordinary differential equations with known initial conditions $TM_i(o)$, $AM_i(o)$, $ET_i(o)$ and $W_k(o)$.

Denoting the RHS of equation (1) - (4) by $f_{1i} + \dot{Q}_{1i}$, $f_{2i} + \dot{Q}_{2i}$, $f_{3i} + \dot{Q}_{3i}$ and f_{4k} respectively for i=1,...N, k=1,...K

where

$$f_{1i} = f_{1i}(W_1,\ldots W_K) \qquad f_{2i} = f_{2i}(TM_i\ldots TM_N,\ AM_1\ldots AM_N,\ W_1\ldots W_k)$$

$$f_{3i} = f_{3i}(TM_1\ldots TM_N,\ ET_1\ldots ET_N,\ P_1\ldots P_N,\ W_1\ldots W_K) \qquad f_{4k} = f_{4k}(P_1\ldots P_N,\ W_1\ldots W_K)$$

one can express the set of equations (1) - (4) in vector form as

$$\dot{\vec{Y}} = \vec{F}(\vec{Y}) + \dot{\vec{Q}} \tag{6}$$

where $\vec{Y} = (TM_1\ldots TM_N,\ AM_1\ldots AM_N,\ ET_1\ldots ET_N,\ W_1\ldots W_K)^T$

$$\vec{F}(\vec{Y}) = (f_{11}\ldots f_{1N},\ f_{21}\ldots f_{2N},\ f_{31}\ldots f_{3N},\ f_{41}\ldots f_{4K})^T$$

$$\dot{\vec{Q}} = (\dot{Q}_{11}\ldots\dot{Q}_{1N},\ \dot{Q}_{21}\ldots\dot{Q}_{2N},\ \dot{Q}_{31}\ldots\dot{Q}_{3N},\ 0\ \ldots\ 0)^T$$

T denoting the transpose, where $\dot{\vec{Q}}$ is a column vector representing various source/sink mass/energy terms within the system such as coolers, heat sources, breakflow addition, heat transfer to walls and internal structures, etc.

A first order semi-implicit integration scheme is used to advance the time dependent system of equations (6), the source/sink terms being treated in an explicit fashion (i.e., explicit coupling with the basic flow equations). Further discussion of the choice of numerical methods used will be given in a later section.

19.3.4 Sub-System Modelling

In addition to the basic fluid flow there are many other components of containment whose overall effect on containment pressure and temperature following a LOCA must be taken into account. Examples of these components are coolers, pressure relief valves, the dousing pressure suppression mechanism etc. As space does not permit a detailed description of how these sub-components of containment are modelled mathematically only a brief description will be given here.

i) Pressure Relief Valves (Multi-Unit Stations)

The pressure relief valves connect the vacuum building to the remainder of containment, providing the flow path for the escaping steam into the vacuum building. A schematic of the relief systems is shown in Figure 19.14. Figure 19.15 depicts a typical pressure relief valve and Figure 19.16 is a schematic of a Bruce Pressure Relief Valve in the closed position.

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During normal operation of the station, the valve is held closed by the piston weight and the downward force exerted on the lower diaphragm by the pressure differential between the lift chamber and the vacuum duct. When the manifold pressure rises following a LOCA the lift chamber pressure also rises because of pressure equalizing the flow through the piston orifices. When the pressure differential acting on the piston and the upper diaphragm overcomes the piston weight and the forces exerted by the lower diaphragm, the piston begins to lift and allows the steam-air mixture to flow from the manifold into the vacuum building via the vacuum duct. The valve piston dynamics are calculated by solving simultaneously, with a Runge-Kutta integration, a set of differential equations for mass contents, in the piston and lift chambers and for the piston motion. The calculated fractional opening of the valve at any time is then used together with the pressure ratio across the valve to give the mass flow rate through the valve (an experimental correlation is used which includes choking effects). This mass flow rate is then used to evaluate the mass/energy/source/sink terms in equations (1)-(3) for the manifold and the vacuum building.

ii) Dousing Spray System

Description of the Water Spray System (Single Unit Containment)

As mentioned previously, in single unit stations such as Douglas Point, Gentilly-2 and Point Lepreau, the dousing tank is located near the roof of the reactor building. If steam is released to the containment, the positive pressure created by the steam activates the dousing valves and initiates dousing at a preset value. The spray is produced by nozzles set in a system of headers, suitably arranged in the upper portion of the reactor building. Steam condenses in the reactor spray and this quickly reduces the pressure. The dousing stops when the pressure drops below another preset valve. This cycle is repeated until the pressure in the RB (reactor building) has stabilized or the water in the dousing tank has been exhausted.

The mathematical model of the spray system is fairly simple. The dousing flow rate builds up to a maximum after an initial delay (due to valve opening and flow acceleration) and then decreases slowly as the water level in the tank drops. When the containment pressure has dropped below the present minimum value, the valves start to close. The flow rate drops sharply until the valves are completely shut and the spray stops. This cycle is repeated each time the RB pressure exceeds the valve opening pressure. Fig. 19.9b shows the dousing spray rate as a function of time. There is a separate model for calculating the amount of energy picked up by the dousing sprays. The energy absorption rate is treated as a sink form in the energy equation in the nodes in which spraying takes place.

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Description of the Water Spray System (Multiple Containment)

If as a result of a loss of coolant accident, the internal pressure of the reactor building (RB) should rise, the steam-air mixture from the RB will be transferred through the pressure relief system, into the vacuum building (VB). The VB is a cylindrical reinforced concrete structure, housing a water spray system or dousing system (Figures 19.17-19.19).

The steam-air mixture which enters the VB increases the pressure and activates the dousing system. The water spray from the dousing system condenses the incoming steam and also cools the air in the VB. The spray system therefore reduces the rate of pressure rise and the maximum pressure in the VB following a LOCA. This in turn increases the VB's effectiveness as a containment pressure suppression device.

The spray system includes the emergency water storage tank, the pressure actuated water displacement system, inlet or suction pipes (riser ducts), a weir, vacuum chamber, central passage and a distribution of sprayheaders. The vacuum chamber (VC) is mounted centrally in the roof of the VB and is isolated from the VB main vacuum space by the water seals formed by the submerged riser ducts and the lower water seal. During normal operation the pressure in the VC is usually maintained equal to the VB pressure by separate vacuum pumps. The central passage is a vertical concrete duct which carries the water from the weir to the lower water seal and then to the spray headers (Figure 19.18). To assure a water seal between the main volume and the VC at all times, a small recirculating system continuously discharges a small quantity of water into the seal.

Description of the Dousing Process (Multiple Reactor Containment)

The flow of the steam air mixture into the vacuum building raises the VB pressure. The water seals prevent this pressure from communicating with the upper vacuum chambers. Under the influence of the pressure difference thus created, water is accelerated up the riser pipes. Air which enters through the spray nozzles pushes some of the lower seal water up the central passage. The water from the riser ducts reaches the weir first since its source is at a higher elevation, and flows downward into the central passage to meet the rising lower seal water. When sufficient water has fallen onto the lower seal water in the central passage, the water flow reverses and starts downward in the central passage.

Spraying then commences and will continue as long as sufficient pressure is available to cause the water to flow over the weir. When the VB main volume pressure falls to such an extent that it balances the lowered water storage tank level and the upper chamber vacuum pressure, dousing will stop.

The mathematical model of the spray system calculates water spray system flows, velocities, water level changes and the total quantity of water sprayed. There is a separate model for calculating the amount of energy picked up by the dousing spray; this energy absorption rate is treated as a sink term in the energy equation for the vacuum building.

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iii) Coolers

Considerable theoretical and experimental work has been directed towards modelling the containment coolers. These studies support the cooler model used in the PRESCON2 code. The model estimates cooler capacity as a function of atmosphere and cooling water temperatures and is multiplied by an empirical factor to account for the fact that the containment atmosphere is a steam-air mixture. Coolers act as an energy sink and are included as an energy sink term in equation (3) in the node in which they are located.

iv) Heat Sources

There are many sources of heating and cooling inside the containment e.g., lights, motors, the hot operating reactor, cooling water pipes, etc. which are constantly adding or removing heat. The net heat added/removed by these sources in each node is modelled in the $\dot{Q}$ term in the energy equation (3).

v) Wall and Internal Structures Heat Conduction

The containment walls and the internal structures are energy sources/sinks within containment. The walls consist of a thick layer of concrete. Some CANDU containment systems are fitted with a partial steel liner. The classical unsteady one-dimensional heat conduction equation is used to calculate the temperature transient and hence energy distribution within the wall. The rate of heat transfer between the containment walls and the surrounding containment atmosphere is calculated using the standard expression

$$\dot{Q} = hA\ (T_W - T_C)$$

where h = heat transfer coefficient based on correlations of relevant experimental data.

A = surface area of wall

T_w = wall surface temperature

T_C = nodal temperature

This term is included as a source/sink term in the energy equation for each node.

vi) Leakage and/or Impairments in Containment

There will be a transfer of mass and associated energy between the containment and the external atmosphere due to either ordinary building leakage or an impairment such as a postulated leaky airlock seal. Models are employed in the code for modelling various types of fluid flow through these paths. They are treated as part of the source/sink terms in the mass and energy equations for the particular node in which the impairment is located.

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19.3.5 Numerical Methods

In order to advance the solution of the spatially discretized flow equations (together with their various source/sink terms) we must choose an appropriate numerical algorithm for the solution of the set of ordinary differential equations modelling the system. In selecting a numerical algorithm both economic and stability considerations must be taken into account.

There are two basic numerical methods from which to choose, these being explicit and implicit techniques. Explicit methods, although much easier to program, invariably suffer from the drawback that there is a stability criterion associated with them limiting the time step which may be used to obtain a stable solution. In certain problems the running costs of a code employing such a method can become prohibitive.

Implicit techniques on the other hand are usually unconditionally stable for physical systems thereby permitting the use of a much larger time step while still maintaining a reasonable degree of accuracy. On a cost per time step basis explicit techniques are economically superior to implicit techniques since the latter schemes invariably involve the inversion of a matrix at each time step. Provided that the transient is not too rapidly varying as a function of time (as in containment problems) implicit techniques are economically superior overall due to the fact that the time step which may be used is sometimes orders of magnitude larger than the maximum explicit time step permitted.

In the PRESCON2 code a semi-implicit numerical algorithm is used to advance the solution of the time dependent flow equations. It is semi-implicit due to the fact that the source/sink terms are treated in an explicit-fashion. The program time step is automatically adjusted based on prescribed user tolerances on the change in state variables (i.e., pressure, energy, mass) over a time step.

19.3.6 Code Verification

Code verification against analytic solutions (where possible) and experimental results forms an important part of overall code development. Wherever possible mathematical models comprising the code should be verified on an individual basis in order that an assessment of the particular model can be made. If possible the integrated code should be verified against an experiment which encompasses all sub-models comprising the code. This is required for global verification. There exist a host of compressible flow experiments, model containment experiments, etc. against which the code may be benchmarked. Some of the benchmark problems against which PRESCON2 has been verified will be presented here. These include a shock tube simulation, a model containment blowdown simulation, a semi-analytic numerical heat transfer problem, and a simulation carried out at the Sheridan Park Engineering Laboratory.

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Shock Tube Simulation: In order to make a direct comparison of the ability of the code to model a strong pressure wave, the code was applied to a standard shock tube problem[2]. The shock tube consisted of a high pressure region initially at 2 atm and a low pressure region at 1 atm in a duct separated by a diaphragm which was ruptured at time t=0. The physical geometry, node structure and initial conditions used are shown in Figure 19.20.

Figures 19.21 and 19.22 show the temporal development of pressure in node 14 (located in the low pressure region) corresponding to the experimental results and code prediction respectively. Figures 19.23 and 19.24 show the corresponding results for the temporal development of pressure in node 20. The smoothed prediction of the discontinuous pressure rise is due to numerical diffusion (characteristic of implicit methods). A better approximation to the step rise in pressure could be obtained by using a finer mode structure. This phenomena is not of great interest to containment modelling. The approximation to wave modelling indicated is satisfactory for the purpose of evaluating the consequences of relief valve opening delay due to the time required to propagate a pressure wave down the long duct connecting reactor vaults to the vacuum building.

Model Containment Blowdown Simulation: PRESCON2 has been benchmarked against the OECD-CSNI containment analysis standard problem no. 2. The model test facility consists of a high pressure coolant system and a model containment divided into several compartments (Figures 19.25 and 19.26). The test runs were performed by discharging a two-phase mixture into the containment for a period of 50 s. Some of the PRESCON2 code predictions[3] are given in Figures 19.27 - 19.32 together with the experimental results (the solid dark curves represent the code predictions).

CSNI Numercial Benchmark Problem - Heat Transfer to Walls: This semi-analytic problem was proposed by A.R. Edwards as a means of studying the convergence of the numerical solution when the mesh size in the wall is refined. To a lesser extent, the problem provided a good check on coding in the relevant sub-routines such as the water property routine. This problem was a one node model with the wall temperature prescribed as a function of time. This allowed an analytic solution of the heat conduction equation into a semi-infinite solid (a valid approximation for small times). Specification of a fixed heat transfer coefficient at the wall permitted containment temperature as a function of time to be determined and hence discharge rates used as input to the code could be calculated. Figures 19.33 - 19.35 show the comparison of the PRESCON2 code predictions with the semi-analytic results for containment pressure, temperature and wall surface temperature respectively. The code predictions are in excellent agreement with the analytic results.

Long Duct Test: This experimental test facility was set up at the Sheridan Park Engineering Laboratory in order to compare code predictions with experimental blowdowns of air in a long duct. Figure 19.36 illustrates the transient rig geometry. In general the test rig consists of a 7.67 m^3 pressure reservoir supplying up to 790 kPa air through a flow nozzle and quick opening butterfly valve to a 1.5 m^3 pressure vessel. Ducts of 15 cm diameter and lengths up to 90 m were connected to this first pressure vessel and a series of increasingly complex downstream piping arrangement were constructed by the addition of a

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second pressure vessel, elbows, tees and branch ducts. Basically four layouts were considered in the experimental program. Some code predictions for Layout #1, Figure 19.37 are presented here. Figure 19.38 shows the code predictions and experimental results for the reservoir pressure of 275 kpa. Figure 19.39 shows corresponding results for the Pressure Vessel.

19.3.7 Summary and Closure

In this short lecture we have given an overview of the function of containment together with a brief description of how the complex containment system is modelled mathematically (basically fluid flow and coupled sub models). Selected comparisons of code predictions with experimental results are shown.

References

1) Cottrell, W.B. and A.W. Sauolainen, Editors, U.S. Reactor Containment Technology, Oak Ridge National Laboratory, ORNL-NSIC-5, UC-80-TID-4500, 1965 August.

2) P.W. Huber, C.E. Fitton, F. Delpino, "Experimental Investigation of Moving Pressure Disturbances and Shock Waves and Correlation with One Dimensional Unsteady Flow Theory", national Advisory Committee for Aeronautics, Technical Note 193 (1949).

3) W.M. Collins, "PRESCON2 Simulation of the OECD/CSNI Containment Analysis Standard Problem No. 2, Proceedings of the 8th Simulation Symposium on Reactor Dynamics and Plant Control, University of Toronto, March 17, 18, 1981.

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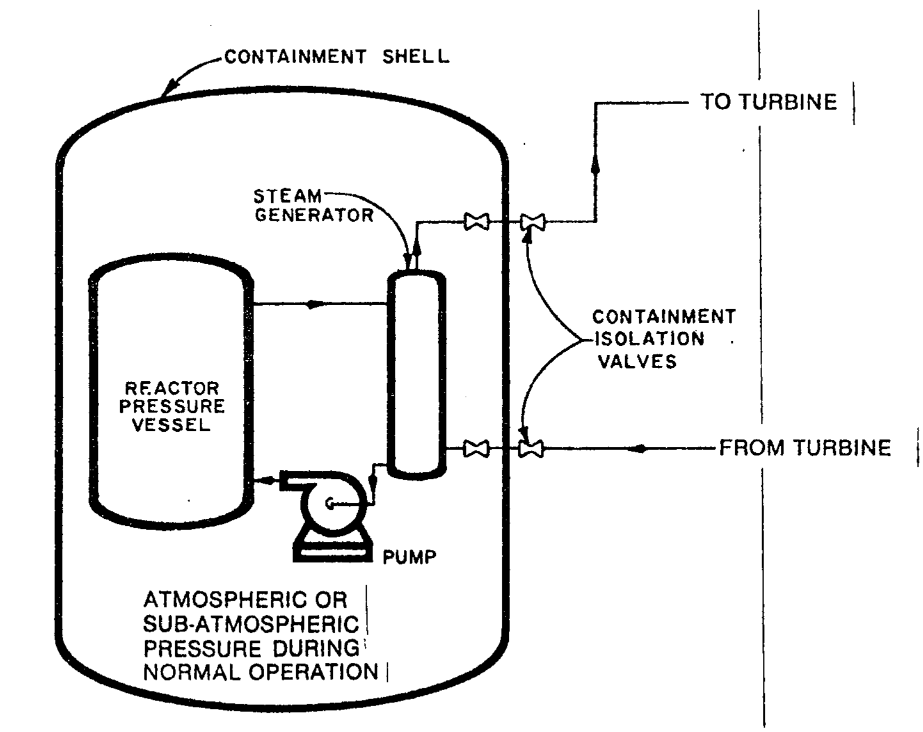


FIGURE 19.1 PRESSURE CONTAINMENT

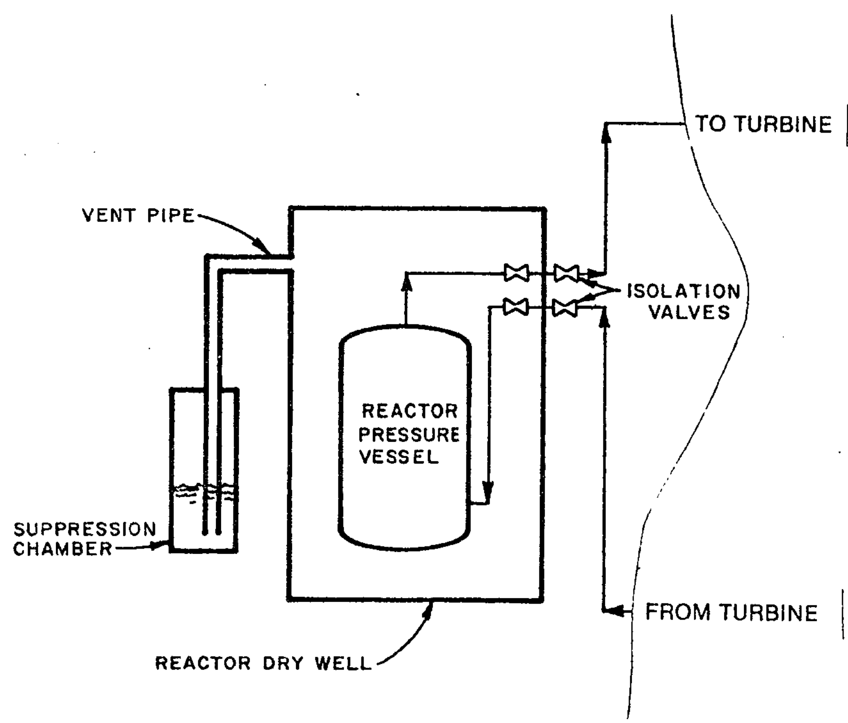


FIGURE 19.2 PRESSURE SUPPRESSION CONTAINMENT

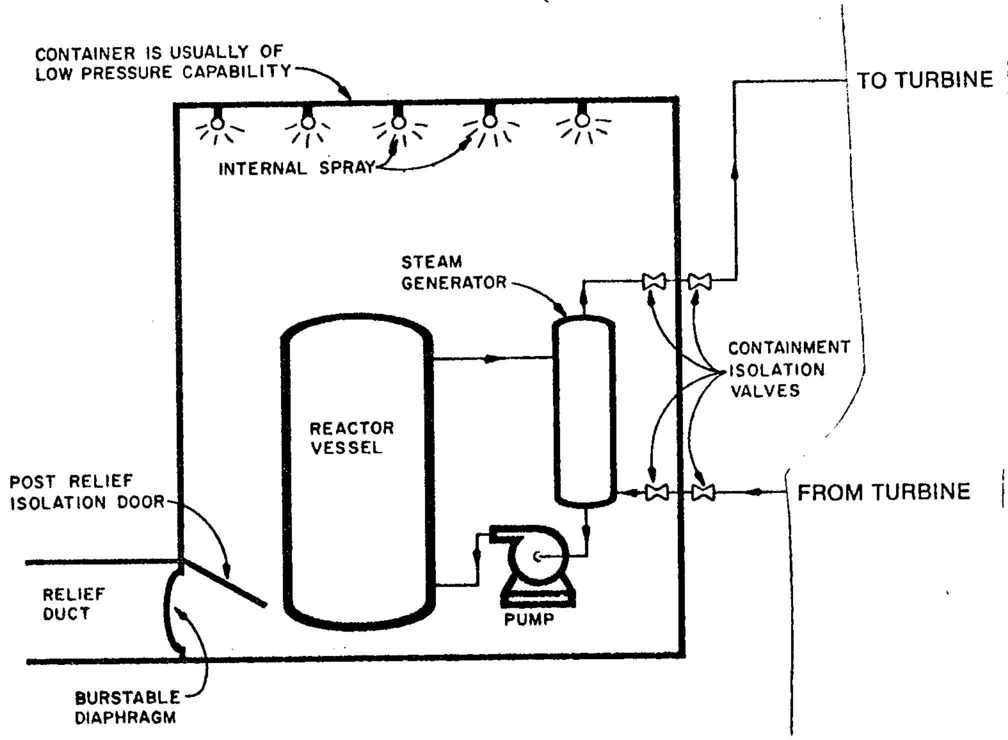


FIGURE 19.3a PRESSURE RELEASE (RELIEF) CONTAINMENT

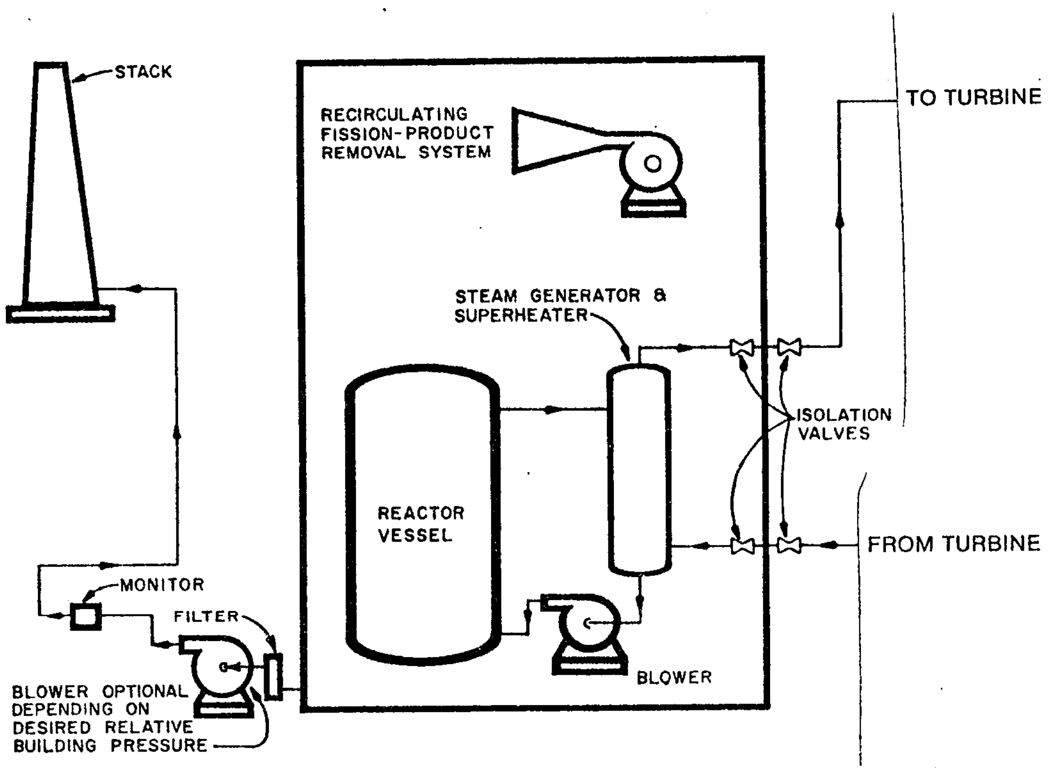


FIGURE 19.3b PRESSURE RELEASE (VENTING) CONTAINMENT

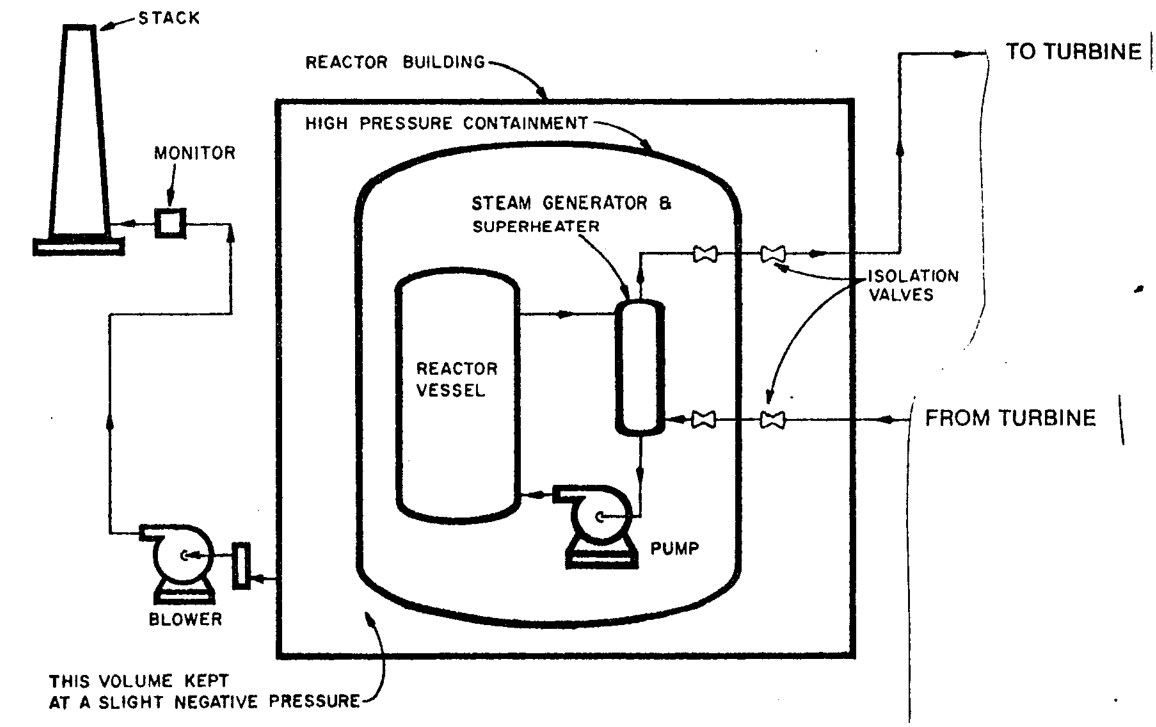


FIGURE 19.4a MULTIPLE CONTAINMENT (BASIC APPLICATION)

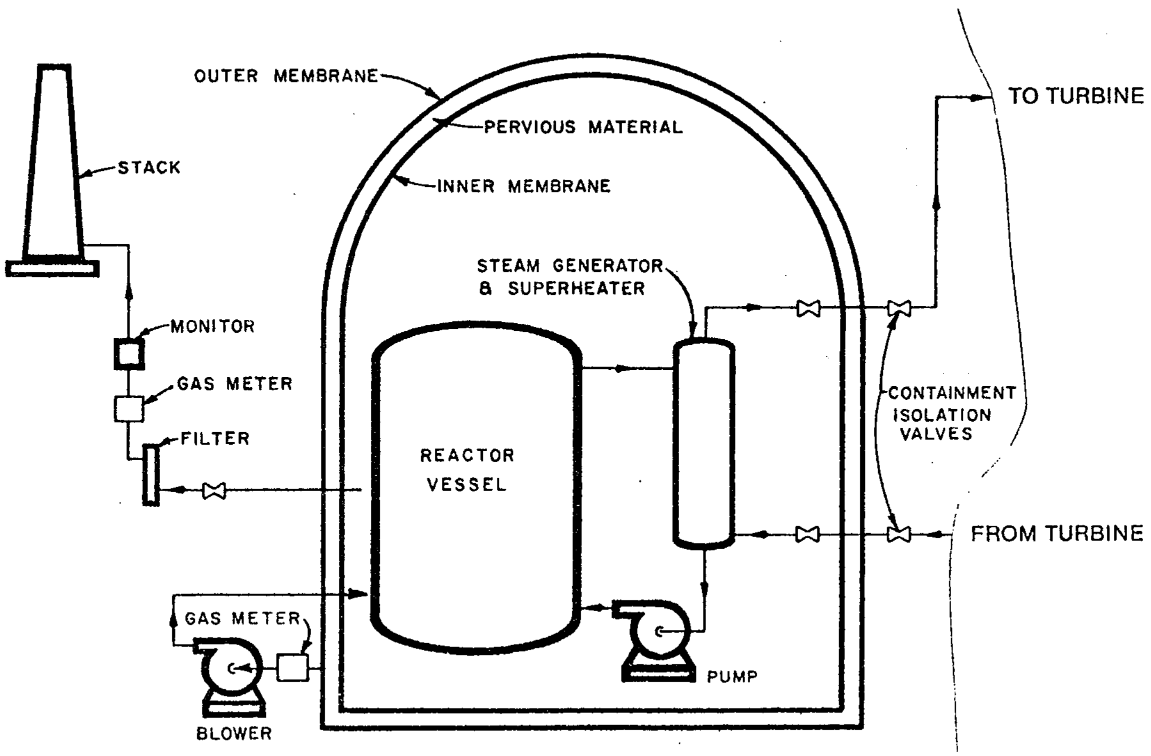


FIGURE 19.4b MULTIPLE CONTAINMENT (ADVANCED CONCEPT)

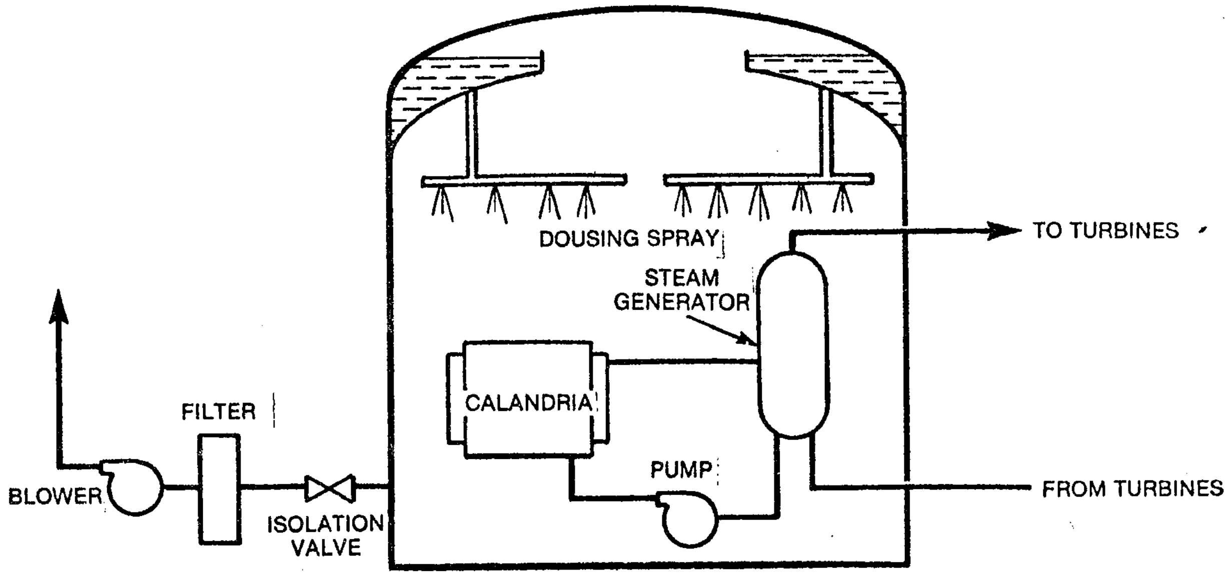


FIGURE 19.5 CANDU SINGLE UNIT PRESSURE SUPPRESSION

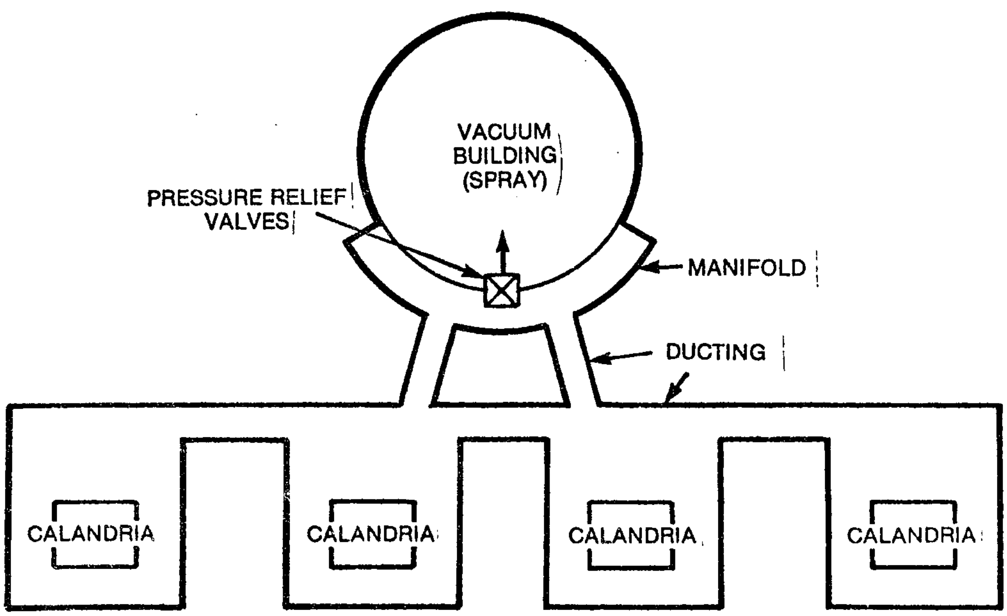


FIGURE 19.6 CANDU MULTIPLE UNIT

Figure 19.7

Reference Dose Guidelines for Plant Design Conditions

Situation	Event Frequency	Individual Dose Guideline	Integrated Dose Guideline
Normal Operation		0.5 rem/yr whole body* 3 rem/yr thyroid	10^4 man-rem/yr 10^4 rem/yr thyroid
Process failure (single failure)	More than 1 per 3 yrs	0.5 rem whole body 3 rem thyroid	10^4 man-rem 10^4 thyroid-rem
Process failure when a safety system is unavailable (dual failure)	More than 1 per 3 x 10^3 years i.e(1/3) x (1/1000)	25 rem whole body 250 rem thyroid	10^6 man-rem 10^6 thyroid-rem

* rem-R/oengten) e(equivalent) in m(an) - the quantity of ionizing radiation whose biological effect is equal to that produced by one roentgen of X-rays.

LIMIT LINES ARE DRAWN THROUGH THE POINTS
GIVEN IN THE FOLLOWING TABLE

Events / year	WHOLE BODY	THYROID
0.3	0.5 rem	3 rem
3×10^{-4}	25 rem	250 rem

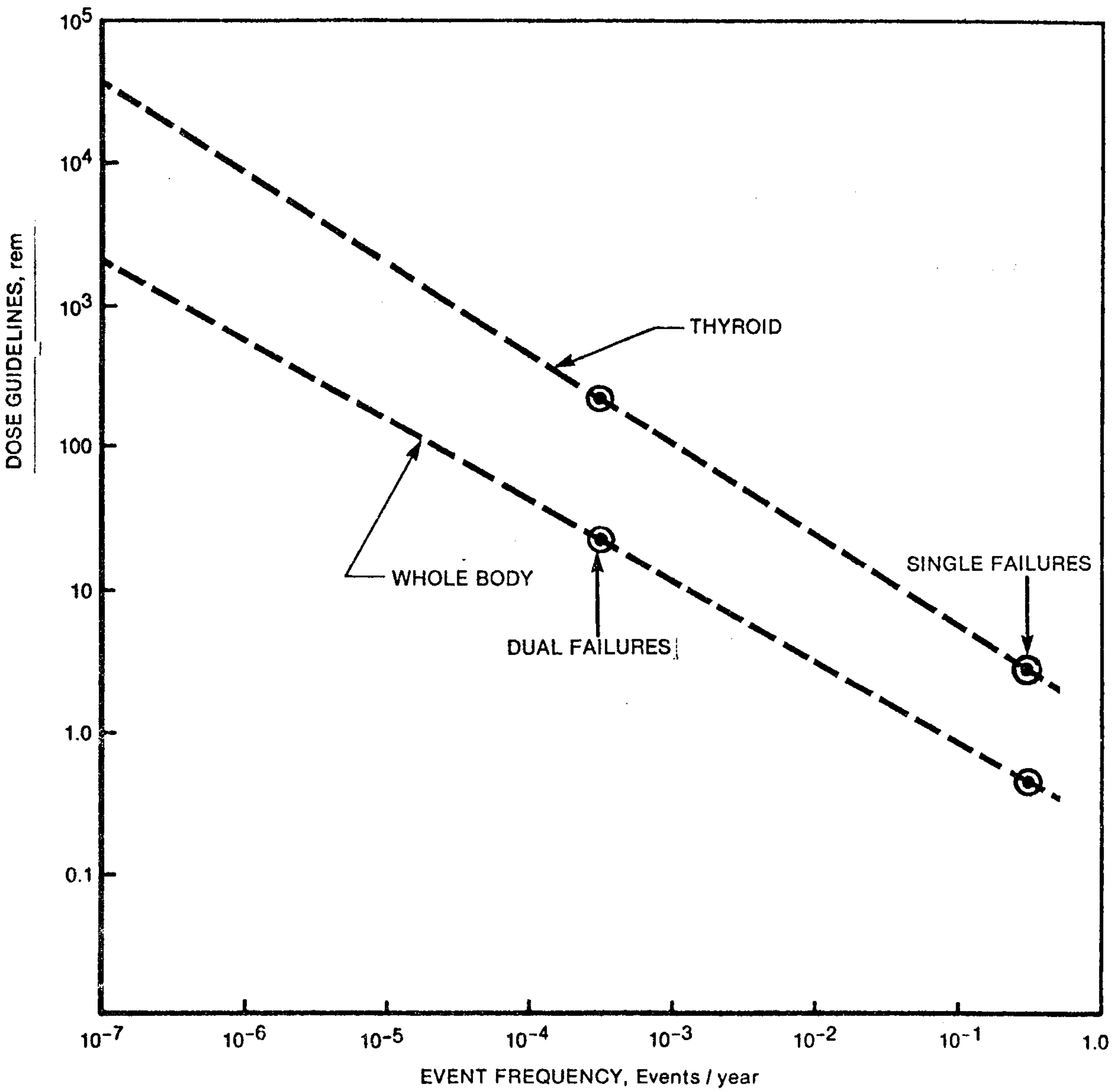
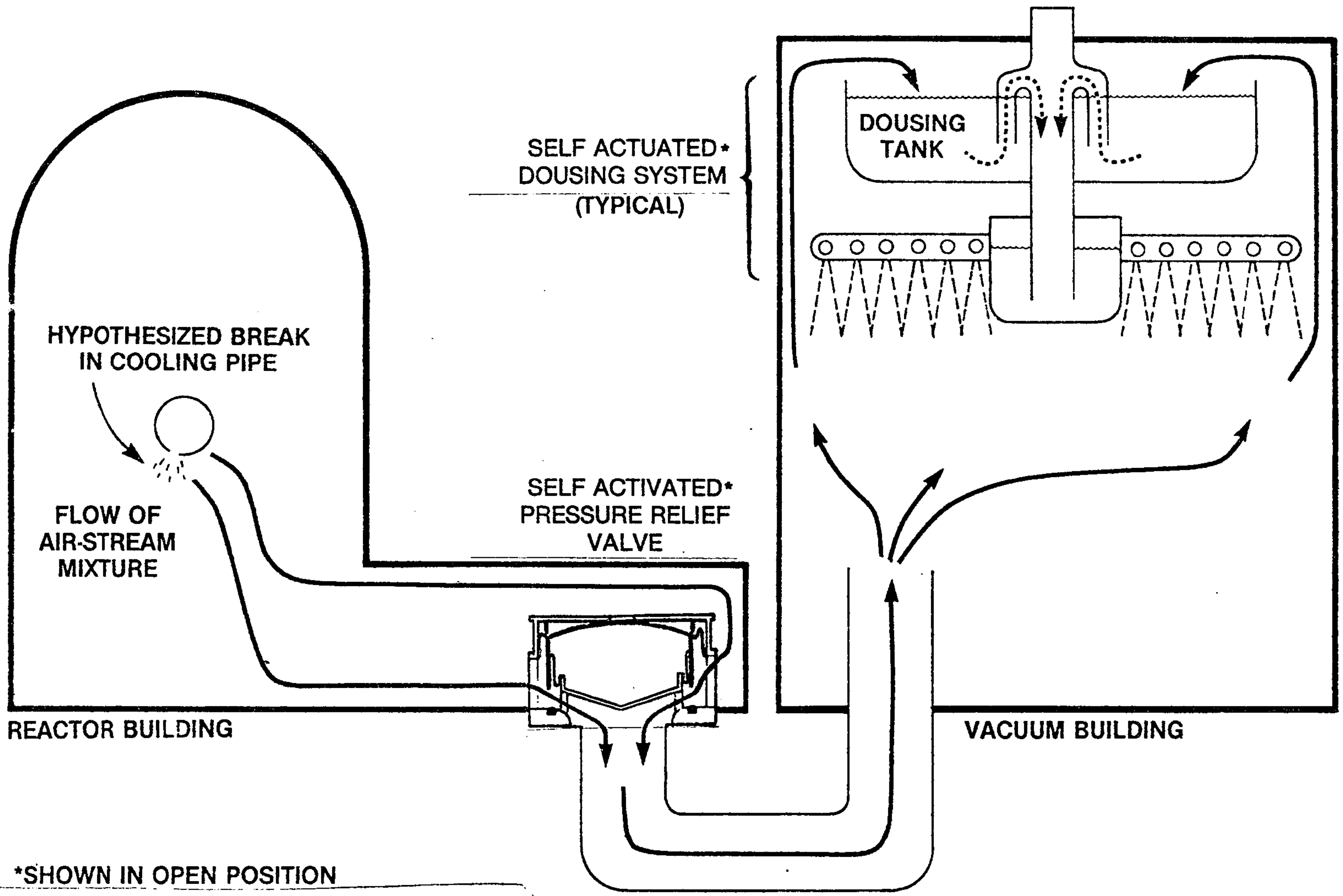
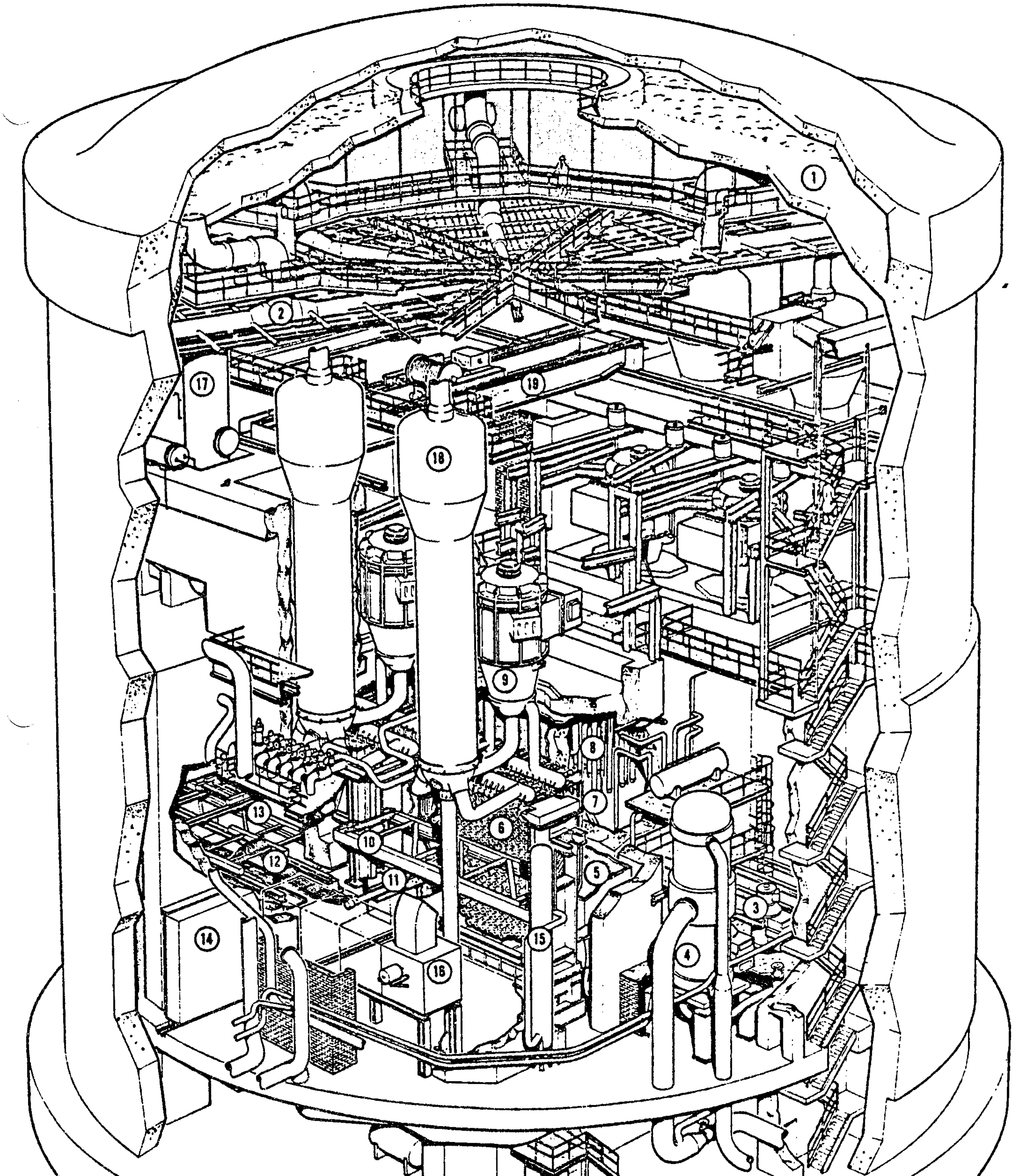


Figure 19.8 Design Dose Guidelines



*SHOWN IN OPEN POSITION

FIGURE 19.9 VACUUM CONTAINMENT SYSTEM SCHEMATIC



1 DOUSING WATER TANK
2 DOUSING WATER VALVES
3 MODERATOR PUMP
4 MODERATOR HEAT EXCHANGER
5 FEEDER CABINETS
6 REACTOR FACE
REACTOR
REACTIVITY MECHANISM
9 HEAT TRANSPORT SYSTEM PUMP
10 FUELLING MACHINE BRIDGE
11 FUELLING MACHINE CARRIAGE
12 FUELLING MACHINE CATENARY
13 FUELLING MACHINE MAINTENANCE LOCK
14 FUELLING MACHINE MAINTENANCE LOCK DOOR
15 END SHIELD COOLING WATER DELAY TANK
16 VAULT COOLER
17 PRESSURIZER
18 STEAM GENERATOR
19 STEAM GENERATOR ROOM CRANE

FIGURE 19.9a 600 MW REACTOR BUILDING CUTAWAY

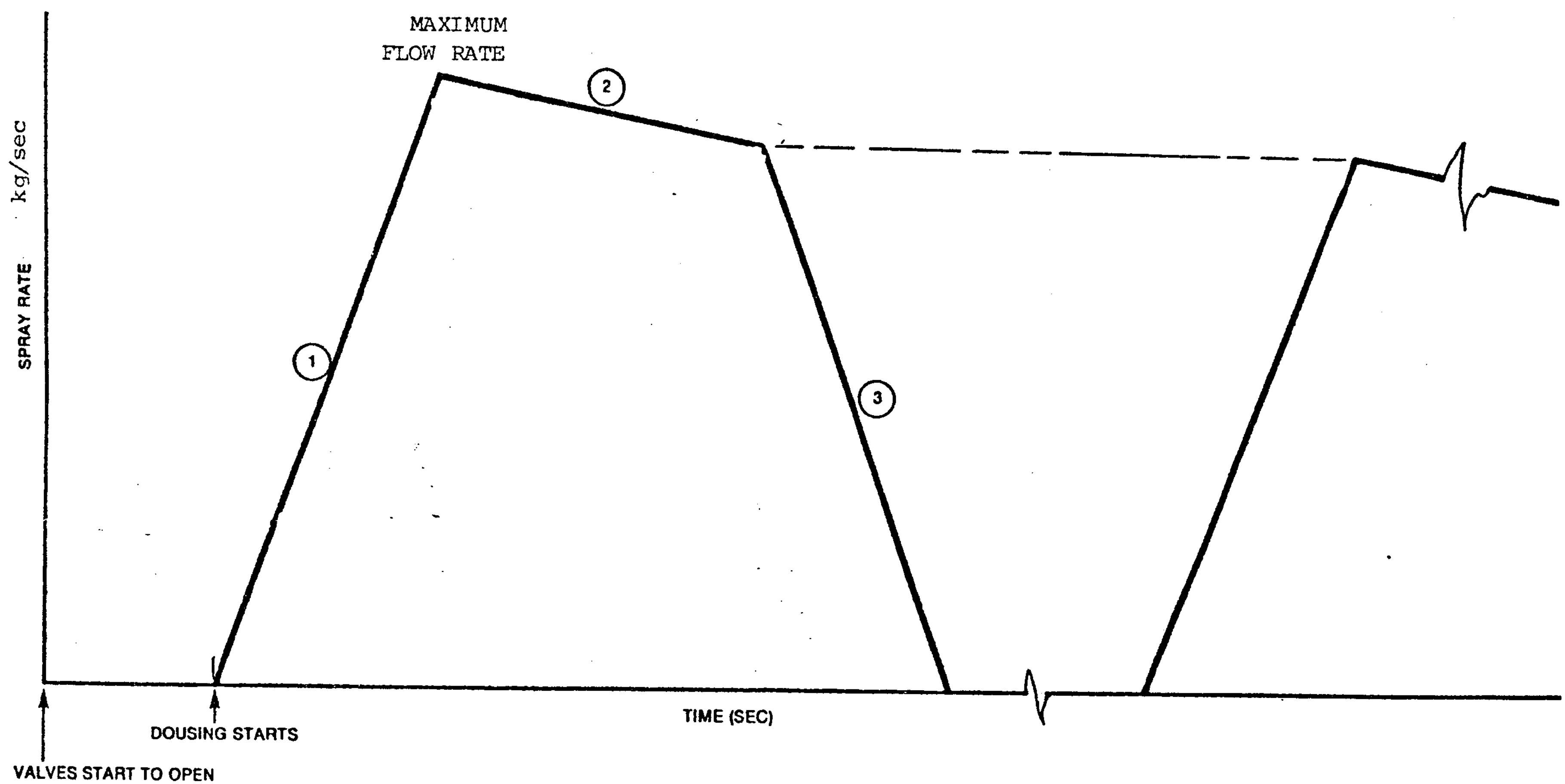


FIGURE 19.9b SPRAY RATE VS TIME

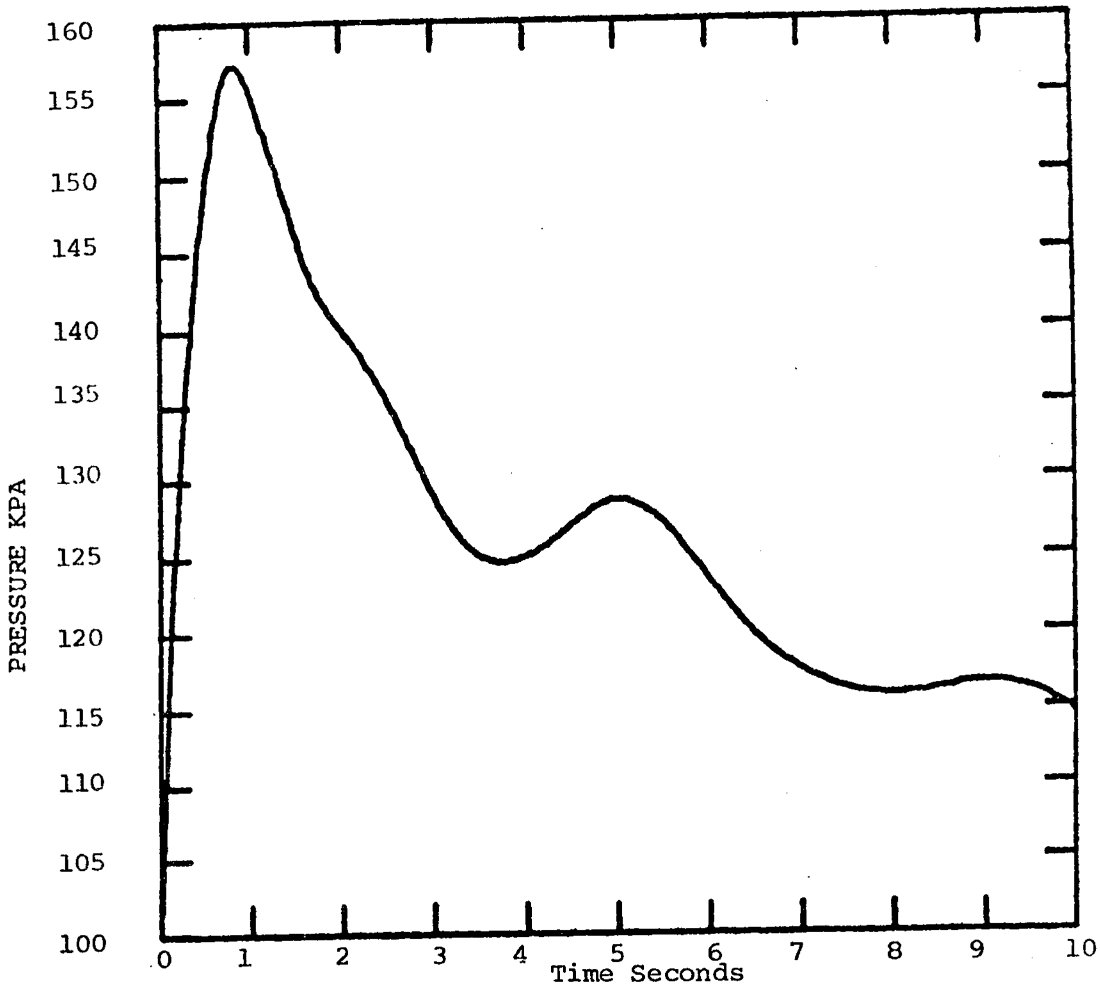


fig. 19.10 Typical Reactor Building Pressure Transient For Multiple Reactor Containment

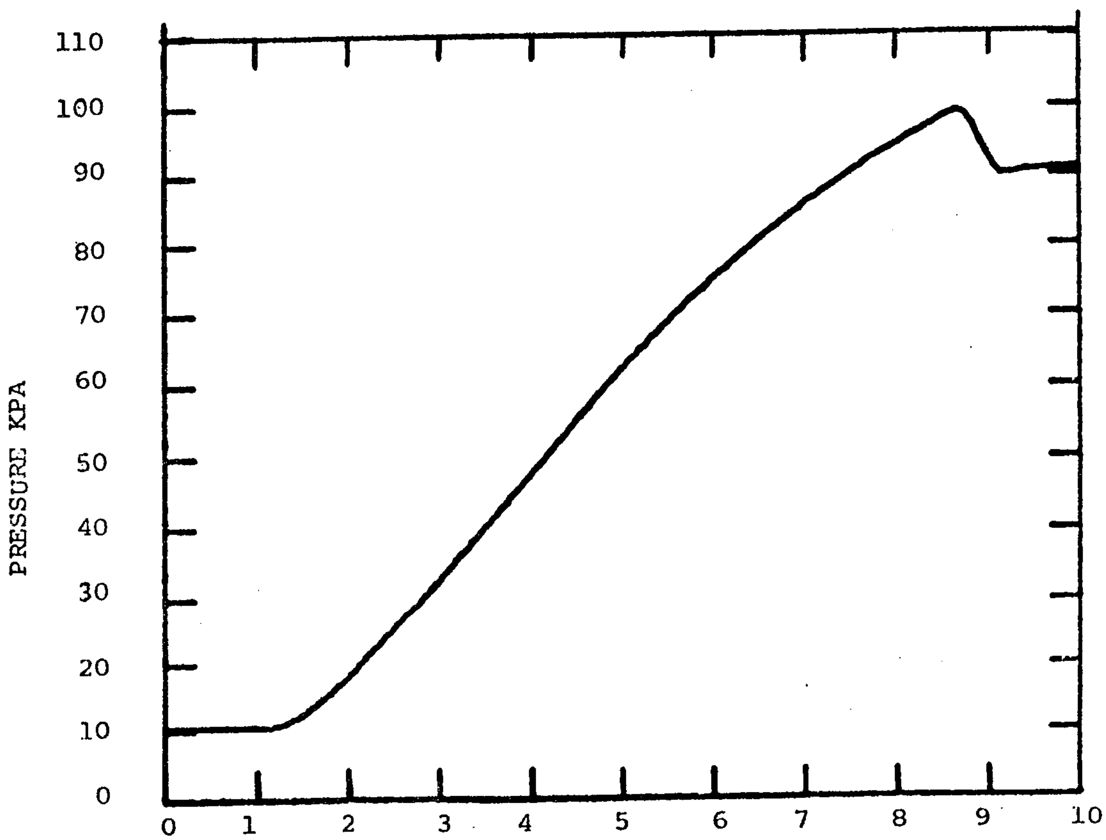


Fig. 19.11

Typical Vacuum Building Pressure Transient for

No page 29
See text for Fig 12

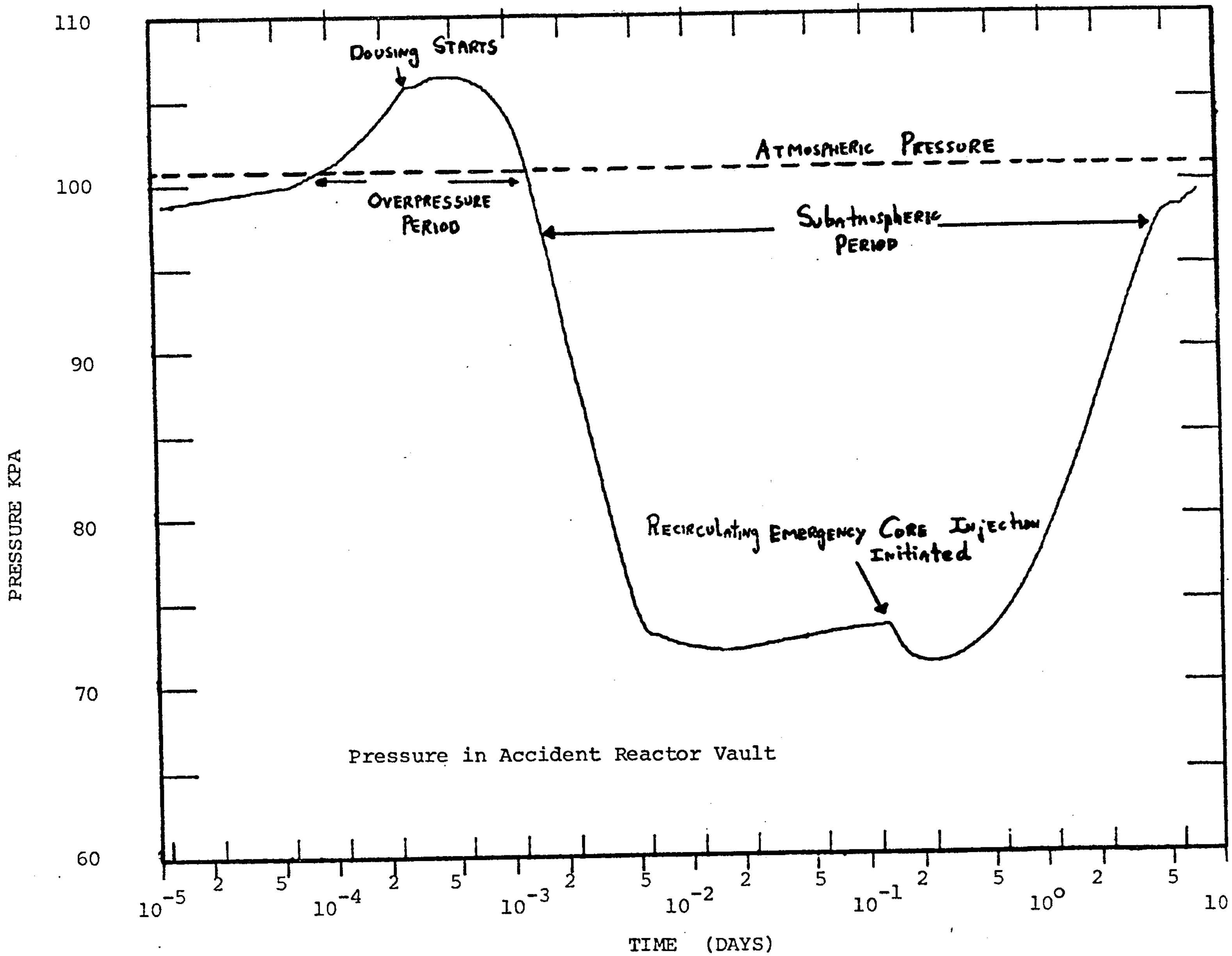


Fig 19.11a LONG TERM CONTAINMENT PRESSURE TRANSIENT

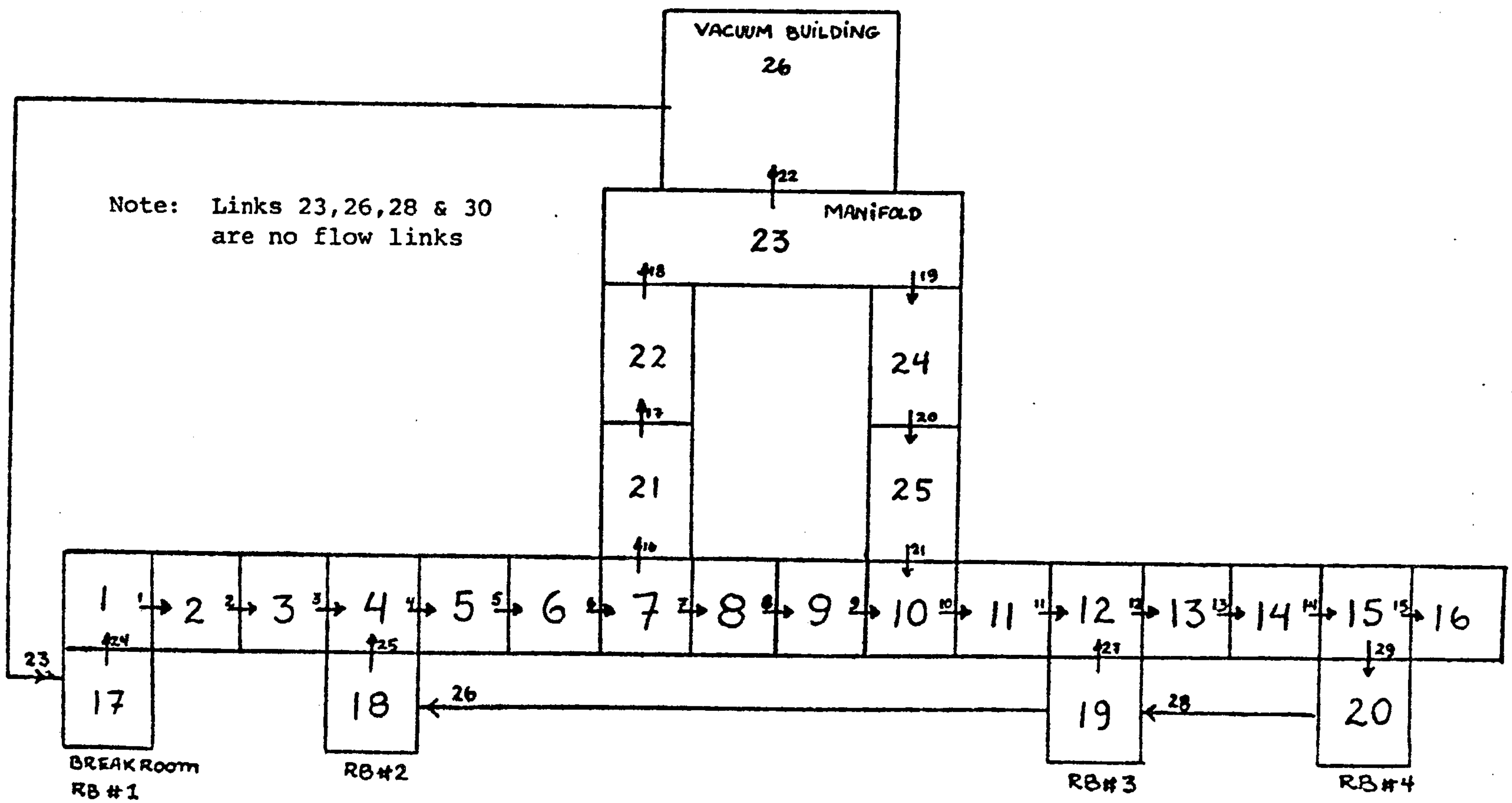


Fig. 19.13 Typical Node-Link Structure Used in PRESCON Simulation

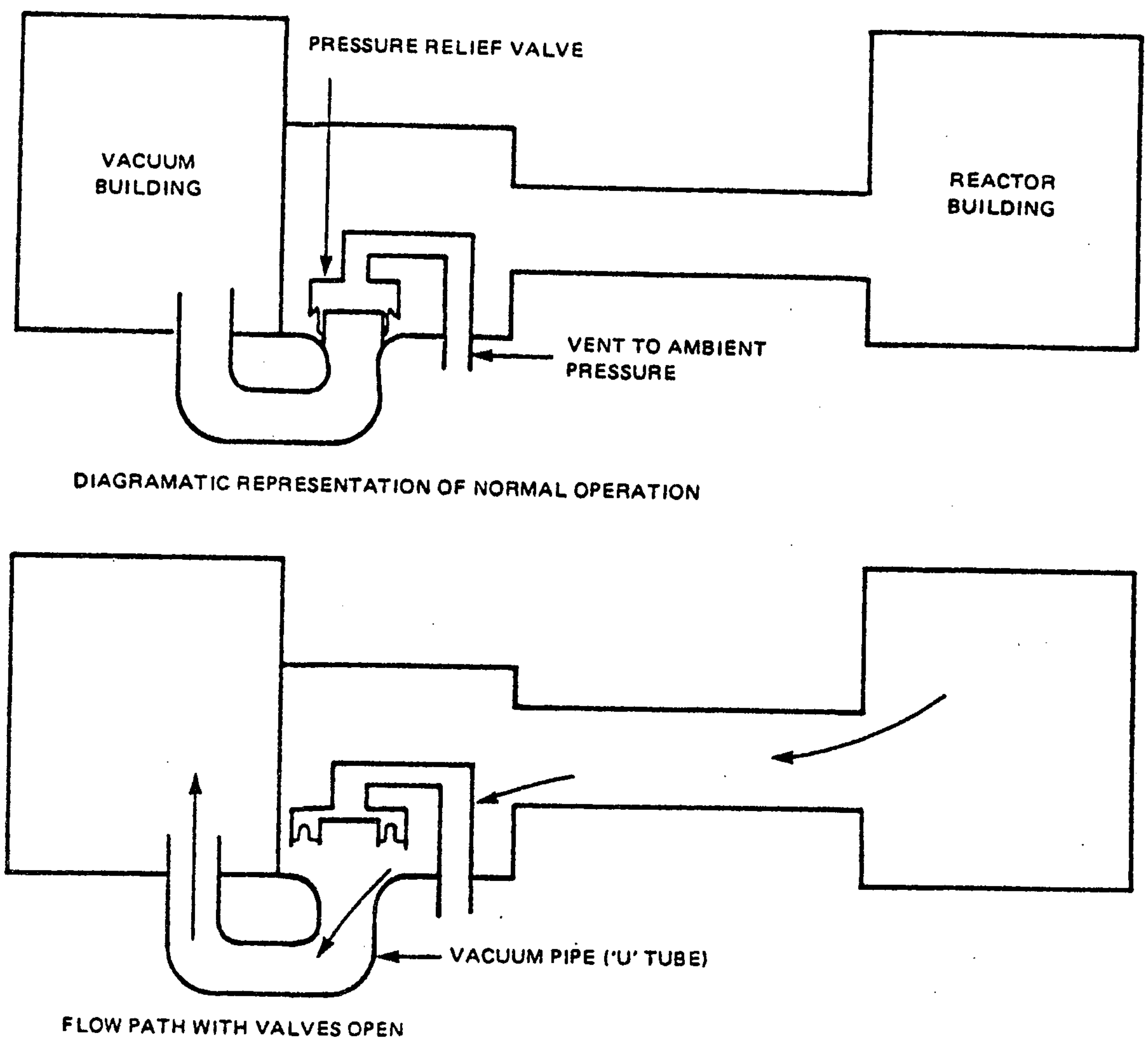


Fig. 19.14 Schematic of the Relief Systems

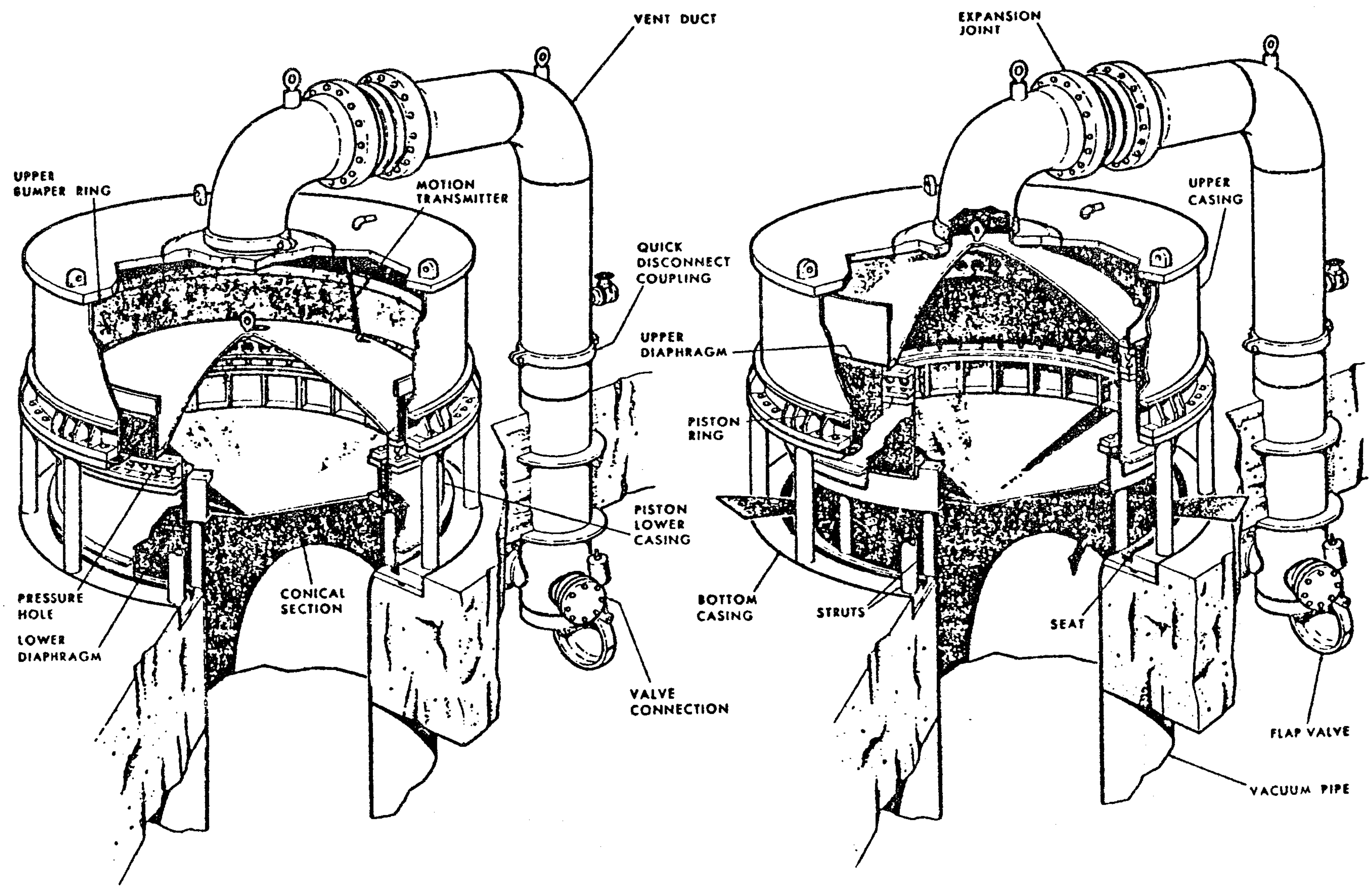


FIGURE 19.15 PRESSURE RELIEF VALVE

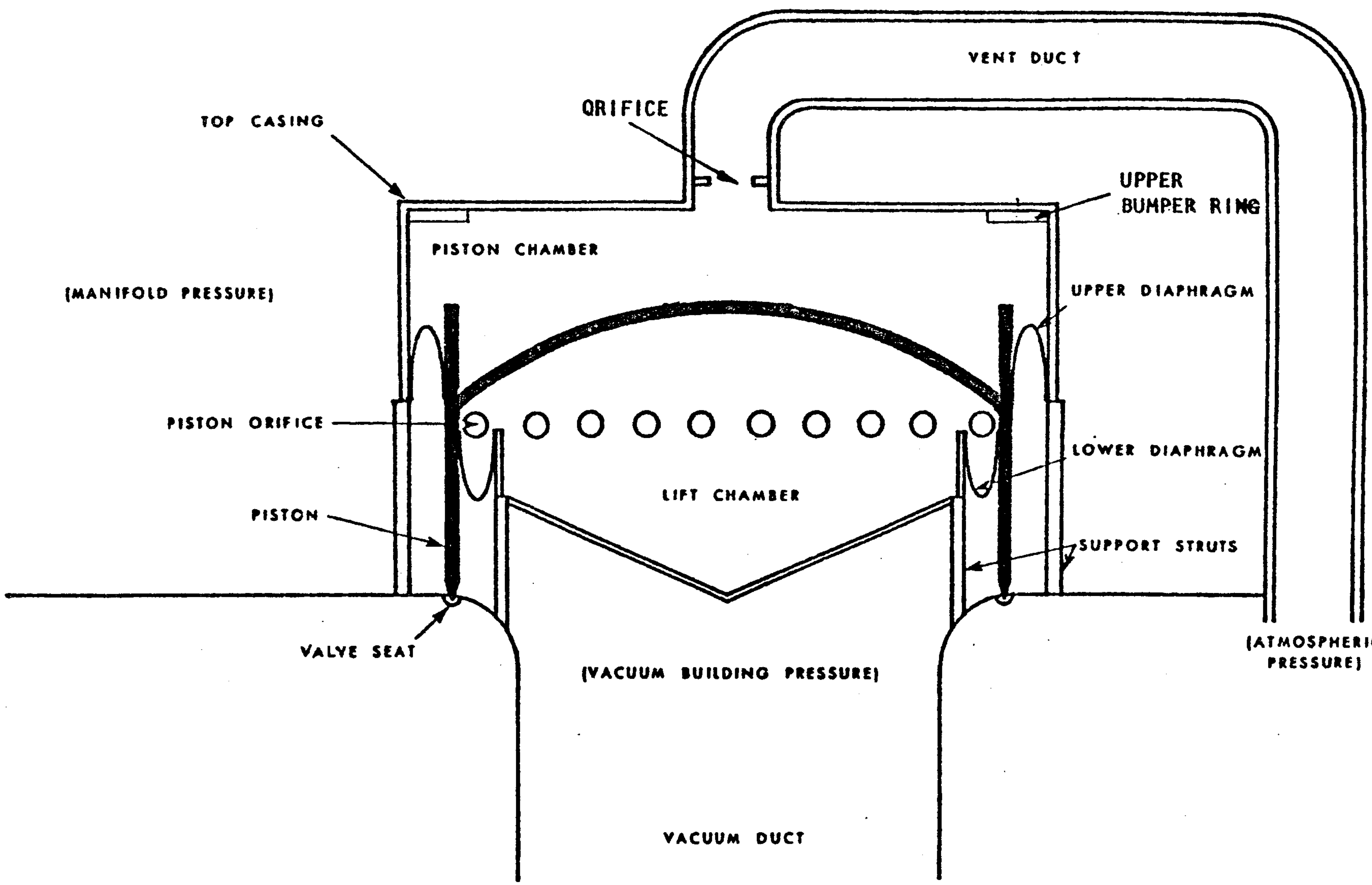
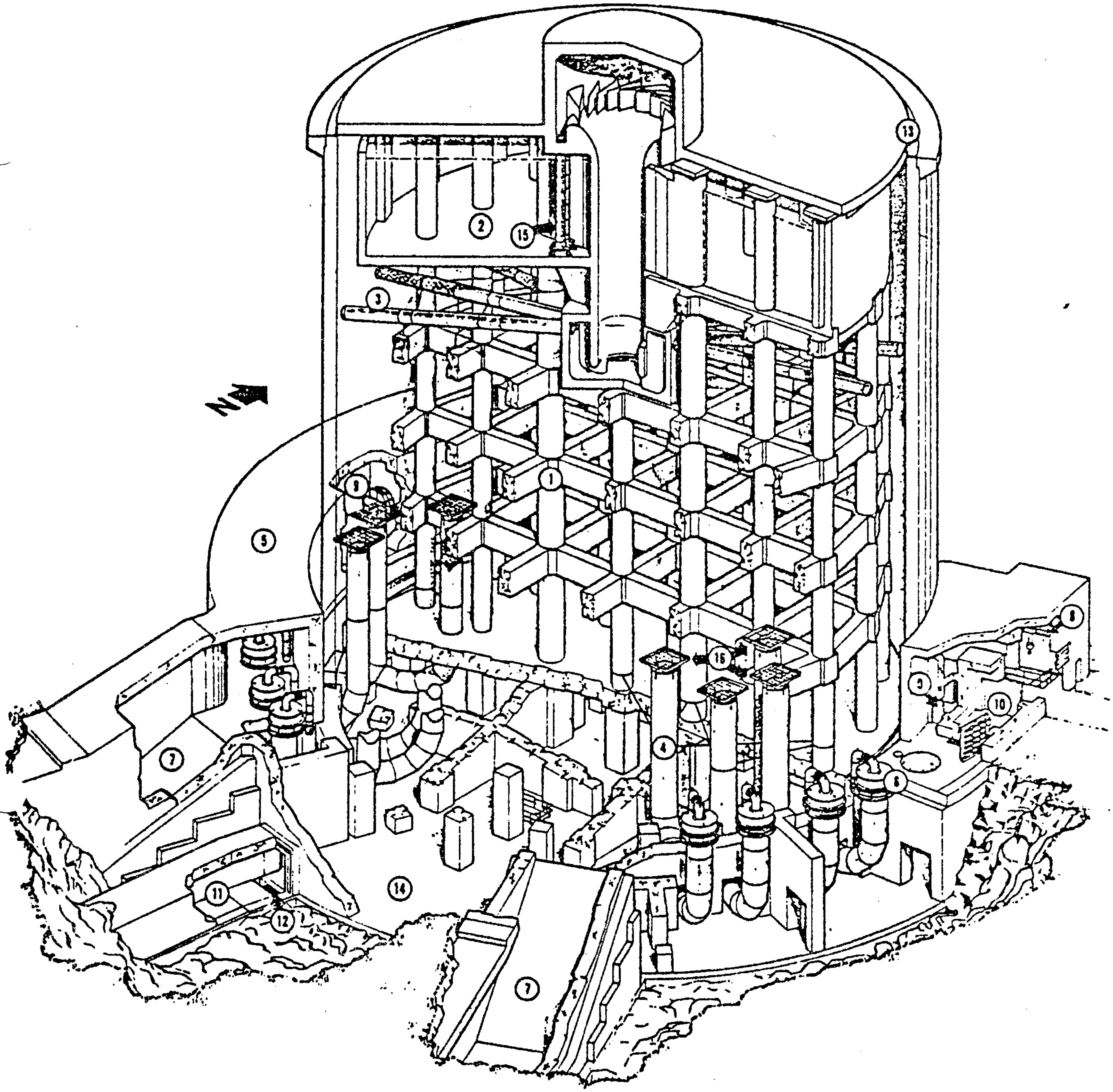


FIGURE 19.16 SCHEMATIC OF A PRESSURE RELIEF VALVE (CLOSED)



1 INTERNAL STRUCTURE
2 EMERGENCY WATER STORAGE TANK
3 DISTRIBUTION AND SPRAY HEADERS
4 VACUUM PIPE
5 VALVE MANIFOLD
6 PRESSURE RELIEF VALVE
7 PRESSURE RELIEF DUCT
8 MONORAIL AND HOIST
9 PERSONNEL AIRLOCK
10 EQUIPMENT AIRLOCK
11 SERVICE TUNNEL
12 CATCH BASIN
13 ROOF/WALL SEAL
14 BASEMENT
15 SUCTION PIPES
16 DIFFUSING SCREENS

FIGURE 19.17 CUTAWAY VIEW OF VACUUM BUILDING AND VALVE MANIFOLD

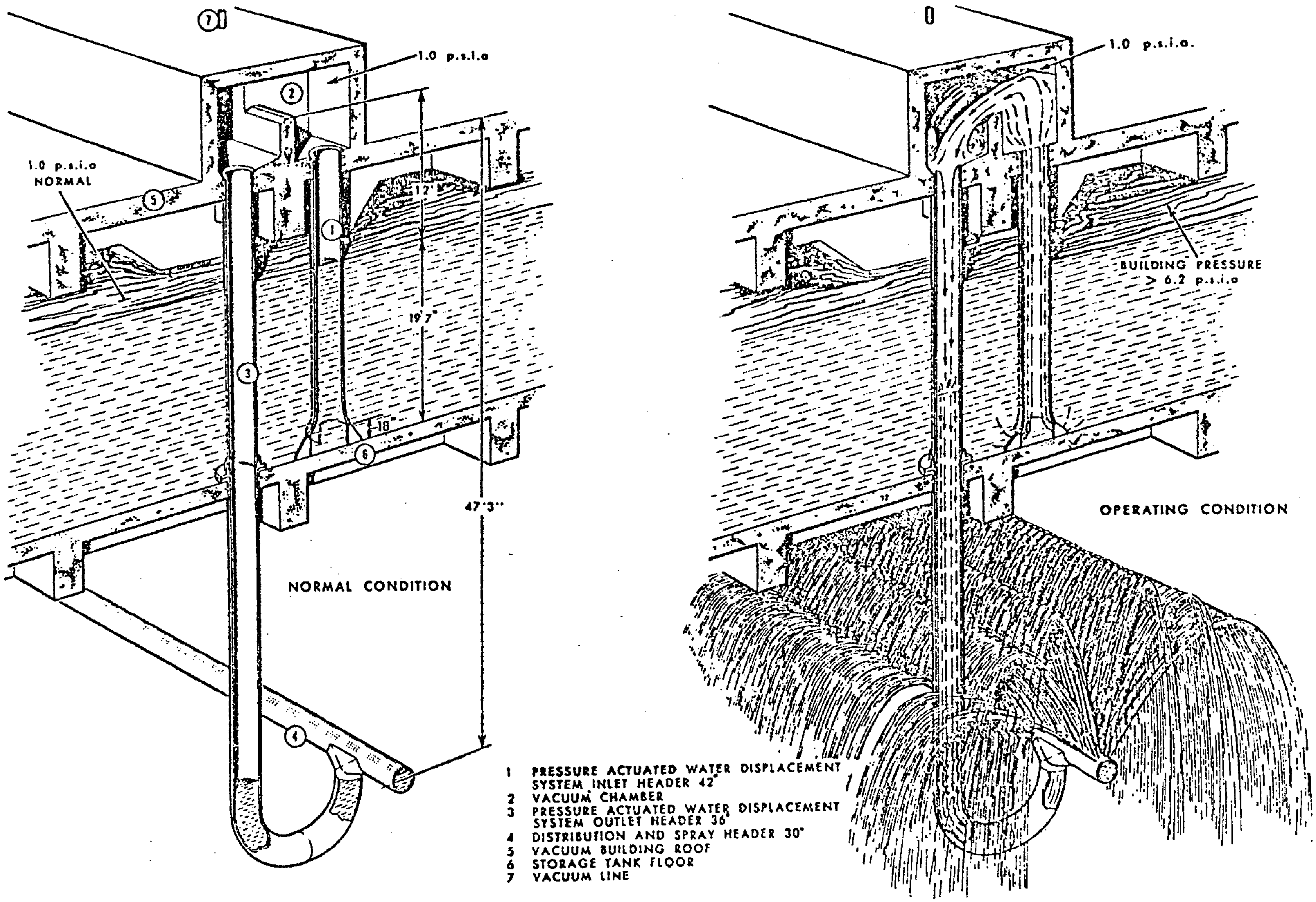


FIGURE 19.18 VACUUM BUILDING SPRAY SYSTEM

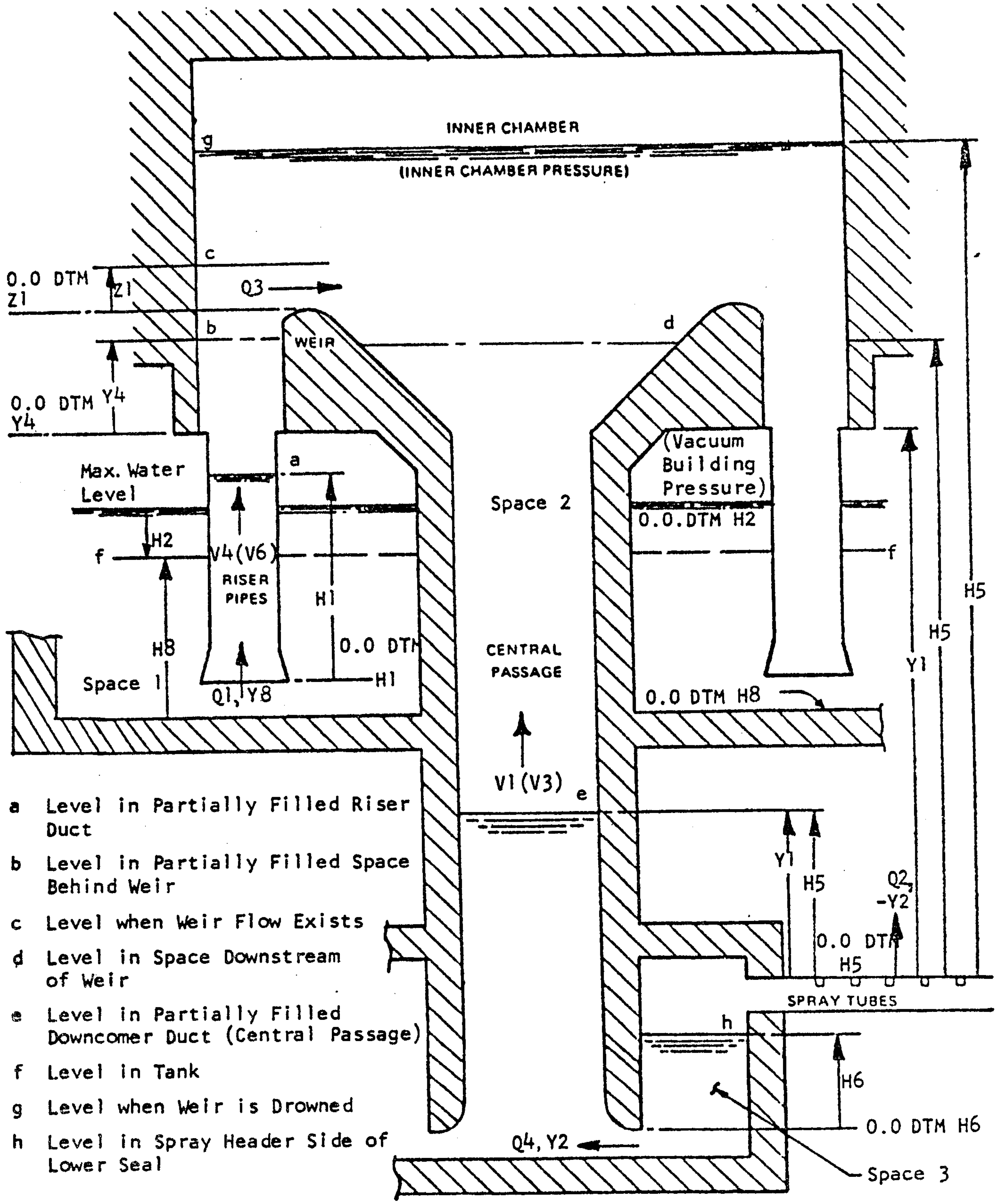
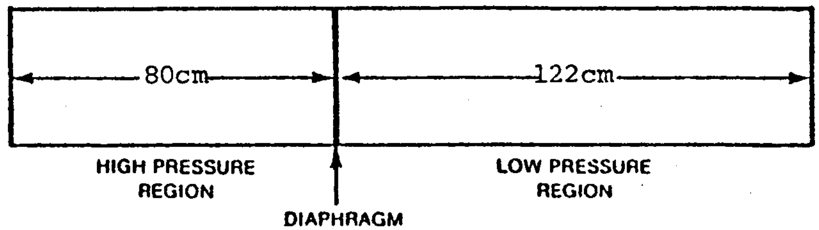


FIGURE 19.19 MODELLED HEIGHTS, VELOCITIES, FLOWS AND DEFINED SPACES

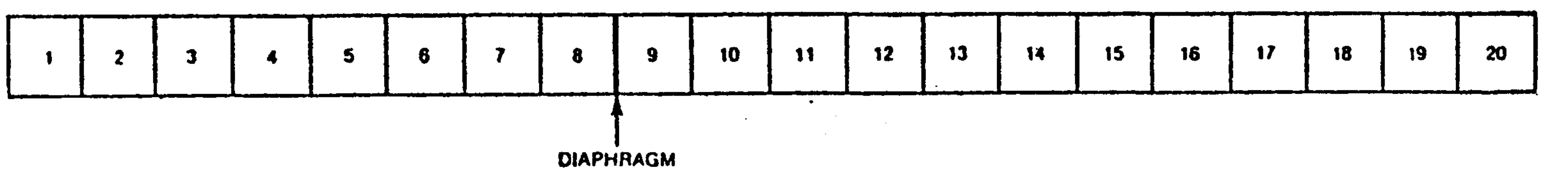


TOTAL LENGTH OF DUCT = 202cm

20 NODES OF LENGTH = 10cm

CROSS-SECTIONAL AREA = $57 cm^2$

VOLUME OF EACH NODE = $600 cm^3$



PRESSURE IN NODES 1 — 8 = 2 ATM

PRESSURE IN NODES 9 — 20 = 1 ATM

FIGURE 19.20 SHOCK TUBE GEOMETRY AND NODE STRUCTURE

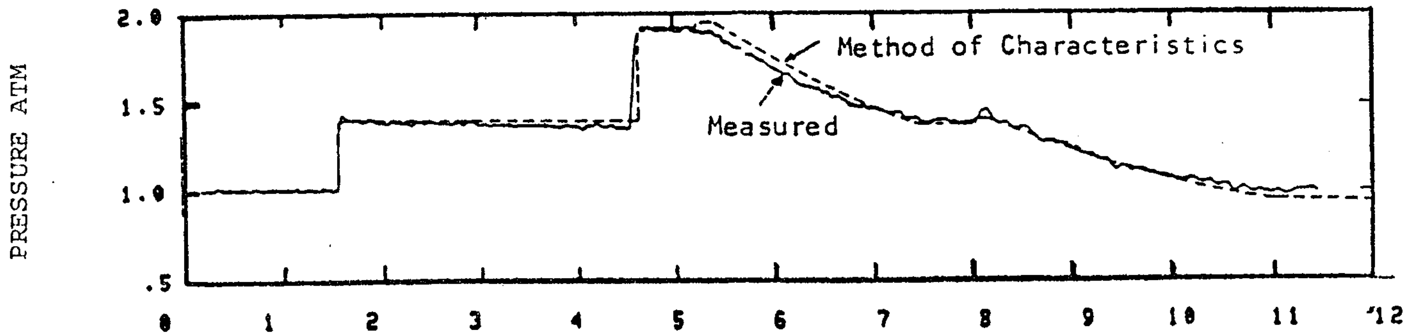


Fig. 19.21 SHOCK TUBE PRESSURE TRANSIENT (EXPERIMENTAL) TIME-MILLISECONDS

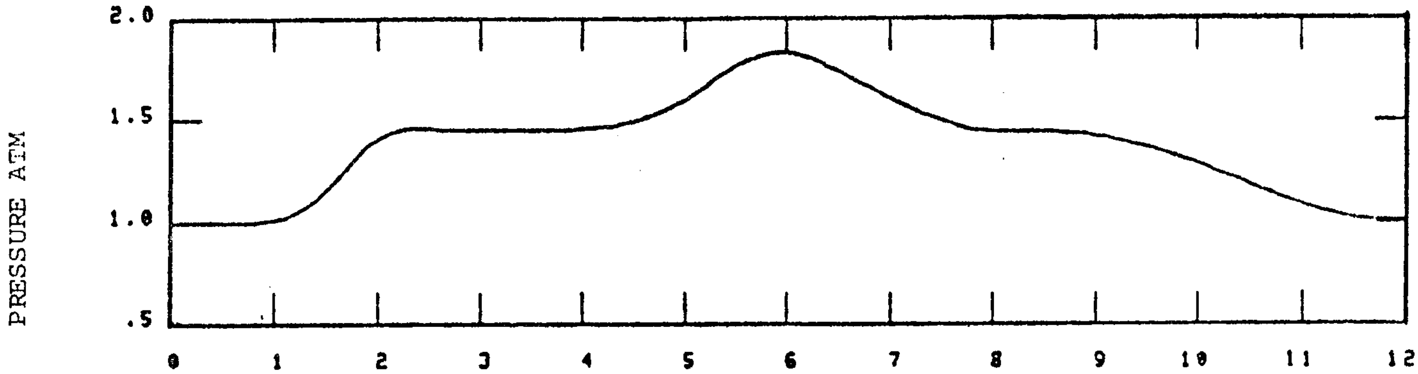


Fig. 19.22 SHOCK TUBE PRESSURE TRANSIENT (PRESCON2) TIME-MILLISECONDS

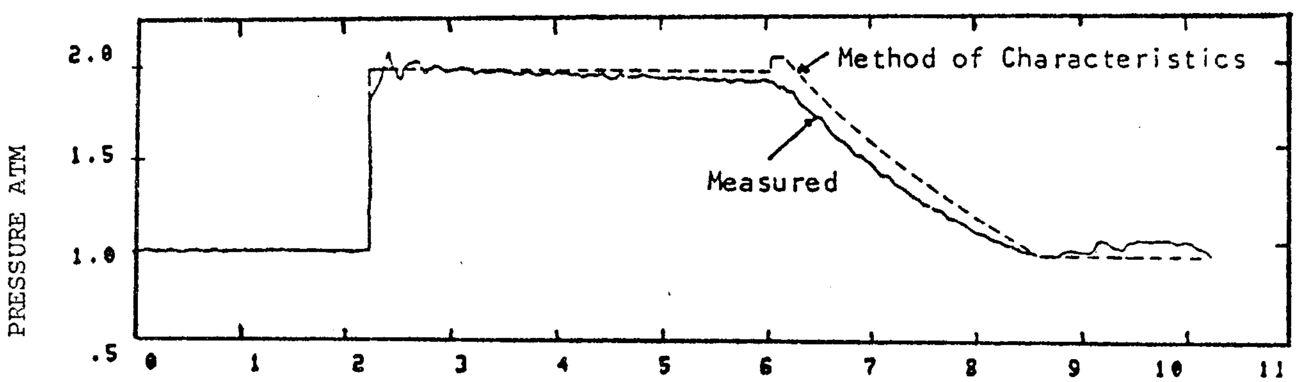


Fig. 19.23 SHOCK TUBE PRESSURE TRANSIENT (EXPERIMENTAL) TIME-MILLISECONDS

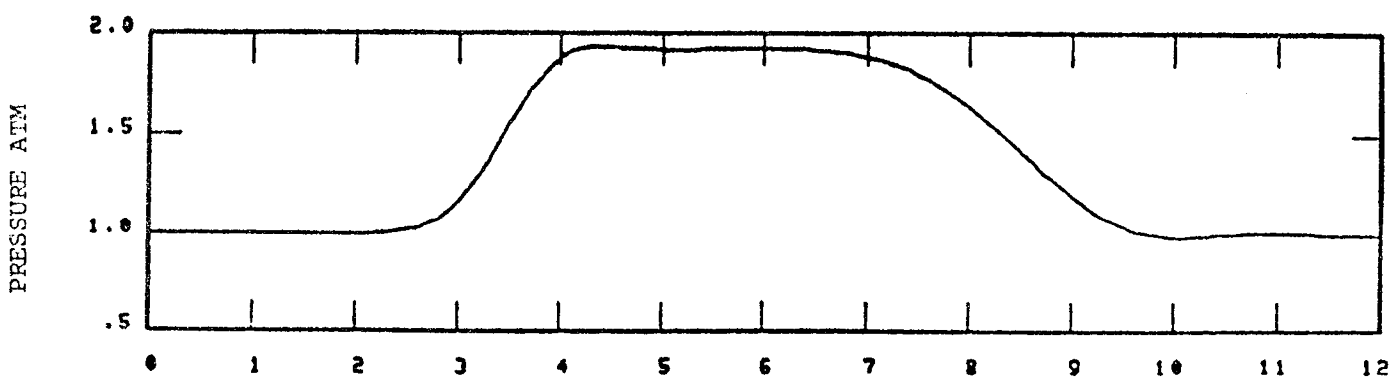
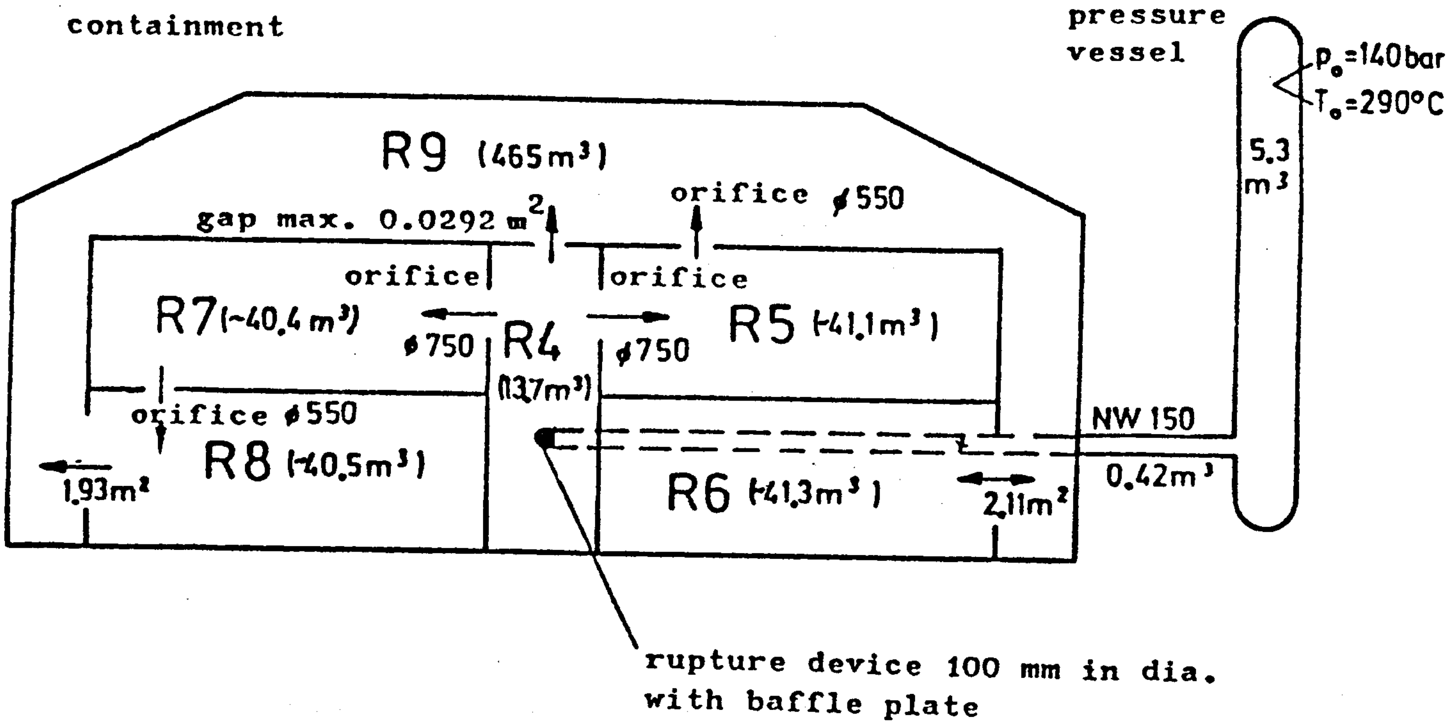


Fig. 19.24 SHOCK TUBE PRESSURE TRANSIENT (PRESCON2) TIME-MILLISECONDS



Ø = diameter (in mm)

Fig. 19.25 CASP2 configuration

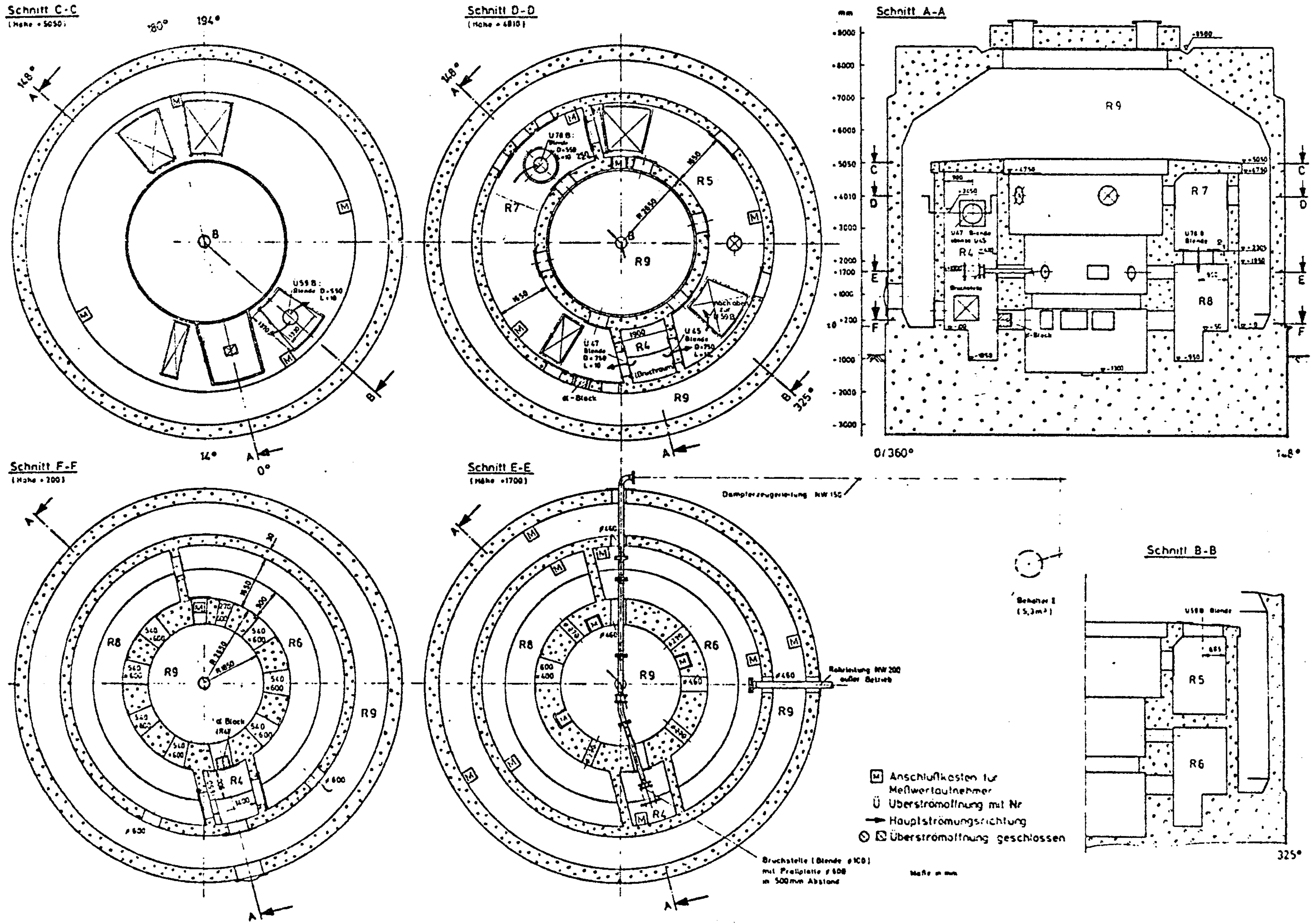


Fig. 19.26 Model Containment in Experiment D16/CASP2

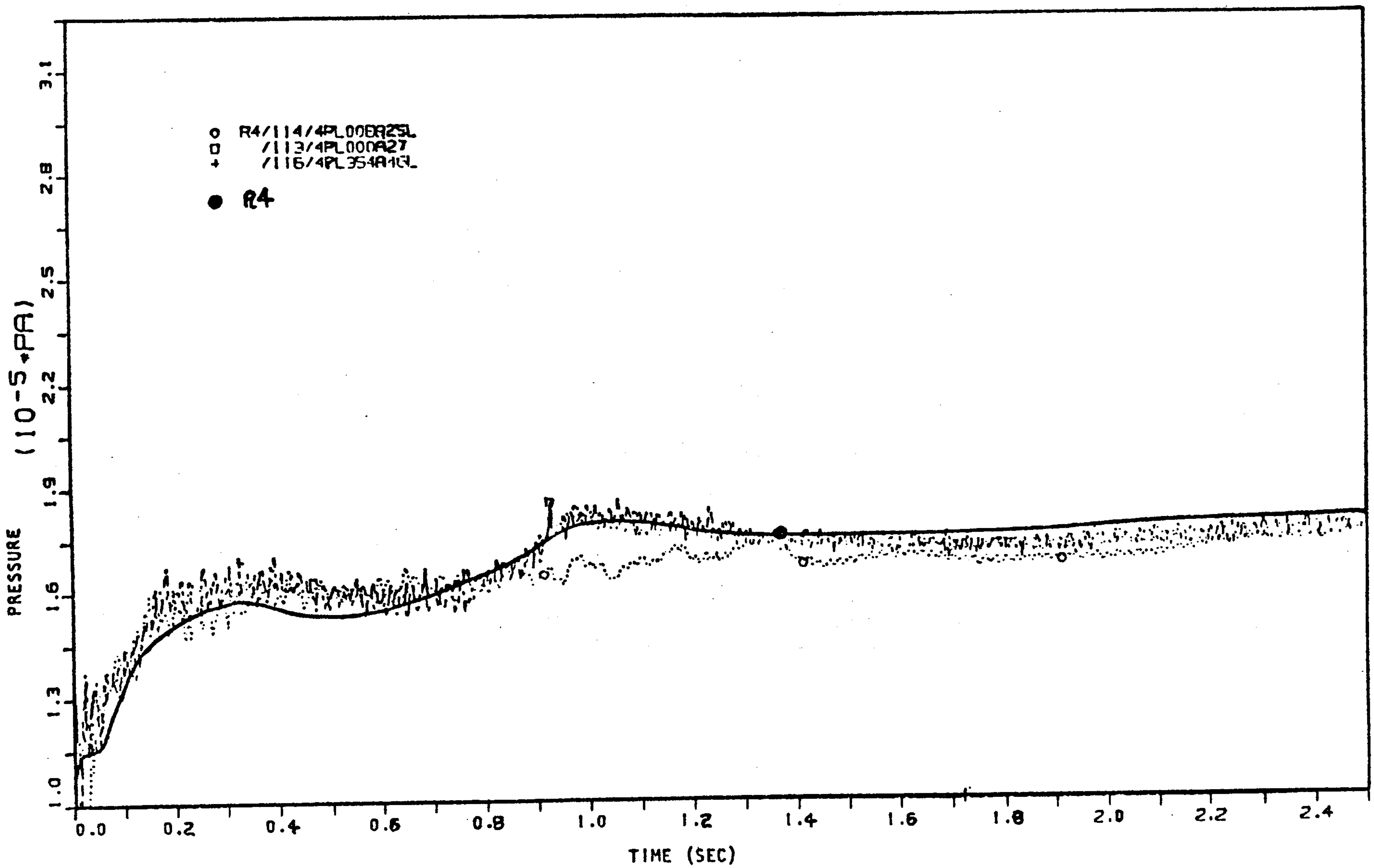


FIGURE 19.27 SHORT TERM PRESSURE HISTORIES

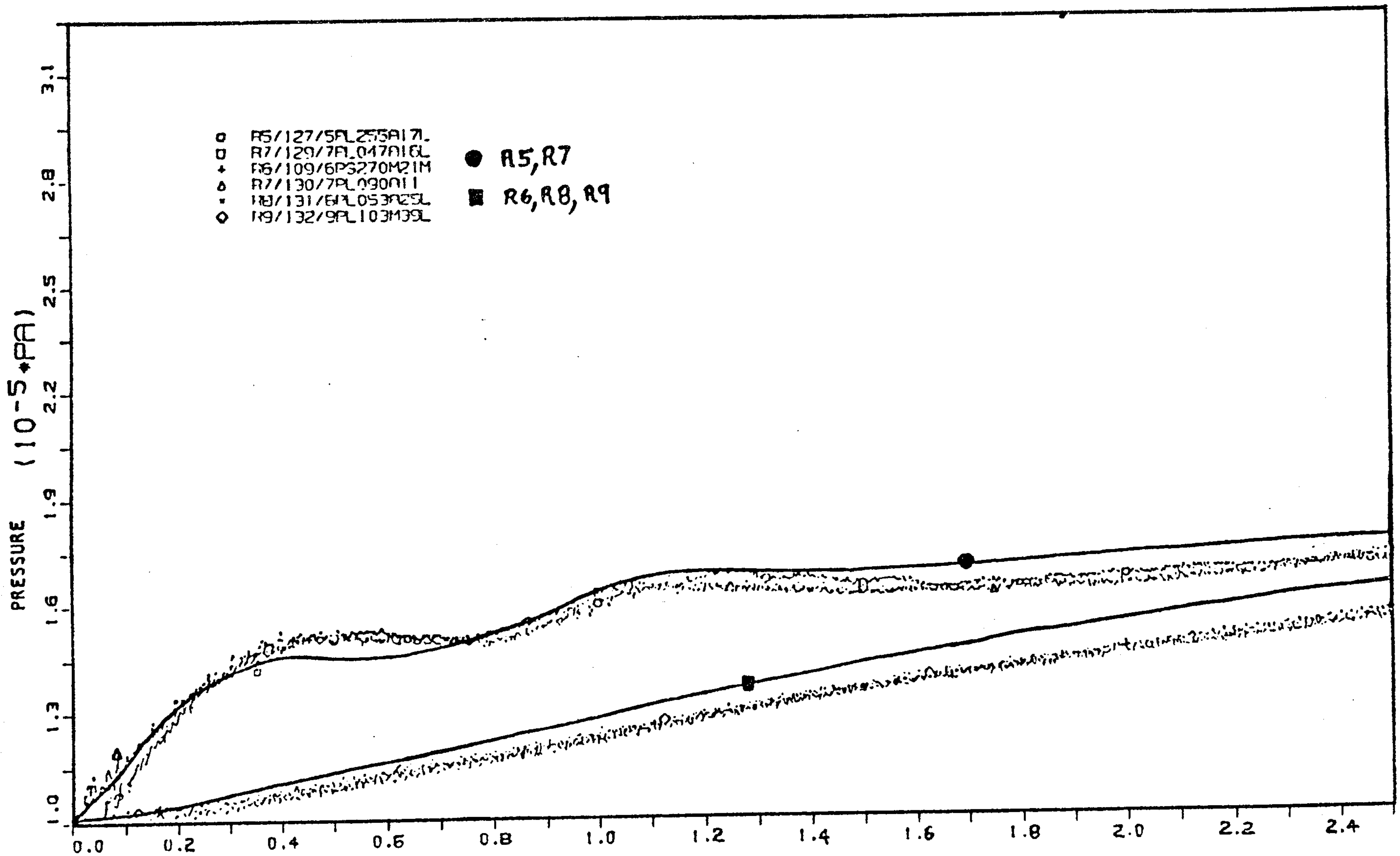


FIGURE 19.28 SHORT TERM PRESSURE HISTORIES

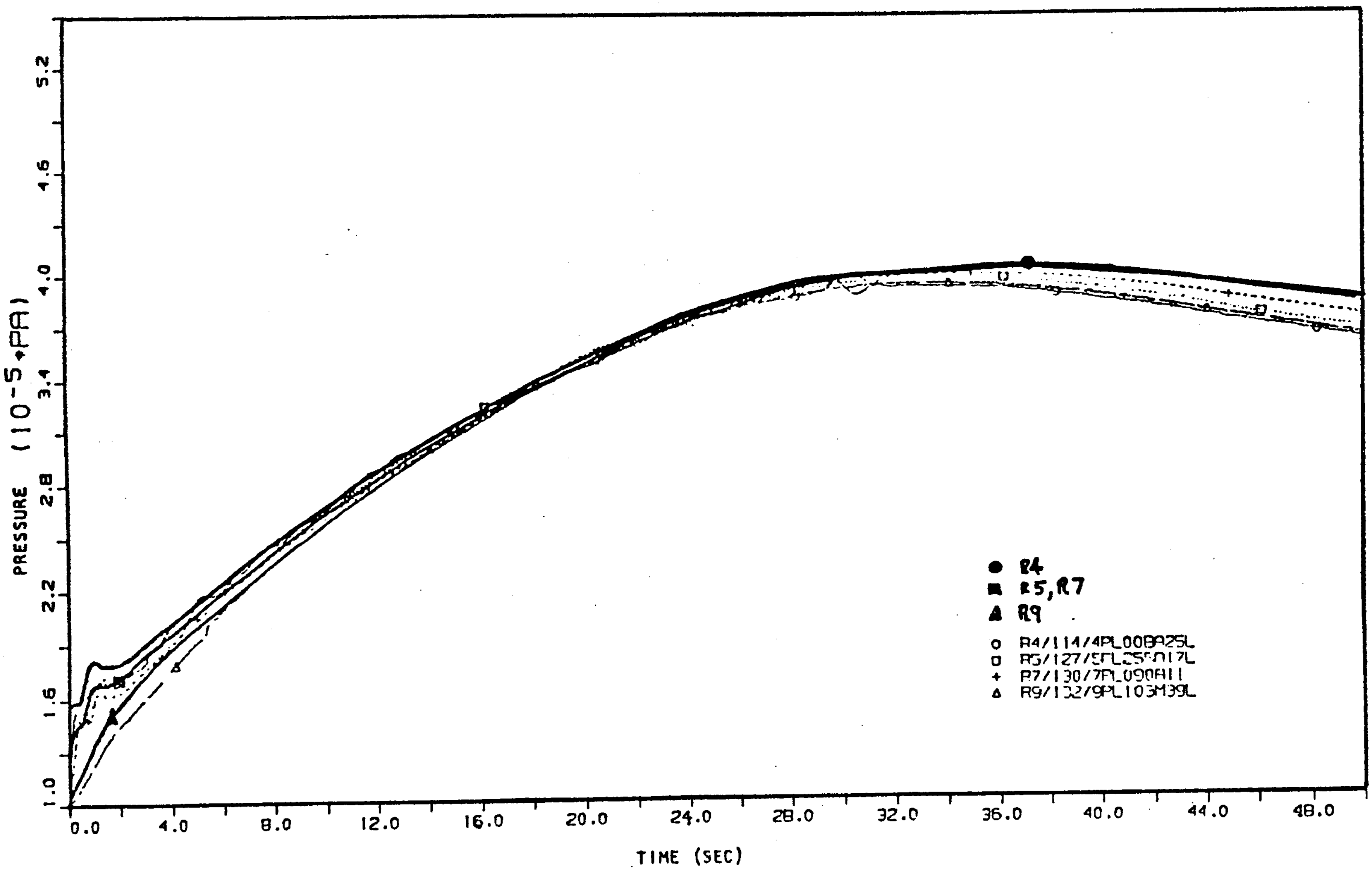


FIGURE 19.29 MEDIUM TERM PRESSURE HISTORIES

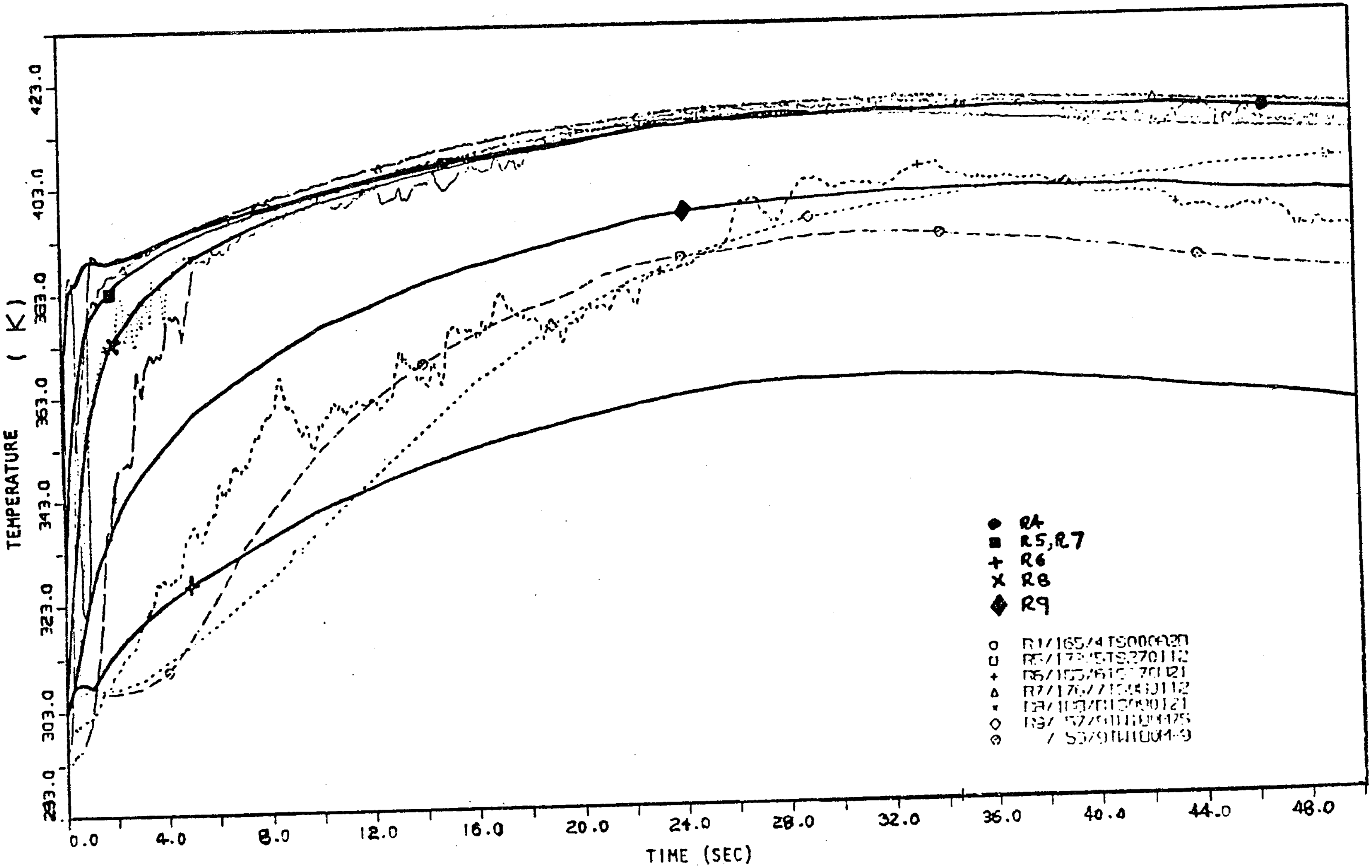


FIGURE 19.30 MEDIUM TERM TEMPERATURE HISTORIES

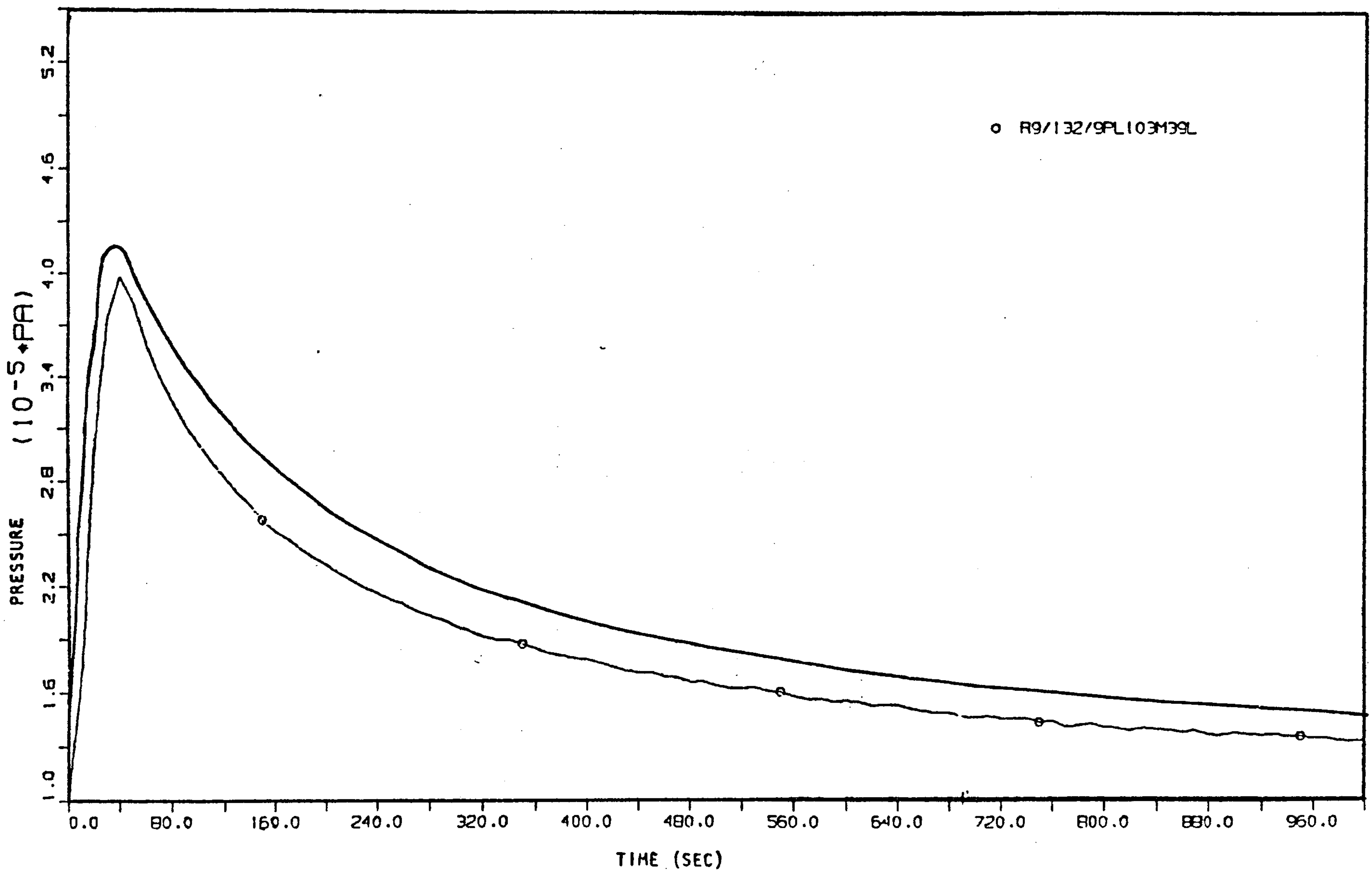


FIGURE 19.31 CONTAINMENT PRESSURE

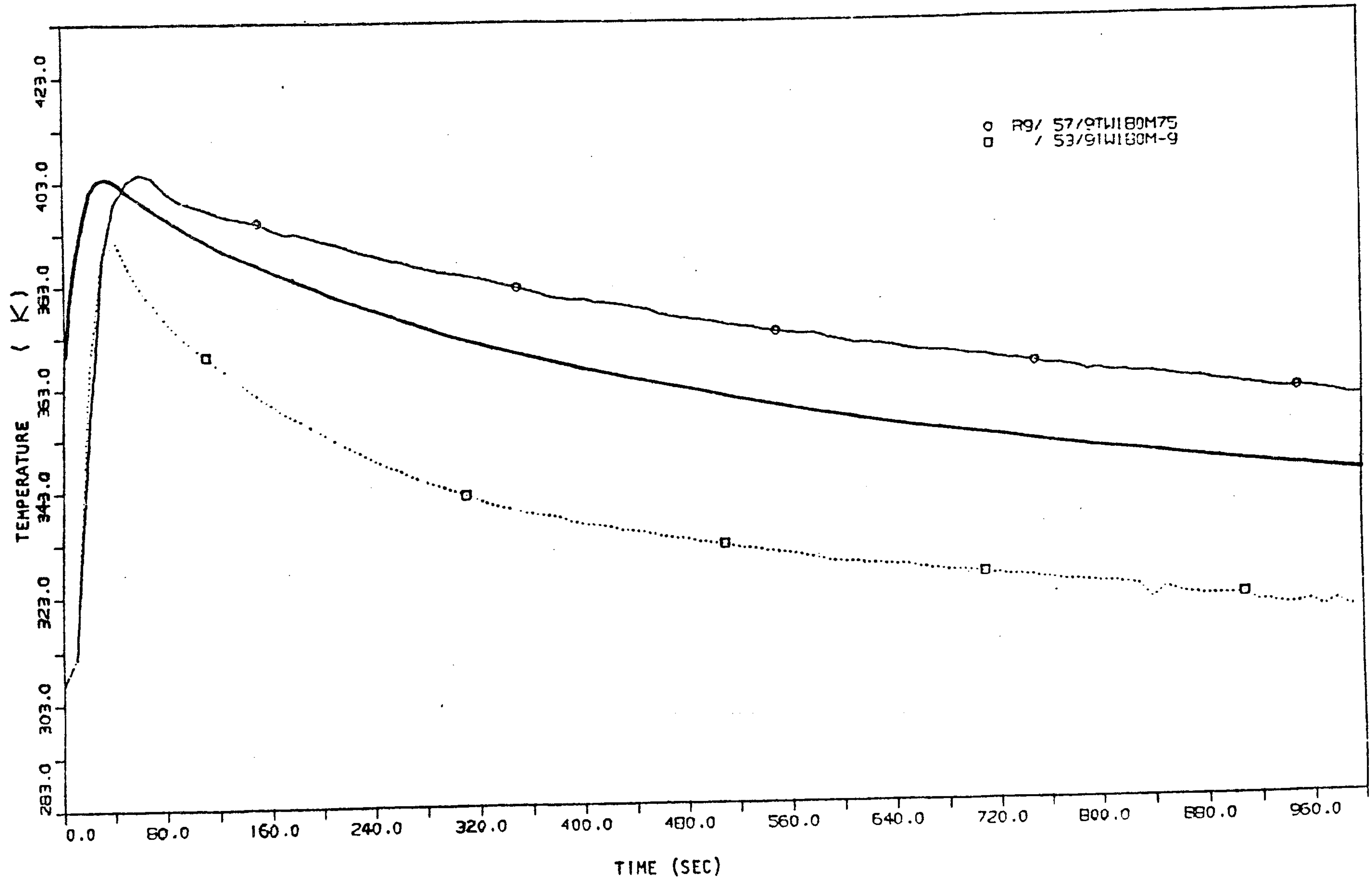


FIGURE 19.32 CONTAINMENT TEMPERATURE

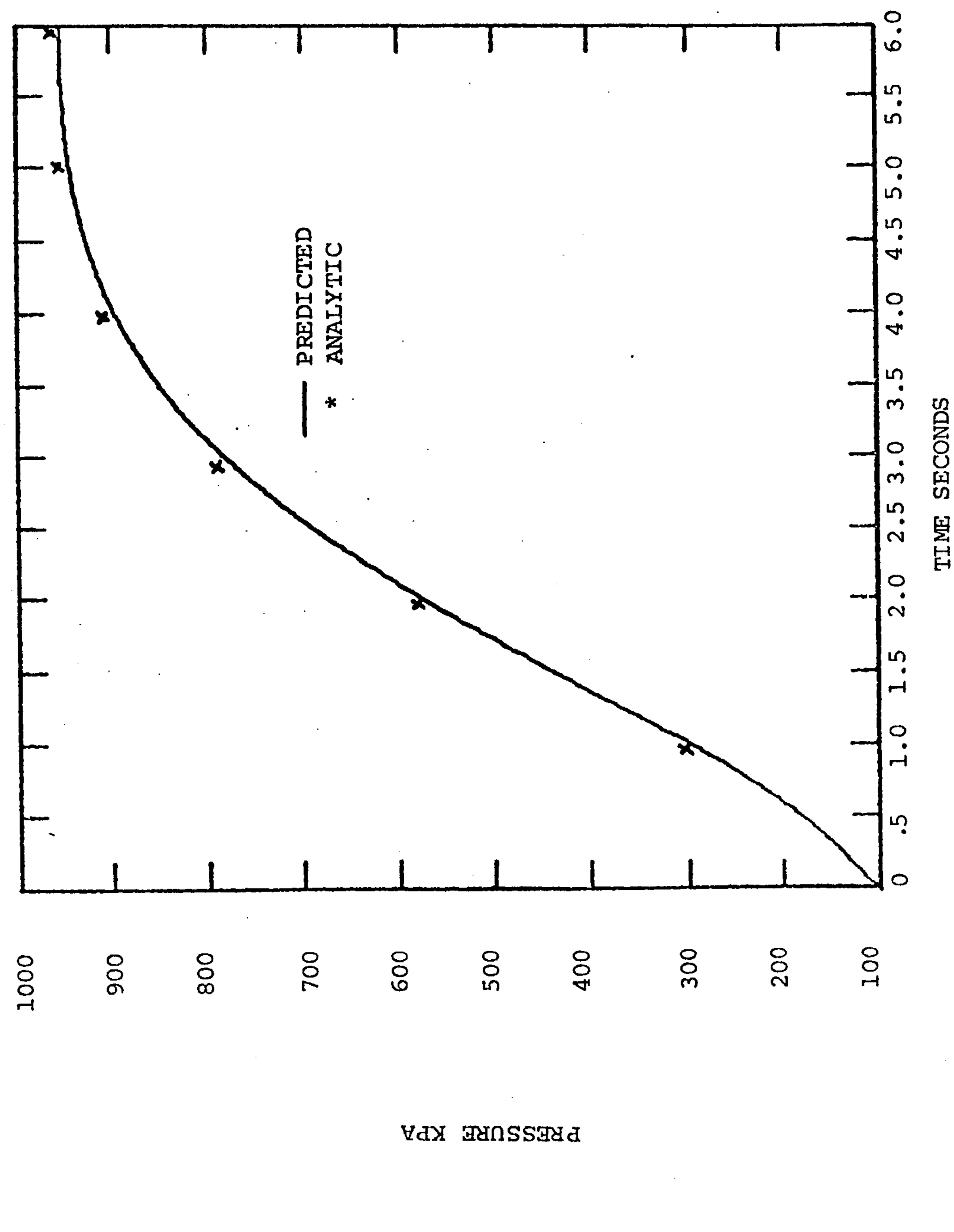


Fig 19.33 CONTAINMENT PRESSURE

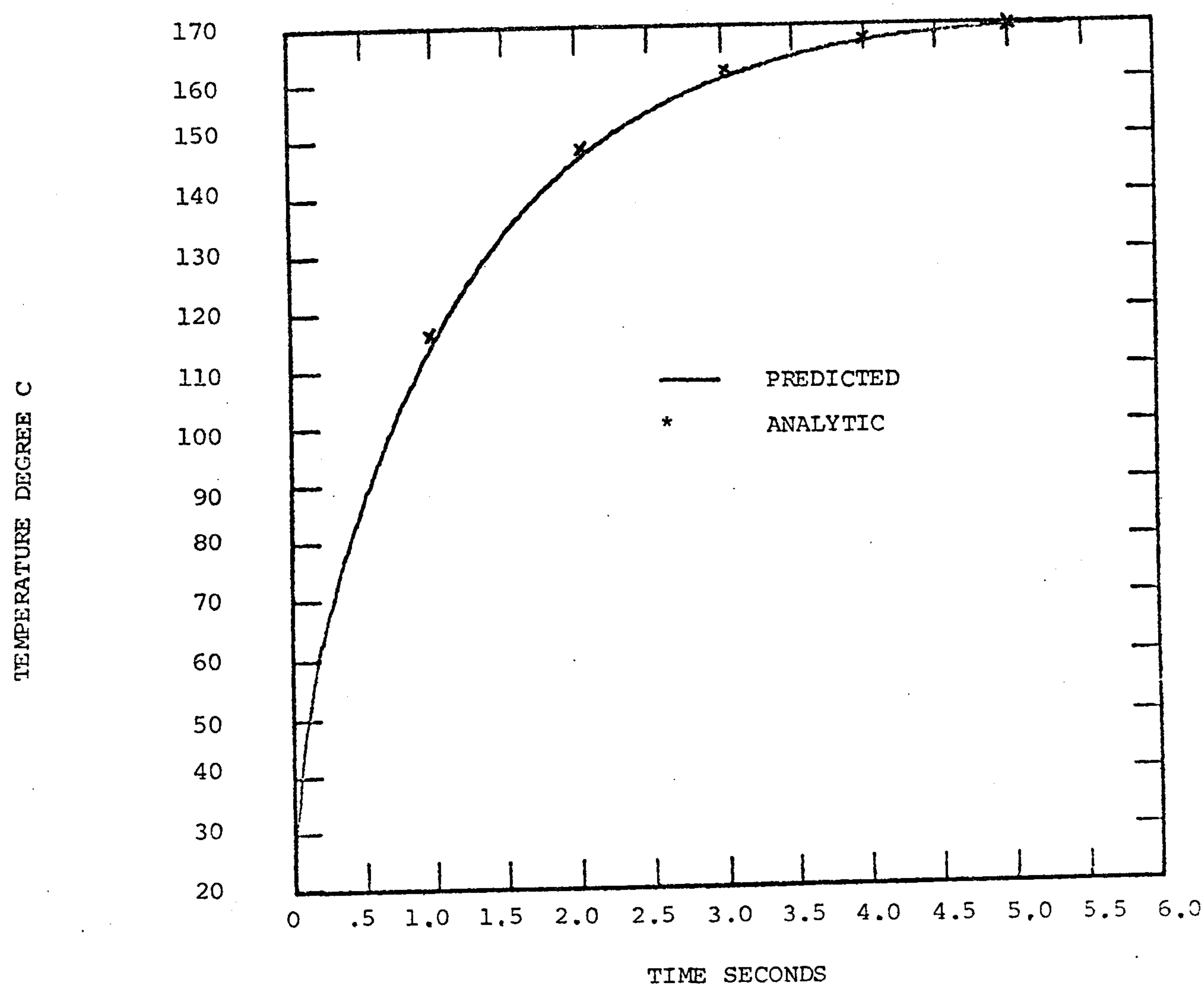


Fig 19.34 CONTAINMENT TEMPERATURE

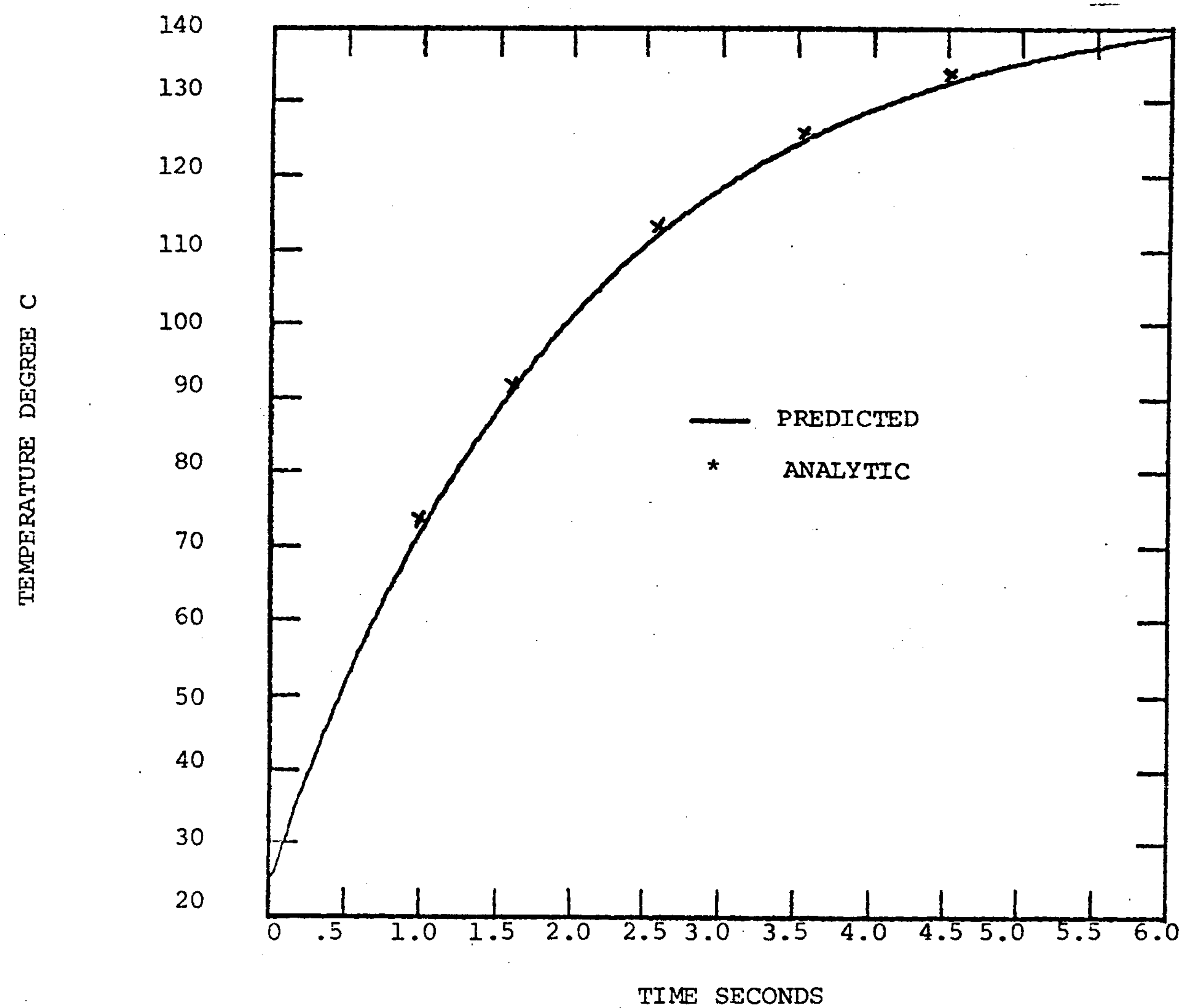


Fig 19.35 WALL SURFACE TEMPERATURE

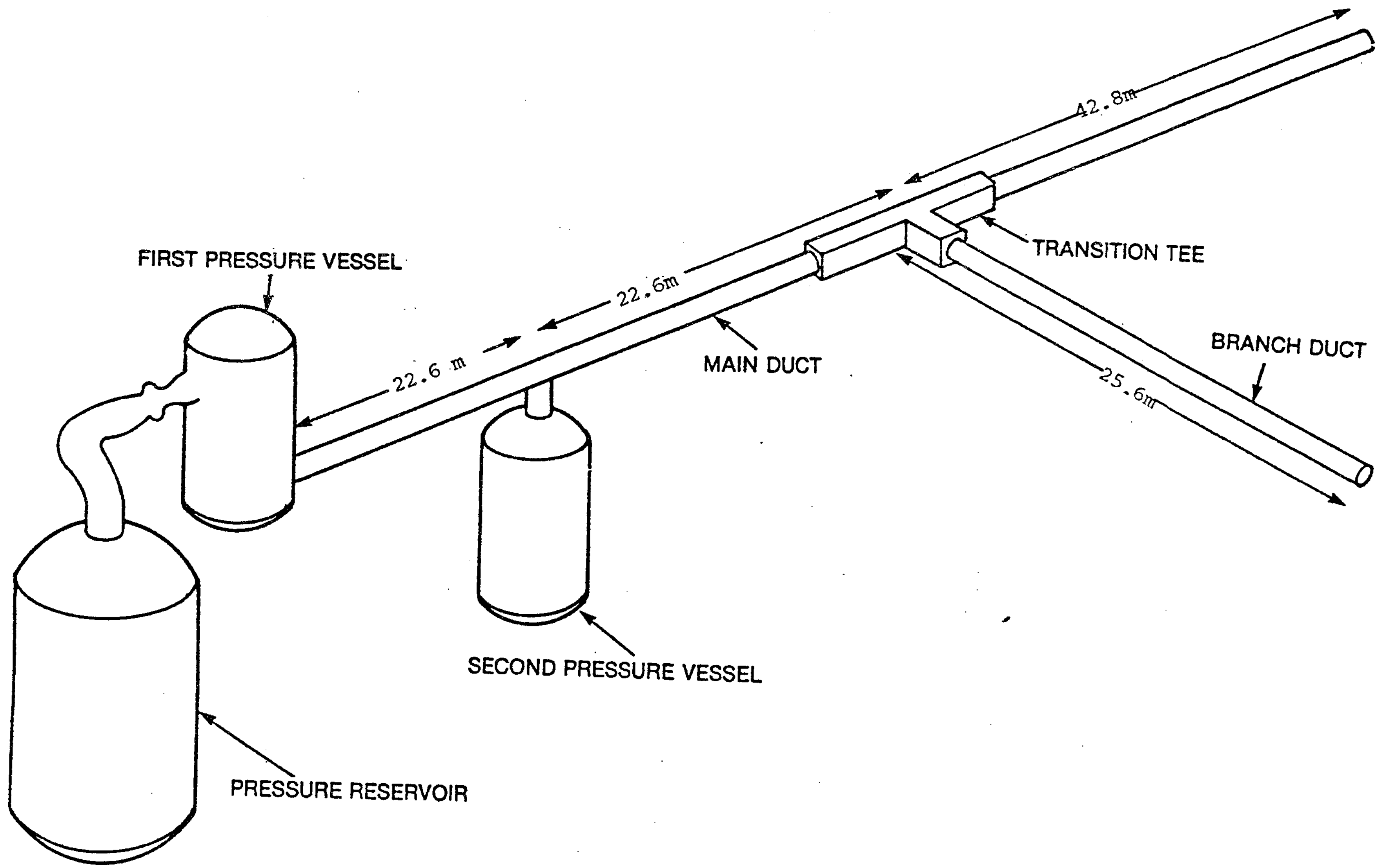


FIGURE 19.36: **TRANSIENT TEST RIG GEOMETRY**

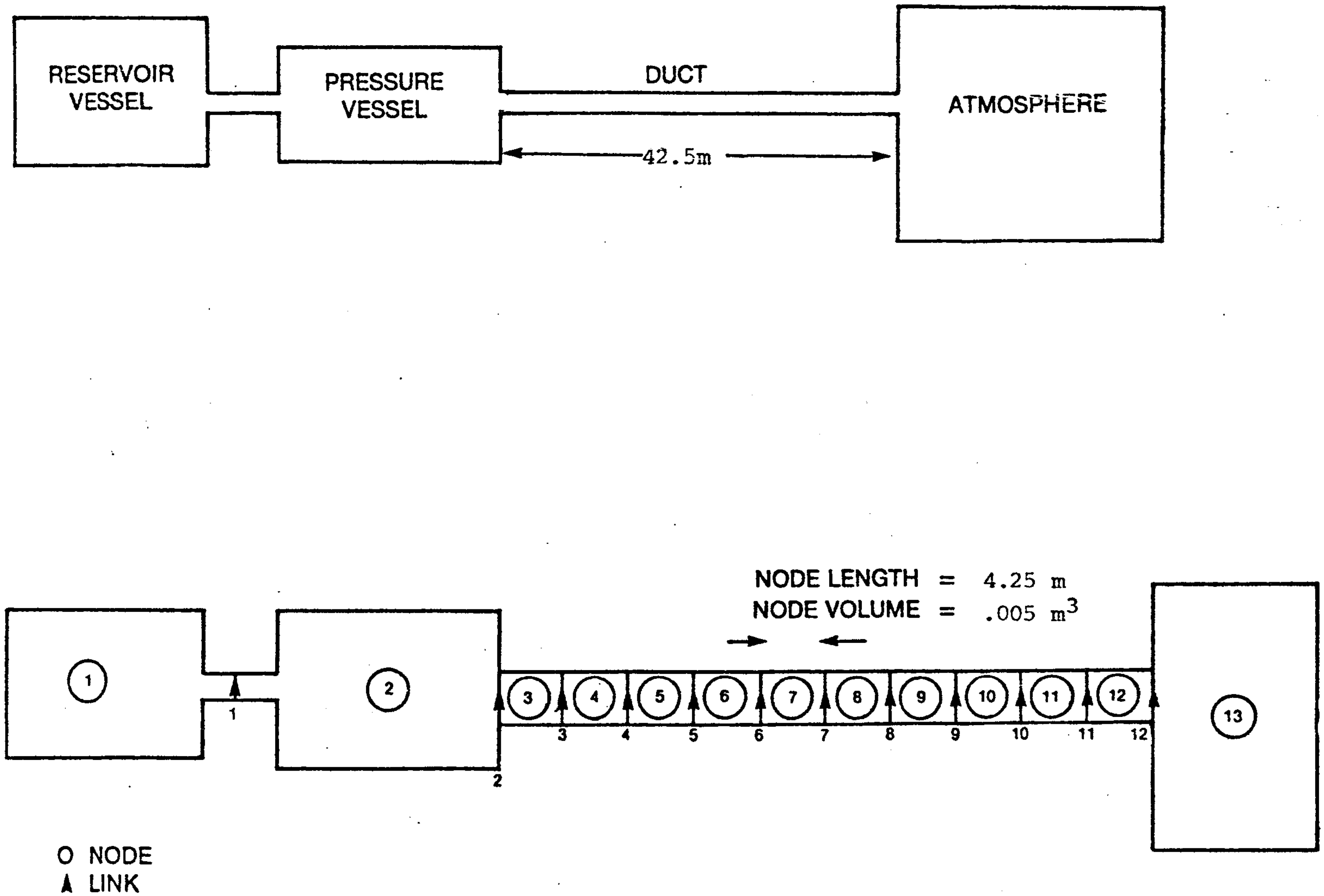


FIGURE 19.37: **SCHEMATIC OF LAYOUT #1 AND CORRESPONDING NODE STRUCTURE**

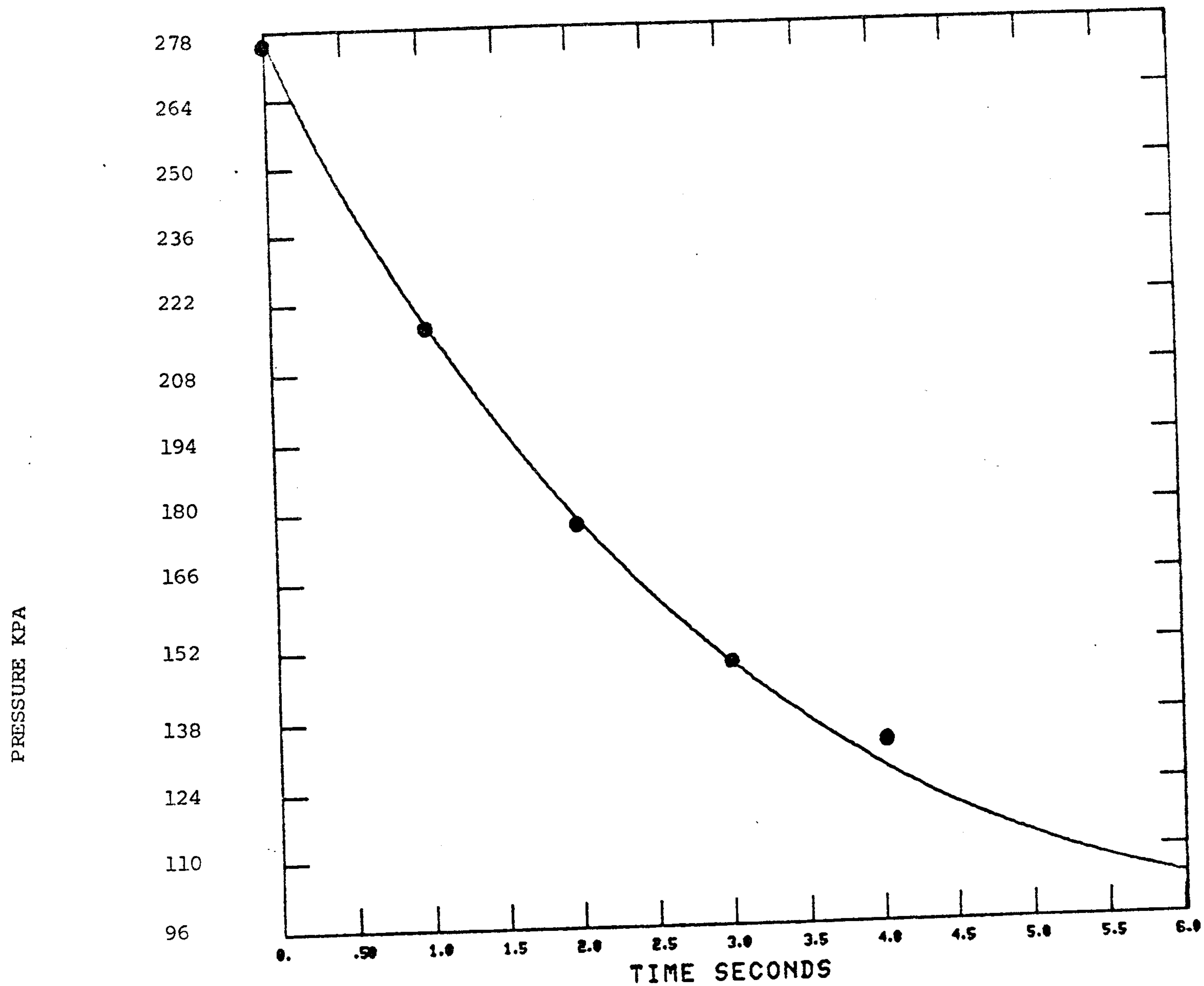


Fig 19.38 RESERVOIR PRESSURE CASE 1 - P_{r_o} = 275 KPa

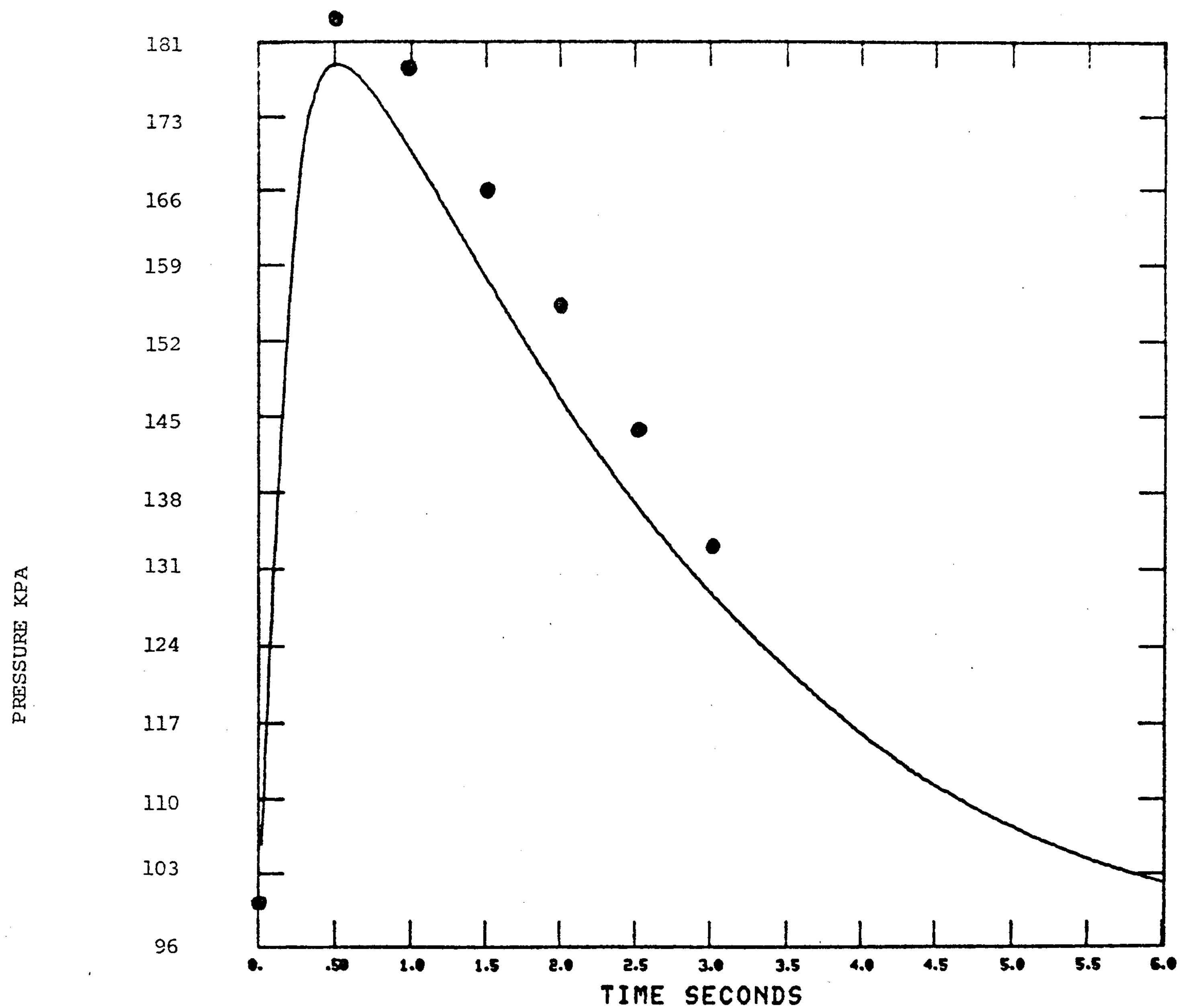


Fig19.39 PRESSURE VESSEL PRESSURE CASE 1 - P_{r_o} = 275 KPa