CHAPTER 16

SAFETY ANALYSIS

16.1 CANDU LOSS OF COOLANT ACCIDENT ANALYSIS

by

V.G. Snell
Manager, Safety Branch
ATOMIC ENERGY OF CANADA LIMITED
CANDU OPERATIONS
Sheridan Park Research Community
Mississauga, Ontario

16.2 CONTAINMENT

by

D.R. Pendergast
Manager, Containment Branch
and
W.M. Collins
Section Head, Containment Branch
ATOMIC ENERGY OF CANADA LIMITED
CANDU OPERATIONS
Sheridan Park Research Community
Mississauga, Ontario

16.3 FUEL AND FUEL CHANNEL BEHAVIOUR

by

E. Kohn
Section Head, Safety Branch
ATOMIC ENERGY OF CANADA LIMITED
CANDU OPERATIONS
Sheridan Park Research Community
Mississauga, Ontario

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Dr. V.G. Snell

Manager, Safety Branch

ATOMIC ENERGY OF CANADA LIMITED

CANDU Operations

Sheridan Park Research Community

Mississauga, Ontario

ABSTRACT

The phenomenology of a loss of coolant accident in a CANDU reactor is described, for the more probable small breaks and also for large breaks. Calculations of reactor power transients and system thermohydraulics are presented for each case. The strain of pressure tubes at high temperature and high internal pressure is described, and representative fuel damage calculations for large and small breaks are presented. Fuel damage is precluded for small breaks other than for special single channel events. For large breaks, fuel damage will be limited to a few high power/high burnup elements, and pressure tube strain is restricted to a small range of large inlet end breaks. Containment response is summarized, and some typical doses are presented, and compared with dose limits.

16.1 CANDU LOSS OF COOLANT ACCIDENT ANALYSIS

16.1.1 PURPOSE

Why do we analyse Loss of Coolant Accidents (LOCAs) anyway? Take large LOCA for example. We spend millions of dollars on R & D and code development so we can analyse the accident yet we'll be lucky (or unlucky) to see one in our lifetime. Are we wasting money?

Actually there are three reasons for LOCA analysis:

1. The results provide a basis for the design of the safety systems. A LOCA is one of the most severe challenges to all three major safety systems: emergency core cooling (ECC), shutdown, and containment. If these are designed to respond well to a LOCA, they will easily handle a much larger spectrum of accidents both more and less probable. Specifically, LOCA analysis confirms or determines: for the ECC system, the pressure, flow, initiation signals and injection points; for the shutdown system, the speed, reactivity depth and certain setpoints; and for containment, signals for dousing, the allowable containment leak rate, design of containment penetrations, and the design pressure.

2. LOCA analysis tells the operator what the accident will look like, what he has to do to manage it, and how much time he has, particularly for small breaks. If interpreted properly, LOCA analysis can avert economic disasters. The problem here is that the conservatisms in the analysis, which are so prominent in our submissions to the regulatory, get in the way of a realistic accident prediction and mislead operational staff as to the potential speed and severity of a small LOCA.

3. LOCA analysis is also used in the licensing of nuclear power plants. It shows that public dose limits are met for very severe conditions and assumptions. It follows that for "real" accidents they will also be met.

16.1.2 HISTORICAL RECORD

What has the actual experience been on LOCA? In Canada there have been no power reactor LOCAs large enough to need the emergency core cooling system.

It's probably worth defining some terms at this point. A large LOCA is a hole of about 300 cm^2 or larger; in other words one well beyond the range of breaks in a feeder pipe (up to 100 cm^2). A small LOCA covers about 2 cm^2 up to 300 cm^2 and includes the feeder pipe range. A very small LOCA is one below 2 cm^2. The actions of the safety systems are as follows: for both large and small LOCAs the shutdown systems, emergency core cooling and containment are all required and are invoked automatically. For very small LOCAs shutdown is manual or automatic, and emergency core cooling and containment are neither needed nor invoked automatically: they serve as backups.

In this context the Canadian experience has been that no large or small LOCAs have occurred, and there have been a few very small LOCAs which required operator control. One can argue that the rupture in the Pickering "A" pressure tube in 1983, being more than 2 cm^2, was a small LOCA and not a very small LOCA. However the discharge was controlled by the flow area through the pressure tube bearings and in fact was well within the normal make-up capability of the reactor, without requiring emergency core cooling.

Worldwide there have been no large breaks in any nuclear power reactor to my knowledge, although there have been several small breaks, the most famous of which is the one at Three Mile Island. As you all know, this small break became an economic disaster when the operator manually blocked and defeated the action of the safety systems.

It is clear from such a limited historical record that we are dealing with highly improbable events, and that analyses, backed up by experiments, are the only methods we will have in the foreseeable future to design safety systems, to define operator actions, and to license nuclear reactors.

16.1.3 DEFINING THE ACCIDENT

The accident is defined by the size and location of the break, in other words by the particular pipe which is assumed to break. How is this chosen? As you know the pressure tube design for CANDU means that there are hundreds of small pipes (less than 8 cm diameter) connecting each channel to the collectors or "headers" above the core. The headers and associated piping around the pumps and boilers are large bore, typically 25 to 50 cm in diameter (Figure 1). Boiler tubing itself is of course very small bore piping, around 1 cm diameter.

There are some obvious lessons from this design. Because of the cost of heavy water, the challenge has been to provide a system which has an acceptably low leak rate. This has been achieved, but LOCA analysis takes for granted that this procedure has failed. For boiler tube failures all we want to do is to detect the leakage quickly before we have wasted too much heavy water into the secondary side. The design response is to provide a deuterium-in-water detection system. Once the operator knows he has a tube leak, his job is to depressurize, then to drain the heat transport system below the level of the leaky tube so that it can be fixed. To help him do this, he uses the shutdown cooling system, which can remove decay heat at full system pressure, meaning that he is not dependent on the secondary side for depressurization/cooldown. There is very little analysis of a boiler tube failure needed beyond this so I shall not dwell on this further: it is classified as a very small leak.

Because of the total length of feeders and pressure tubes, a small break is about 100 times more probable in a CANDU than a large break. Clearly any CANDU LOCA analysis will look carefully at this range for both economic and safety reasons. Finally, in common with the rest of the world, we look at large pipe breaks.

In short, analysis assumes a break in a pipe of a size up to twice the cross-sectional area of the pipe. We include in-core breaks (pressure tube breaks) assuming both that the calandria tube is intact and also assuming it fails. Normally we assume all safety systems work but as a unique feature of CANDU safety practice we also look at failure of subsystems of the safety systems, or in some cases the entire safety system.

Specifically, we look at the following failures in combination with any single process system failure:

- unavailability of a shutdown system (there are two);
- no emergency coolant injection, and no cooldown of the boilers (see Section 4.2.1 for a description of cooldown);
- no containment isolation;
- failure of one dousing sub-system;
- deflated airlock door seals;
- failure of vault coolers.

I shall not be covering these combined failures further in this paper.

16.1.4 SYSTEM BEHAVIOUR

Figure 2 shows a roadmap of how we go about the analysis. Briefly we start off with a calculation of the power transient (reactor physics). We use this together with a thermohydraulics code to predict the circuit and individual channel thermohydraulic behaviour. A more detailed look at channel behaviour gives us hydrogen production and pressure tube deformation. Pressure tubes may expand and contact the calandria tubes, in which case the heat transfer on the outside of the calandria tube is of interest, and this leads us to an investigation of moderator circulation and local temperatures. A further level of detail gives us sheath temperatures and fission product releases. These are used in a containment calculation to determine how much activity leaks outside the building. The dispersion and dilution of this material before it reaches people is the subject of an atmospheric dispersion/public dose calculation. The public dose is the end-point of our calculation.

Let's consider each of these in turn.

16.1.4.1 Reactor Physics

For a large break, a reasonably rapid shutdown is desirable to reduce the amount of stored heat that the emergency core cooling system has to handle. Boiling in the channels in a CANDU following a loss of coolant can introduce positive reactivity at a rate of up to 8 mk per second, for a couple of seconds. Solving the neutron kinetics equation, we find that for the worst large breaks, power will double about every 1½ seconds. So we have to get our shutdown systems going within a second, and we can do this with fairly simple mechanical devices. Most of the fission power has gone after 10 seconds or so. Decay heat of course, is still being produced at about 6% of the initial thermal power. This will decrease exponentially, reaching about 1½% after an hour. Interestingly enough, in the CANDU design after about 3 months we can get rid of all our heat from the fuel through the pressure tube to the calandria tube and to the moderator, while keeping fuel temperatures below 600°C, without any coolant in the channels at all.

How then do we go about doing a reactor physics analysis? Remember that only 1 loop of a CANDU voids so that the void reactivity and power will be asymmetrical. If you imagine a vertical line down the face of the reactor on Figure (1b), passing through its centre, then all the channels on one side of the line are in one loop, and all the channels on the other side are in the other loop. For a large break, that calls for a three-dimensional transient solution, which we carry out. From this we get the power transient for any location in the core, so we can generate "hot" channel and "hot" bundle power transients for later use. Usually we assume only the less effective shutdown system is operational (in this case, the shutoff rods, or SDS #1) and even then we assume two rods are unavailable - the two most effective rods in fact. The hot bundle tends to be towards the bottom part of the core in the area of the missing rods. Figures 3 and 4 show power transients for the voided half core and the hot bundle. These are input to the subsequent circuit, channel and bundle calculations.

Obviously we had to get started somehow, and indeed we use an approximate power pulse in a circuit blowdown simulation, in order to get detailed void predictions for the reactor physics calculation. It turns out that the final power transient is not at all sensitive to the power transient that went in, in the front end, since the fuel acts as an unresponsive heat storage mass.

16.1.4.2 Thermohydraulics

You will remember that the CANDU 600 geometry consists of two independent figure-of-eight loops (Figure 5). These isolate very shortly after a loss of coolant so that only one loop loses substantial inventory before ECC injection. Let's concentrate on this one.

16.1.4.2.1 ECC Design

To get some idea of timing, let's look at a few typical numbers. A loop contains about 40 thousand kg of D_2O. Subcooled break discharge at operating pressure is about 60 kg/cm^2-s. However as the circuit voids, quality will build up very rapidly and this rate will fall within a few seconds or less to a tenth of this value or less. If left alone, for the largest break a circuit would be mostly empty (pressure $< \frac{1}{2}$ MPa) by about 60 seconds. Fuel heat-up could start quite early if the break was chosen appropriately, and we shall discuss this later on. In the worst circumstances, fuel in half the channels of the loop could lose most of its cooling within about 2 seconds.

The first effect would be to redistribute the stored heat within each fuel element in a period related to the time constant of the fuel - 7 seconds. Thus the sheath heats up at about $100^{\circ}C$ to $200^{\circ}C$ per second until the fuel and sheath are at the fuel average temperature, which for the hottest pin is $\sim 1200^{\circ}C$. Thereafter the heat-up is limited by decay power and if there is no cooling at all on the fuel, this will proceed at about one degree per second per percent full power for the hottest fuel element.

Fuel at $800^{\circ}C$ to $1200^{\circ}C$ will not fail until the coolant pressure falls well below the internal gas pressure inside the sheath. At this point the sheath will start to strain and if the temperature is not reduced and the coolant pressure continues to fall, sheath damage would occur. If we look at typical depressurization rates for loss of coolant this will begin to happen at around 30 seconds for the most severe large break.

This then suggests the parameters of our emergency core cooling system. If we wish to limit sheath damage we must get substantial amounts of water in quickly before the pressure falls to around 1 MPa since it takes some time, typically 30 seconds or so, before water injected into the headers can penetrate the feeders, cool the channel, and re-wet the fuel. This led us to the accumulator concept for high pressure ECC injection. These are sized to refill the core; they contain about three times the loop inventory and are driven by gas at a pressure of 4 MPa. The actual pressure setpoint prevents fuel damage for small breaks, as we discuss later on, and provides the required refill speed for large breaks.

Once the core is refilled we must keep it full, and this leads to the medium and low pressure pumped ECC which is similar to other concepts. The medium pressure ECC takes water from a tank and injects initially at a pressure of about 1.4 MPa, and at low pressure can give a flow of 700 kg per second which can keep up with the largest break. It is started automatically once the high pressure water is near exhaustion. Once the medium pressure tank is empty, at about 15 minutes to half an hour for large breaks, the operator will switch over to low pressure recovery in which the same pump is used to take water from the floor of the building, pass it through heat exchangers, and re-inject it into the heat transport system. The flow and pressure are similar, albeit somewhat reduced, because of the loss of head upstream of the pumps.

So much for ECC design. Actually, not quite. The system we have described will not work well on its own for small breaks. For these the boilers act as giant pressure control reservoirs, and will hold the primary system near boiler pressure throughout much of the accident. If the accumulator pressure were to drop below boiler pressure, as it eventually will, then it would no longer be able to inject ECC. To stop this happening, the boilers are automatically or manually cooled by opening up all of the main steam safety valves; these can release about 110% of normal steam flow. This ensures that the primary side depressurizes and keeps the ECC water coming in.

Where do we inject the water? In the present design we inject it into all reactor headers, in both broken and intact loops, regardless of break location. This means that a certain fraction of the water is wasted directly out of the break, if the break happens to be at a header, but the accumulator is sized to account for this. We have tried more sophisticated schemes on occasion as I shall mention later.

In short, very simple timing considerations indicate a high pressure ($\sim$4 MPa) ECC initially with both flow and pressure sufficient to maintain around 1 MPa in the channels during refill. This is coupled with rapid cooldown of the boilers to ensure continued injection for small breaks.

16.1.4.2.2 Phenomenology of LOCA

In CANDU 600 MW plants our design basis assumes the availability of Class IV power; thus the main PHT pumps are left running during the LOCA and, as we shall see, provide a very important flow bias. We also studied the effect of tripping the pumps, and I shall discuss this later on, but for the time being let's look at the running pumps case.

14.1.4.2.2.1 Small Breaks

Let's begin with small breaks. The interest for these is to prevent fuel damage so that the plant can be restarted without an economic disaster. For breaks even up to a severance of 10 feeder pipes, the pumps are much more influential in determining flow than the break. Thus the flow is always forward or recirculating, although as quality builds up with time, the pump head will decrease and the resistance of the circuit will get higher; thus the flow will fall with time.

At what point do we have to intervene? The first thing is to shutdown the reactor before fuel sheaths experience prolonged dryout at high power. This is more a precautionary (economic) requirement than a safety requirement. This is accomplished by shutdown signals on process parameters such as low flow, low pressure, low core pressure drop or high building pressure within containment. For a feeder size break this is necessary within the first three or four minutes. As the circuit continues to empty, the flow in the headers or channels will eventually fall low enough that the coolant phases will separate. This means that we would get steam cooling of some of the upper pins in a channel or of some of the channels connected to the mid plane of the header. Prolonged stratification will lead to sheath damage in the order of a few minutes so this defines the time at which we have to get water in. An ECC injection pressure of about 4 MPa limits the duration of stratification to the order of seconds and is clearly adequate. Once refill has occurred the pumps keep recirculating flow and this pattern continues into the very long term.

The break is a major heat sink for decay heat. Breaks greater than half the size of a feeder break can remove all the decay heat from one loop. The boilers therefore act more in terms of a huge pressure control system than an effective heat removal system; as mentioned earlier the trick is to cooldown the boilers fast enough that injection is not blocked. Crash cooldown takes the boilers from 4MPa to close to atmospheric in about 15 minutes and this allows continued makeup flow.

These parameters will prevent fuel damage for a small break. There is a special case of small breaks for which we can't prevent limited fuel damage, and that is certain breaks which occur in the piping connected to one channel only. The rest of the heat transport system sees just a small break, but the affected channel could have both fuel and channel damage. When we analyse these, our emphasis is on the affected channel. Such accidents include: - complete flow blockage of one channel - spontaneous channel rupture - end fitting failure (Figure 1b shows the location of these components). All of these can damage fuel: the first by overheating due to reduced flow, the second by mechanical damage following the pressure tube rupture, and the third by ejection of the fuel into the calandria vault followed by mechanical damage and oxidation in the vault atmosphere.

Whether the analysis of these accidents helps the station operator is a moot point: a "clean" endfitting failure which allows the fuel string to be discharged seems mechanically difficult and a flow blockage severe enough to cause pressure tube failure due to overheating would have to be greater than 90% of the channel flow area. It is probably better to view an endfitting failure analysis as a test of the containment response to low discharge rates and high activities, and to view a flow blockage or a pressure tube rupture analysis as a test of the calandria response to overpressure.

4.2.2.2 Large Breaks

Three break locations are enough to define the heat transport system behaviour: at the core inlet (inlet header), at the core outlet (outlet header), and upstream of the pump (pump suction header).

These breaks affect the two core passes in different ways. The core pass upstream of the break (by upstream and downstream I mean with respect to the nominal flow direction) always has the flow accelerated towards the break. The fuel cooling is if anything increased and emergency core coolant refill is rapid, in the nominal flow direction towards the break. The core pass downstream of the break will have its flow reduced by the break and this is where most of our analytical effort goes.

First consider breaks at the core inlet or the inlet header. A guillotine inlet header break will reverse the flow in the downstream core pass, and as we noted earlier, a small break at the inlet header will leave the flow in the normal direction. It is therefore possible on paper to choose a break size which leads to a momentary period of zero flow in the downstream core pass. This low flow results from a balance between the break and the pumps. It is by nature transitory: as the circuit depressurizes, the break discharge will decrease and at the same time the pump force will decrease as quality builds up in the pump. The latter effect is faster so that after a short period of zero flow the break takes over and the flow reverses towards the break.

Figure 6 shows core flow versus time for a few inlet header breaks. The time at which the low flow occurs varies slightly with break size: sheath temperatures are highest if the stagnation occurs at the time when the stored heat is still high - that is within a second or so of reactor trip - and therefore this particular break attracts most of our analysis dollars.

It is clear what the long term solution will be. If the void at the pump is collapsed and the break is at the small end of the spectrum, both refill flows and long term flows will be forward: that is, in the normal recirculating fashion. If the pump remains voided or the break is large, refill will be in the reverse direction and this will persist into the long term. A narrow range of intermediate behaviour is predicted around the transition size of a 35% inlet header break; the actual switchover from reverse to forward flow is rapid and no long term balance of break and pump forces seems possible.

Because inlet header breaks are the closest to the downstream core pass they can reduce the flow fastest and therefore lead to the highest fuel temperatures. By the same token stagnation does not persist and fuel is cooled down while the coolant pressure is still relatively high. The combination of high temperature and high coolant pressure means that we look at 1) the possibility of embrittlement of the fuel sheath due to oxygen uptake and 2) the possibility of pressure tube strain due to overheating at high pressure. I will be discussing these later on.

I noted earlier that large breaks tend to attract a portion of the analysis effort out of proportion to their probability. Even in the large break range the stagnation breaks which represent about 10% of all large breaks, attract most of our large break effort despite their even lower probability. This is one of the consequences of a licensing analysis: If events can be bounded by a more severe event, then the severe event will be exhaustively analysed at the expense of the less severe ones, to husband limited resources. Stagnation breaks supply limiting conditions for both fuel behaviour and pressure tube behaviour. They therefore are analysed almost exclusively in the large break range.

The largest break at the <u>outlet</u> end of the core (reactor outlet header) is just able to reverse the flow in the downstream core pass. Smaller outlet header breaks allow continued forward flow: refill is in the forward direction and the long term flow pattern is recirculating. For the largest outlet header break the voiding of the downstream core pass is slower than for inlet breaks since the path from the break to the core is longer and the resistance is higher. Thus when stagnation does occur there is less stored heat in the fuel. Fuel temperatures during stagnation rise to slightly lower values than for the inlet header case. Figure 7 shows the flow versus time for two large outlet header breaks.

At first the flow goes backwards through the downstream pump but as the circuit depressurizes the pump acts more and more like a check valve. Injection water into the inlet header of the downstream core pass is therefore prevented from going through the pump to the break and instead is forced in the forward direction through the core pass. We can see that refill of both core passes is in the forward direction and that the long term flow pattern for the largest ROH break, is recirculating. Figure 8 shows the flow patterns during refill. Because the depressurization is more rapid than the inlet break, fuel heat-up occurs at a lower coolant pressure. Pressure tube strain is of less interest but fuel sheath strain is of more interest due to the larger driving force across the sheath. (Figures 9 and 10)

Pump suction breaks are hydraulically similar to reactor outlet header breaks. However, being closer to the affected core pass one gets the same results with a smaller break area.

So far we have been talking about the behaviour of the core average channel. There is of course no such thing. A CANDU reactor core pass consists of 95 parallel channels of varying powers, elevations and resistances. How do they really behave? Basically the running pumps define a flow direction so that the behaviour of an individual channel is similar to that of an average channel. Indeed the 600 MW channels are flow/power matched; in other words the resistance of each channel is tuned to its average power so that the exit quality of all channels is about the same. Thus variations in refill time due to channel power (i.e. stored heat) are offset by lower resistances in the feeders. In LOCA, with its high void, the effect of lower resistance is larger than the channel power so that we expect that high power channels will be the first to fill, with high-elevation, orificed, low-power channels the last to fill.

16.1.4.2.3 Effect of Loss of Forced Circulation

It is clear that the main heat transport pumps influence refill behaviour. What happens if we lose power on loss of coolant so that we do not have the pumps to assist the flow? Inlet header breaks and pump suction header breaks now become hydraulically similar since the pump provides less resistance to reverse flow. Flow in the downstream core pass is now strongly biased towards the break so that refill and long-term flows are reversed in the downstream core pass for all large inlet end breaks. Flow in the upstream core pass, as before, is towards the break. For large outlet end breaks, a force balance across the downstream core pass can be sustained for much longer than if the pumps were running. The reason is as follows (Figure 11): Injection into the downstream inlet header can now go backwards through the stopped pump; most of it does and instead of going through the downstream core pass as before goes backwards over the boiler and to the break. See Figure 11. This means that the only source of refill for the downstream core pass is injection into the outlet header which would have to go backwards through the core pass to refill it. However this water also has another route which is over the boiler through the upstream core pass and to the break, and we find that for very large outlet breaks most of it goes this way. Thus refill for large outlet breaks is prolonged - of the order of many minutes. This accident is not a design basis accident in the Canadian licensing context, so that such a delayed refill has not required design solutions. However in many foreign countries, loss of off-site power coincident with an accident is a design requirement.

Clearly what we want to do in such a case is break the balance of forces across the downstream core pass. We can do this by injecting water only into the unbroken end of the core. This gives a pressure gradient from unbroken to broken end allowing the upstream core pass to refill in the forward direction as previously, and the downstream core pass to refill in the reverse direction. To encourage this process we do not cool down the boiler at the unbroken end to prevent injection into the downstream outlet header from going over this boiler and bypassing the core pass.

Small breaks with loss of forced circulation result in thermosyphoning once the circuit has been refilled. However since thermosyphoning forces are of the same order as the forces due to small breaks, if they oppose each other a dynamic balance can be set up. For example a break on the inlet side of the core, if small, can oppose the thermosyphoning force and result in a reduction of the header to header pressure drop to near zero. The result is not indeterminate stagnant flow, but rather cyclic formation of a steam bubble in a channel, growth of the bubble until it reaches either feeder pipe and then a sudden rush of water back into the channel as the bubble shoots up the feeder pipe (Figure 12). This behaviour is called "standing start" and has been observed in full scale channel experiments as shown by Figure 13. Obviously it produces cyclic fuel heat-up and cooldown and for most conditions of interest these are not high enough to result in much fuel damage.

16.1.4.2.4 Experimental Support

It is worth asking at this stage what evidence do we have that our predictions are meaningful for the large breaks? The CANDU design permits blowdown and refill testing on full scale reactor channels under a variety of pressure drop conditions. Many of these tests have been performed and the results compared to thermohydraulic code simulations of the same tests. The results are shown in Figures 14 and 15 in terms of predicted versus measured rewet time and sheath temperature rise. We see that for refill times of the order of 200 seconds or less the accuracy of the FIREBIRD-III code is quite satisfactory; for longer refill times the code tends to underpredict refill by a slowly increasing factor. The arrow on the figures down near the origin shows the predicted rewet time of the 600 MW CANDU. It is well within the region of acceptable code accuracy.

16.1.4.3 FUEL CHANNELS

If the pressure tube temperature is greater than about 650°C and the internal pressure is greater than about 1 MPa the material will start to deform. This will continue either until the pressure tube cools down or until it expands to contact the calandria tube. At this point its stored heat is rapidly transmitted through the calandria tube to the moderator. This cooldown, taking place over about 1 to 2 seconds, arrests further expansion and provides a direct path for channel heat to be rejected to the moderator as well as an expanded flow area which assists channel refill.

The speed of cooldown of the tube assembly depends on the moderator subcooling. At high sub-cooling the surface of the calandria tube will be in nucleate boiling and cooldown will be very rapid. As subcooling decreases towards saturation, patchy film boiling will appear in places on the calandria tube and these patches will take some tens of seconds to cool down. At very low subcooling large areas of the calandria tube will be in film boiling and failure of the tube assembly is possible. Figure 16 shows contact temperature versus pressure for an expanding pressure tube assembly and shows the areas of nucleate boiling, patchy film boiling and extensive film boiling.

In the actual analysis only reactor inlet header breaks have the combination of high pressure and high temperature sufficient to cause pressure tube deformation to any great extent: about 50 pressure tubes would strain to contact the calandria tube for the worst break. Contact conditions are 800°C pressure tube temperature and 25°C subcooling for the worst case (a high elevation, high power channel near the moderator hotspot): nucleate boiling is expected following contact.

16.1.4.4 FUEL

The fuel sheath is the primary barrier to release of radioactive material. Under the loads produced by a LOCA it is subject to high temperature and is stressed due to the difference between the internal gas pressure and the falling cooling pressure. The failure modes of interest under such conditions are as follows:

(i) plastic instability (ballooning);

(ii) Beryllium braze penetration;

(iii) sheath embrittlement due to oxygen uptake;

iv) cracking of oxide layer under stress.

A sheath when stressed will strain uniformly to some extent and then the strain will localize (balloon) leading to failure. The uniform strain that will be reached before ballooning starts varies with temperature and is at a minimum of 5% in the α/β transition region. At higher and lower temperatures it can be as high as 10%. For purpose of release calculations we assume failure at 5% regardless of temperature.

Sheath strains are highest for the outlet end breaks which combine high sheath temperatures and low coolant pressure. However no matter how strong or weak a sheath is, it can only expand to the point where the internal gas pressure is equal to the external coolant pressure; indeed if the sheath has any strength at all it will expand less than this. Such a strain is called "terminal strain" and represents an upper bound to the amount of uniform strain. The high pressure injection system of the CANDU 600 can maintain a back pressure in the channels of about 1 MPa during refill even for the largest outlet breaks (Figure 10). A pressure of 1 MPa is sufficient to limit sheath strains to about 5 - 6% even for the elements with the worst combination of power and burn-up. Thus even if refill times were longer than predicted the amount of fuel damage would still be fairly small.

The other mechanisms of sheath failure have been covered extensively elsewhere* and I will not deal with them here; in any event they are not limiting for LOCA.

16.1.4.5 CONTAINMENT

In principle the CANDU reactor design and the choice of containment concept are fairly independent. Two different concepts of containment have been used in Canada, the single unit containment and the vacuum containment. I shall describe them briefly.

The CANDU 600 design consists of a single unit containment with an active pressure suppression: dousing via ordinary water sprays. Large breaks will initiate dousing within a couple of seconds and dousing will remain on until the dousing water is exhausted. Thereafter the containment pressure will rise until the heat input can be handled by the local air coolers. Figure 17 shows the typical containment response for a large LOCA.

Canada is also home to the vacuum containment concept in which a number of reactor buildings are connected to a vacuum building, which is kept at reduced pressure. The vacuum building also contains a supply of dousing water which is initiated automatically if the pressure inside the vacuum building should rise. The pressure suppression given by this concept is very powerful: it can hold the containment pressure subatmospheric for a few days after a LOCA.

16.1.4.6 RELEASES AND DOSES

The final stage of our analysis is calculating the dispersion of any radioactivity released from containment and the resulting public dose. Dispersion of a release tends to vary inversely with distance from the source along the cloud centreline, and exponentially (gaussian) away from the centreline.

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* See References 8 & 9.

Given a release, the resulting dose is highly sensitive to weather: switching from the worst weather (a moderate wind of 2 metres per second with no wind swing) to average weather gives a reduction in dose of about a factor of 100.

For a small LOCA the doses are negligible because there is no fuel damage other than perhaps propagation of a few incipient defects. Doses for a large LOCA for pessimistic weather assumptions are shown in Table 1 and reflect the limited amount of fuel damage that is expected. We can also do a bounding calculation: failure even of every element in the downstream core pass would result in the doses shown in the second line of the Table, still well below the single failure dose limits shown in the third line.

16.1.5 CONCLUSIONS

A small LOCA in a CANDU exhibits continuous recirculating flow and no fuel damage except for some cases both specific to and limited to a single channel

For a large LOCA the refill direction is determined by the interaction of the running pumps and the break. Refill times are of the order of 2 minutes or less and sheath damage, while not precluded, is expected to be limited to the high power, high burn-up elements. Pressure tube expansion will occur for a range of large reactor inlet header breaks. The resulting doses are well below regulatory guidelines.

TABLE 1

DOSES TO THE CRITICAL INDIVIDUAL FOLLOWING A LARGE LOSS OF COOLANT

	Thyroid Dose (Sv)	Whole Body Dose (Sv)
Release based on probability of fuel element failure	6×10^{-4}	5×10^{-7}
Release of quarter core free inventory to containment	8×10^{-3}	5×10^{-5}
Atomic Energy Control Board dose limits for single failure	3×10^{-2}	5×10^{-3}

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2. V.G. Snell, "Safety of CANDU Nuclear Power Stations", Atomic Energy of Canada Limited publication AECL-6329, July 1980

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4. G. Kugler, "Distinctive Safety Aspects of the CANDU-PHW Reactor Design", Atomic Energy of Canada Limited publication AECL-6789, January 1980

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7. A.R. Dastur, G. Kugler, B. Rouben, and H.Y.M. Li, "Confirmation of CANDU Shutdown System Design and Performance during Commissioning", Atomic Energy of Canada Limited publication AECL-5914, October 1977

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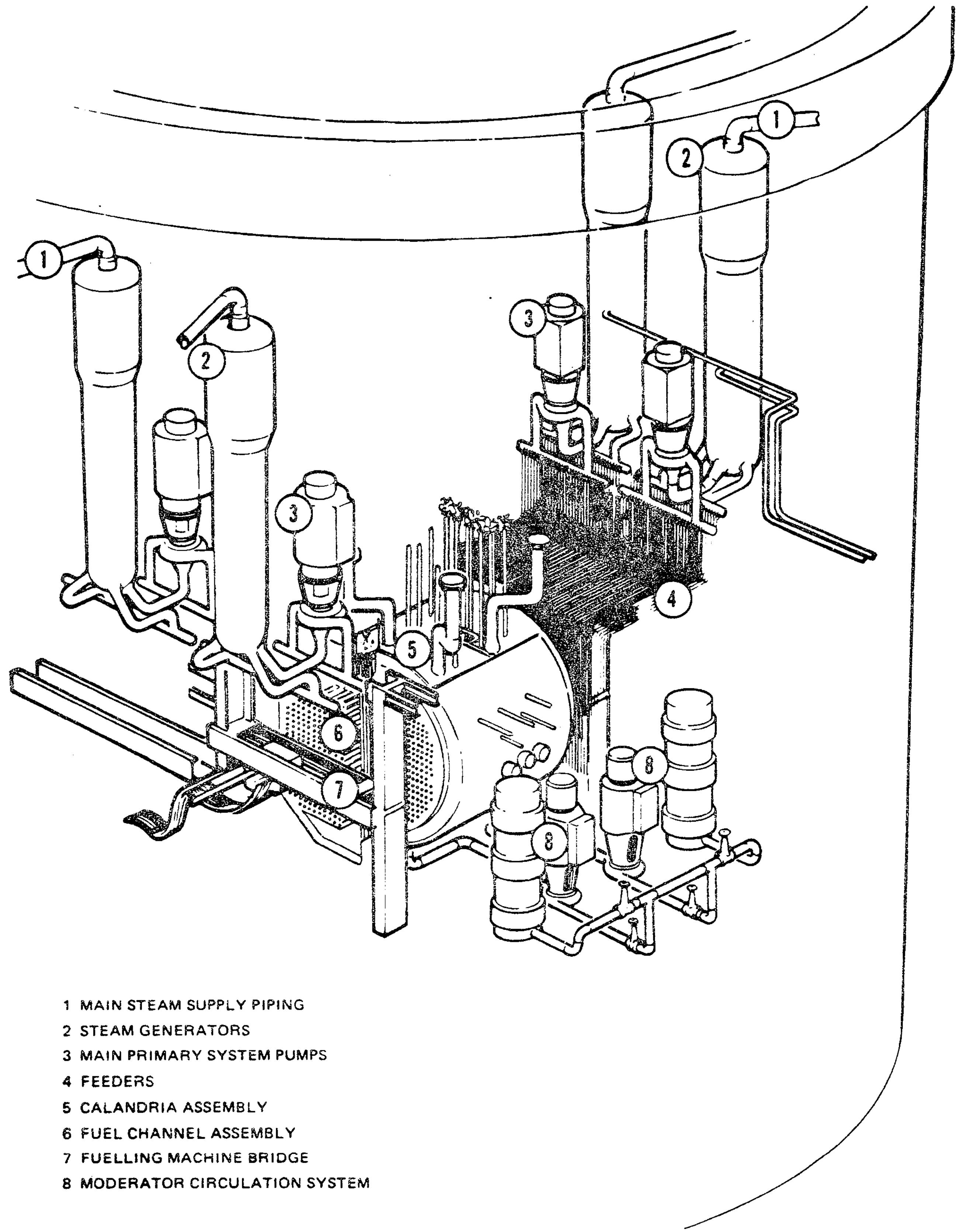
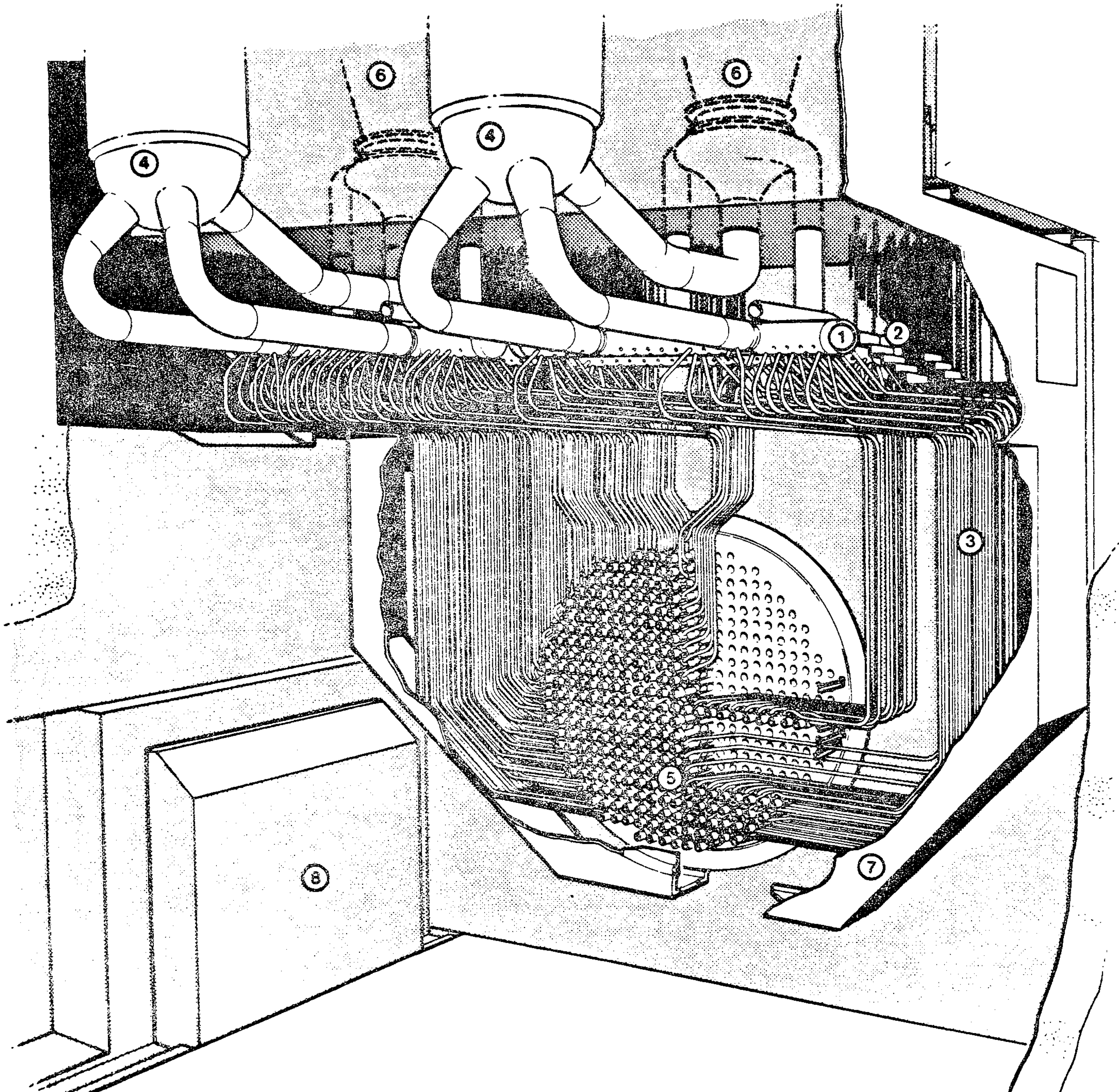
8. H.E. Sills, "ELOCA - Fuel Element Behaviour during High Temperature Transients", Atomic Energy of Canada Limited, Publication AECL-6357, March 1979

9. A. Sawatzky, "A Proposed Criterion for the Oxygen Embrittlement of Zircaloy-4 Fuel Cladding", Zirconium in the Nucl. Industry (Fourth Conference), ASTM STP 681, American Society for Testing and Materials, 1979, pp 479-496



1 MAIN STEAM SUPPLY PIPING
2 STEAM GENERATORS
3 MAIN PRIMARY SYSTEM PUMPS
4 FEEDERS
5 CALANDRIA ASSEMBLY
6 FUEL CHANNEL ASSEMBLY
7 FUELLING MACHINE BRIDGE
8 MODERATOR CIRCULATION SYSTEM

FIGURE 1a CANDU-PHW REACTOR



1 OUTLET HEADER
2 INLET HEADER
3 FEEDERS
4 STEAM GENERATORS
5 END FITTINGS
6 HEAT TRANSPORT PUMPS
7 INSULATION CABINET
8 F/M MAINTENANCE LOCK SHIELDING DOOR

FIGURE 1b HEAT TRANSPORT SYSTEM — TYPICAL FEEDER AND HEADER ARRANGEMENT

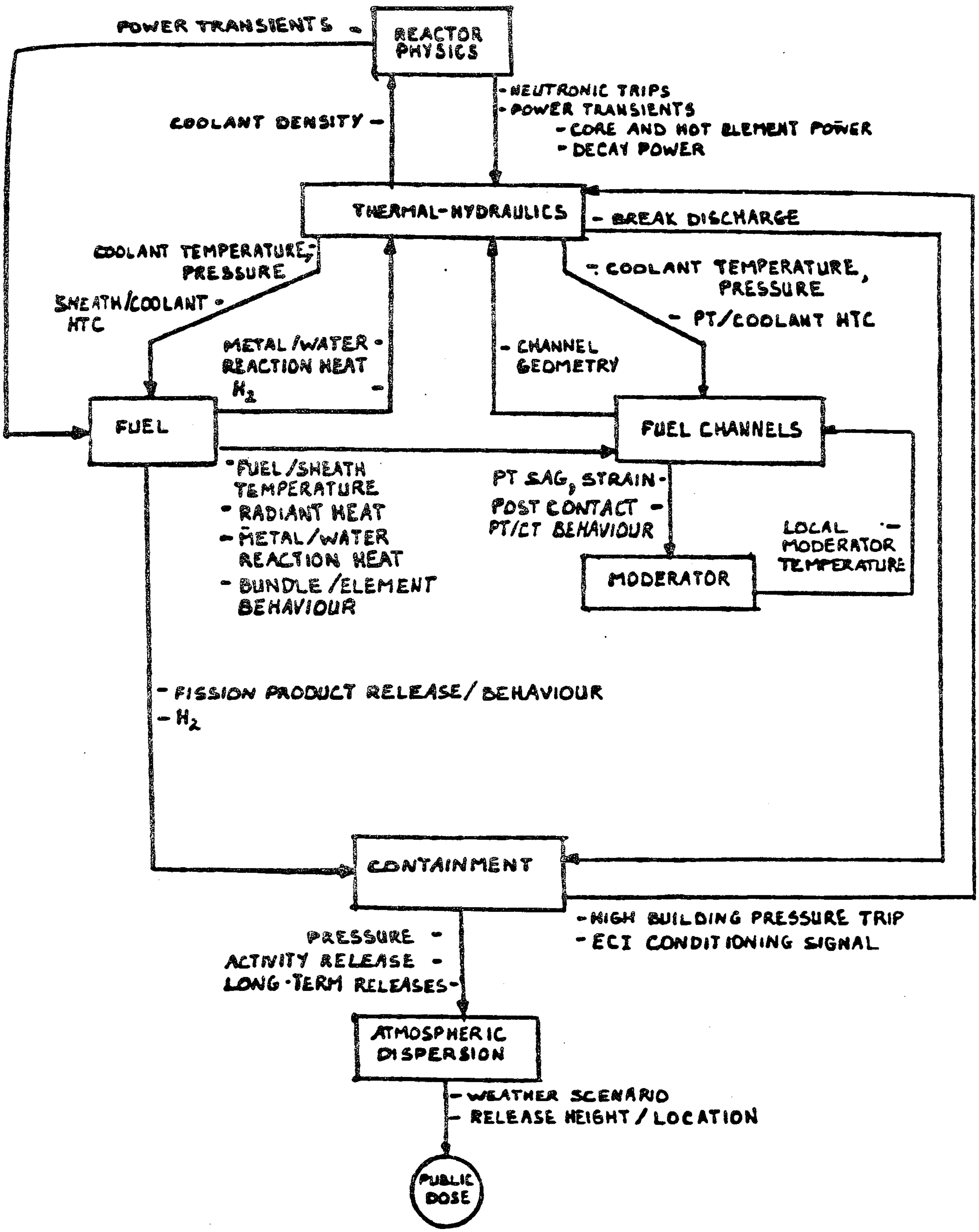


FIGURE 2 INTERACTION BETWEEN ANALYSIS DISCIPLINES — LARGE BREAKS IN THE PHTS, SINGLE FAILURE

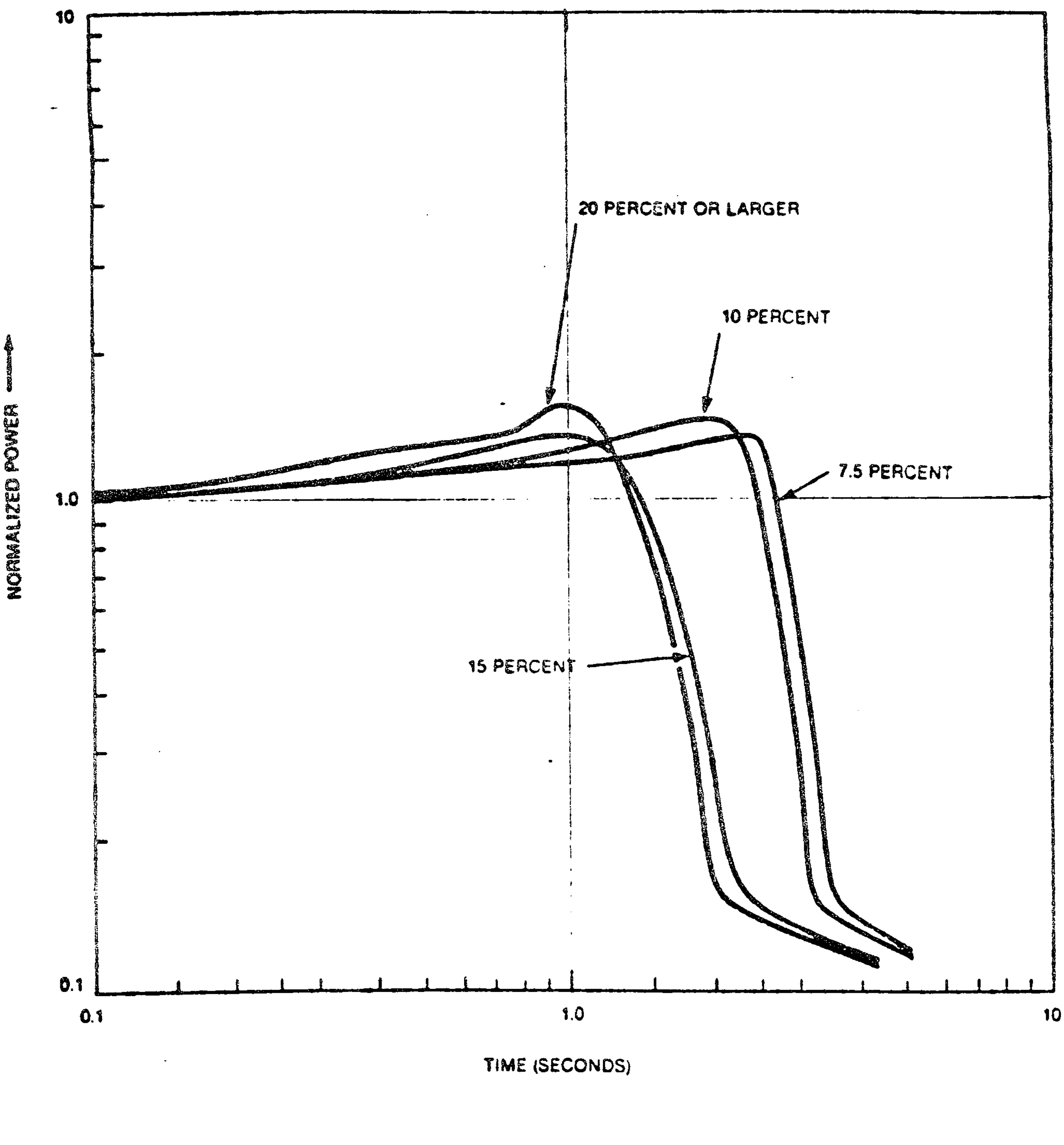


FIGURE 3 VOIDED HALF CORE POWER TRANSIENTS FOR REACTOR INLET HEADER BREAKS

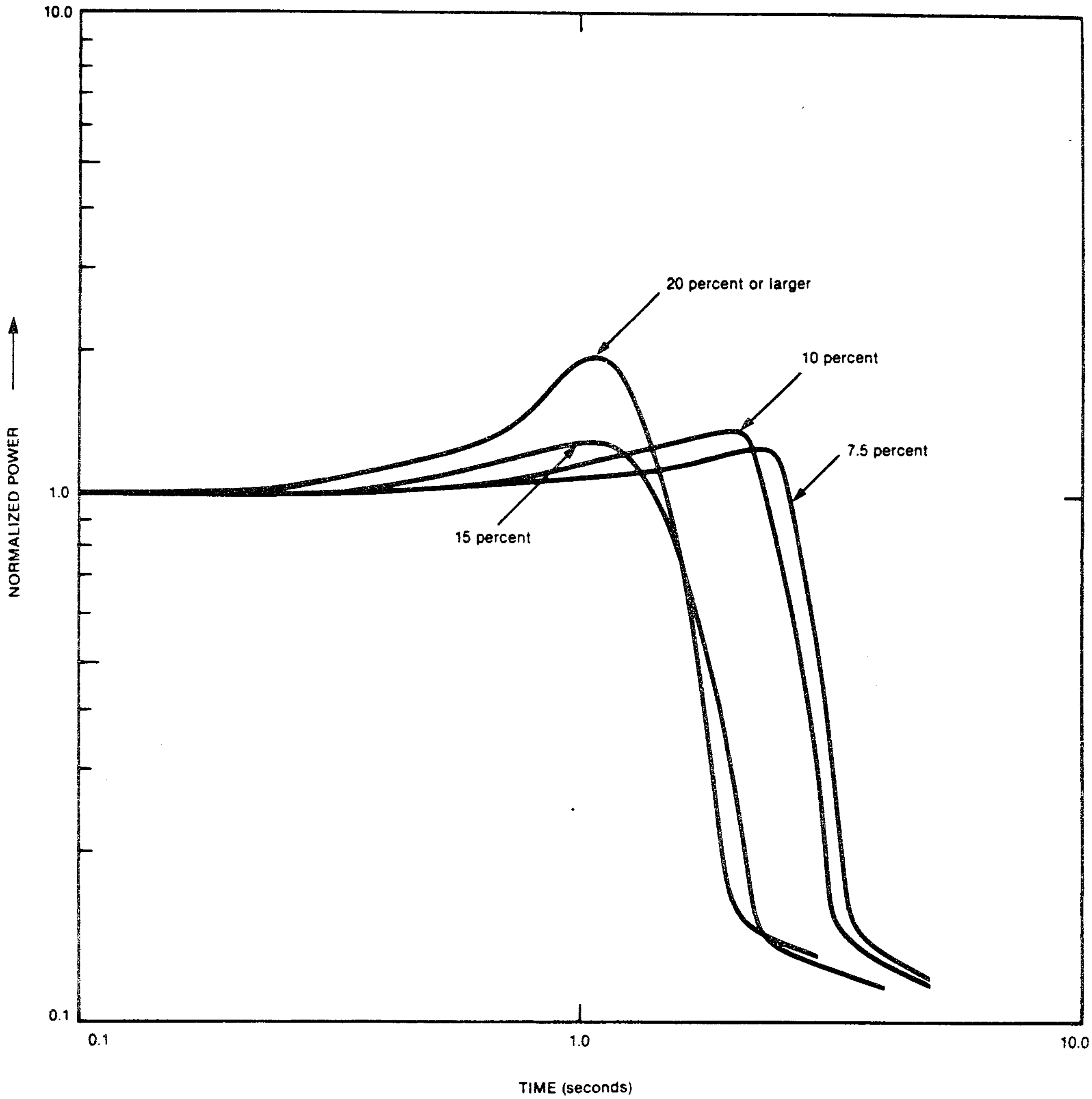


FIGURE 4 HOT ELEMENT POWER TRANSIENTS FOR REACTOR INLET HEADER BREAKS

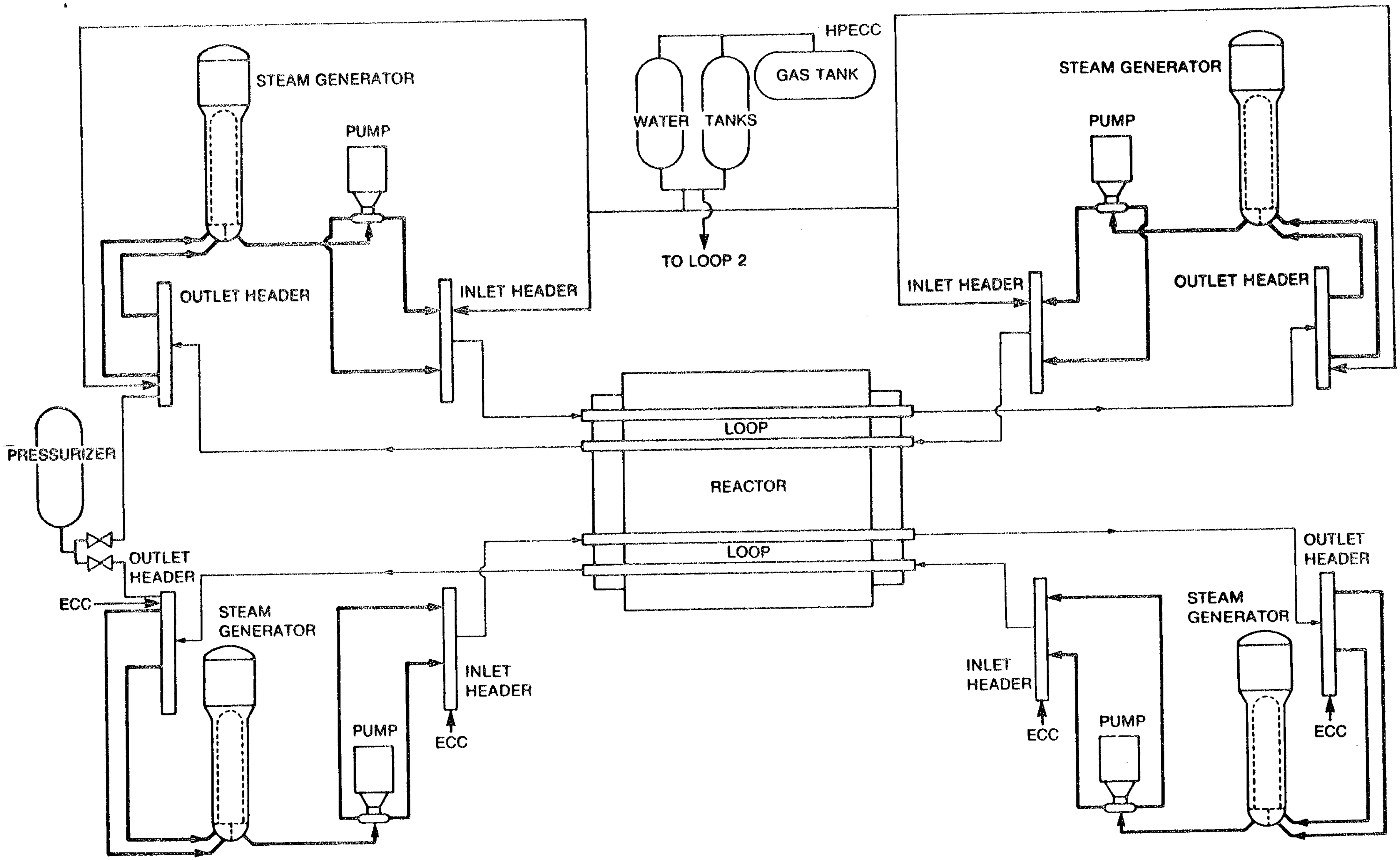


FIGURE 5 HEAT TRANSPORT SYSTEM AND ECC (SIMPLIFIED)

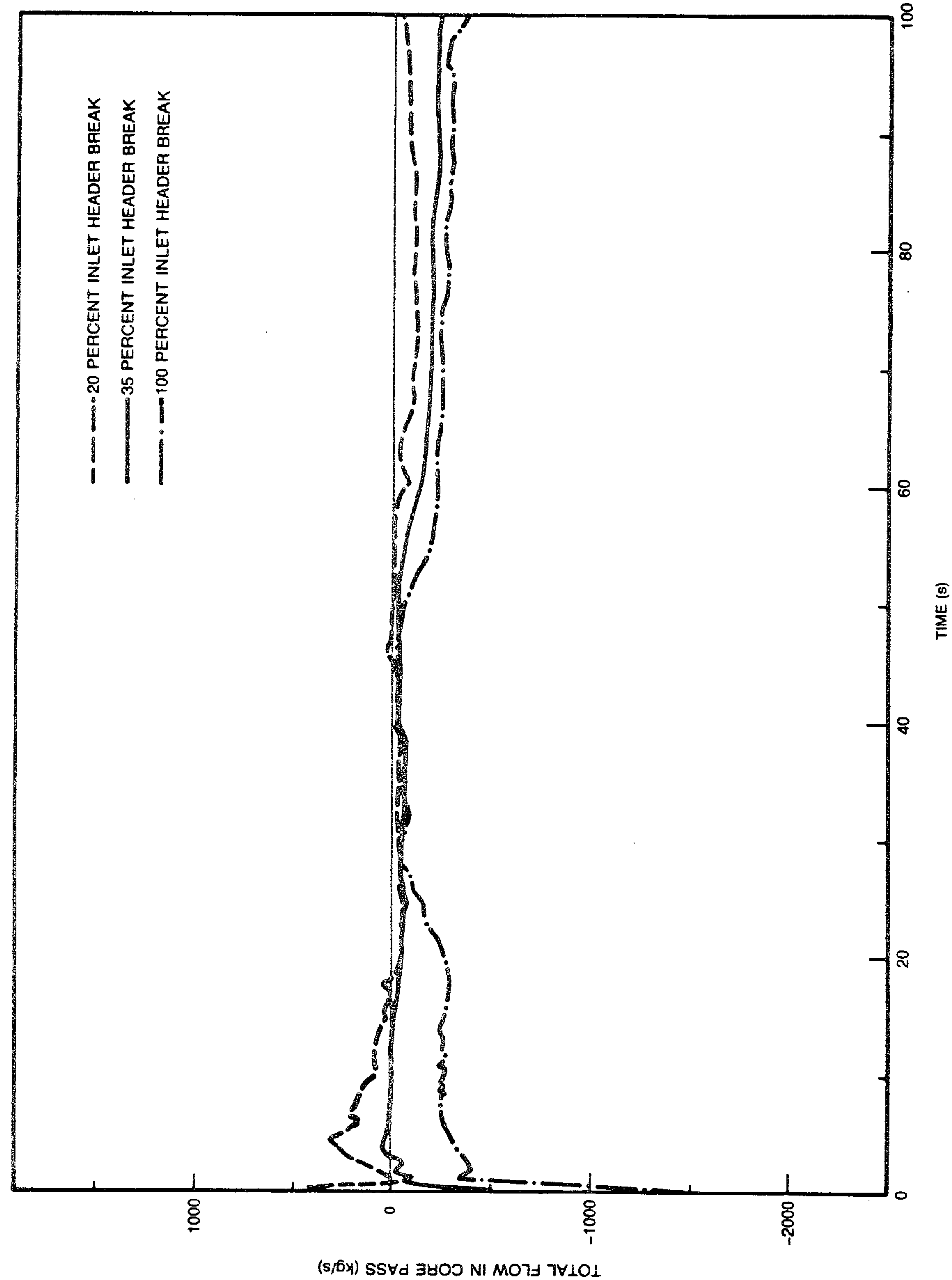


FIGURE 6 FLOW IN CORE PASS DOWNSTREAM OF 20, 35 AND 100 PERCENT REACTOR INLET HEADER BREAKS

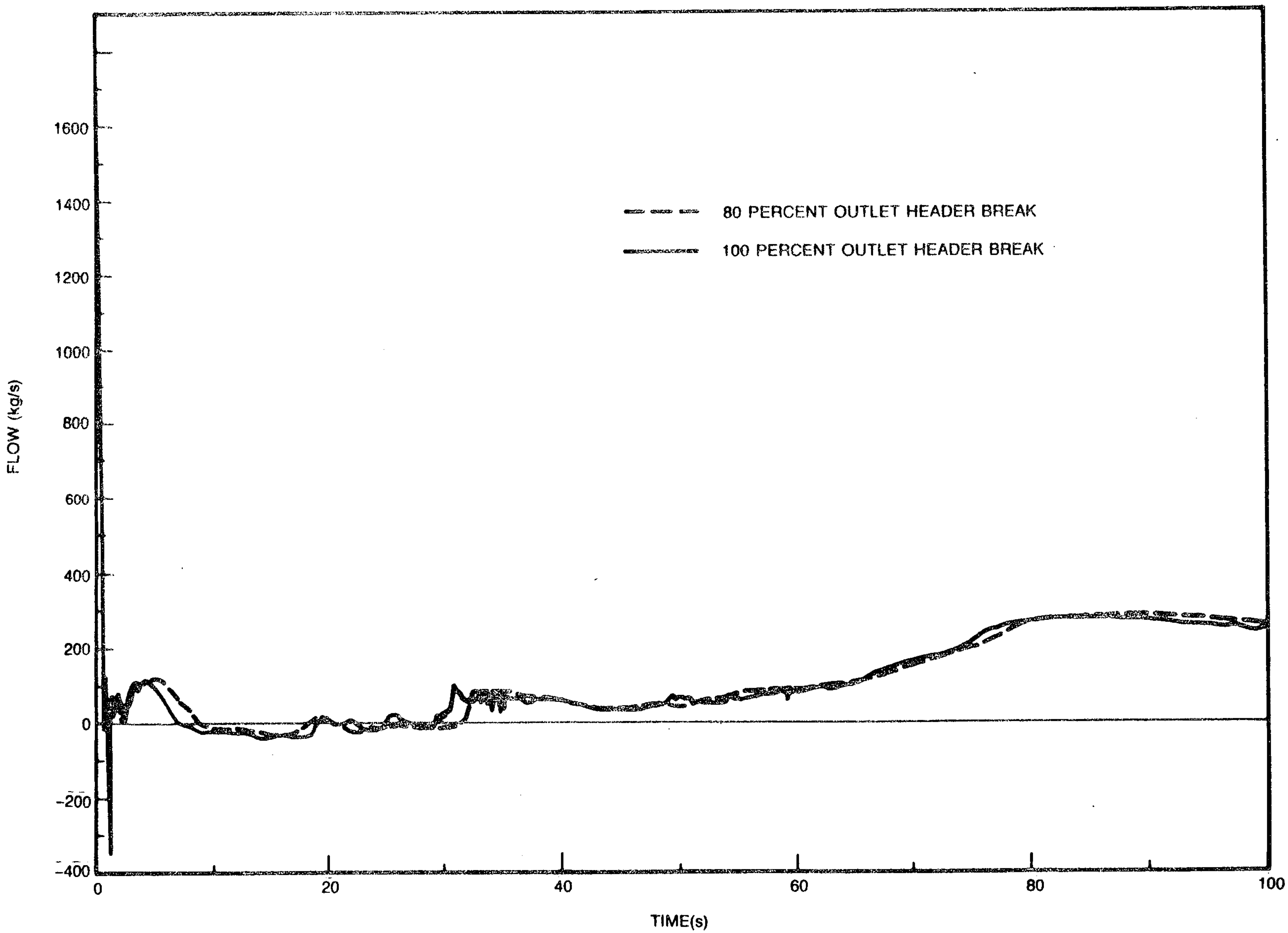


FIGURE 7 FLOW IN CORE PASS DOWNSTREAM OF 80 AND 100 PERCENT OUTLET HEADER BREAKS

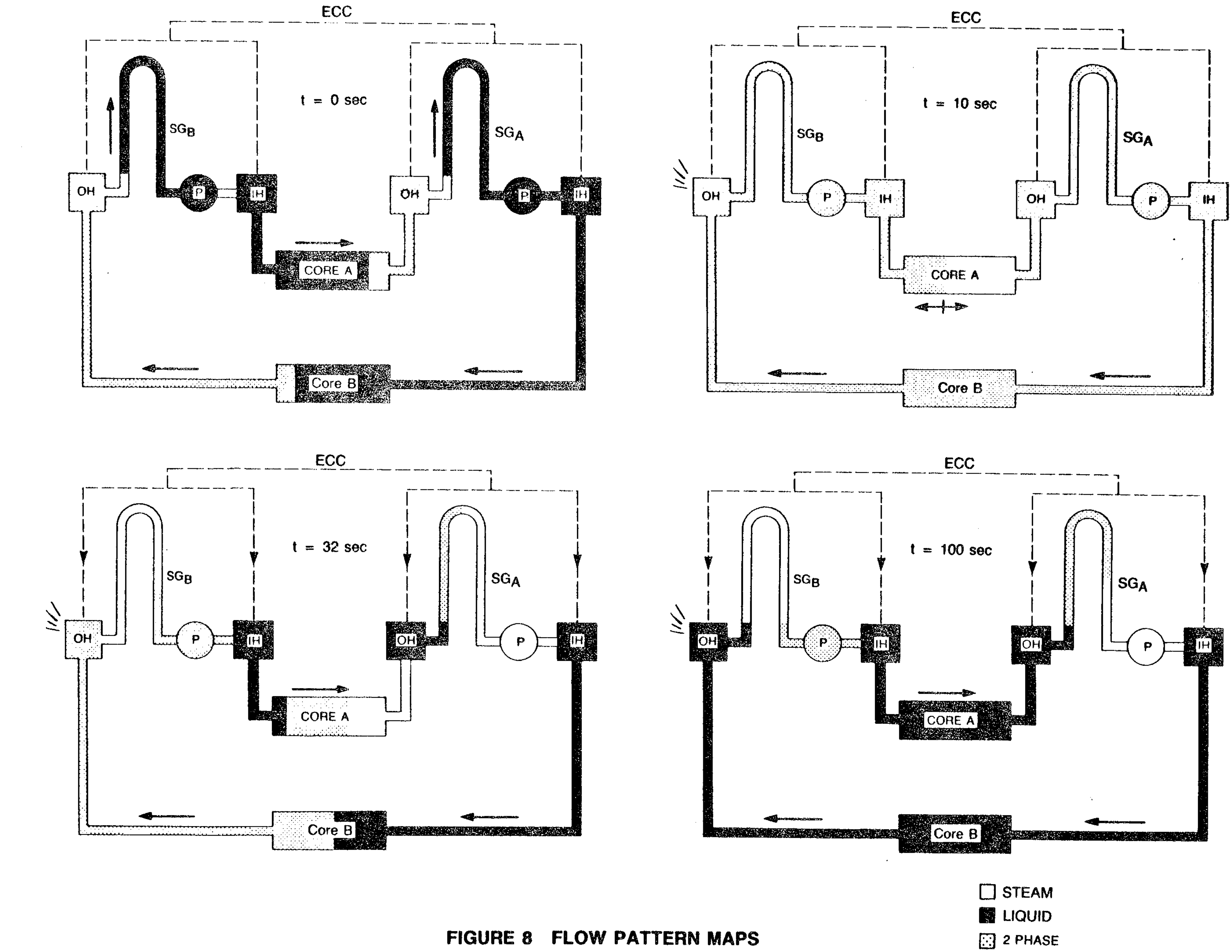


FIGURE 8 FLOW PATTERN MAPS

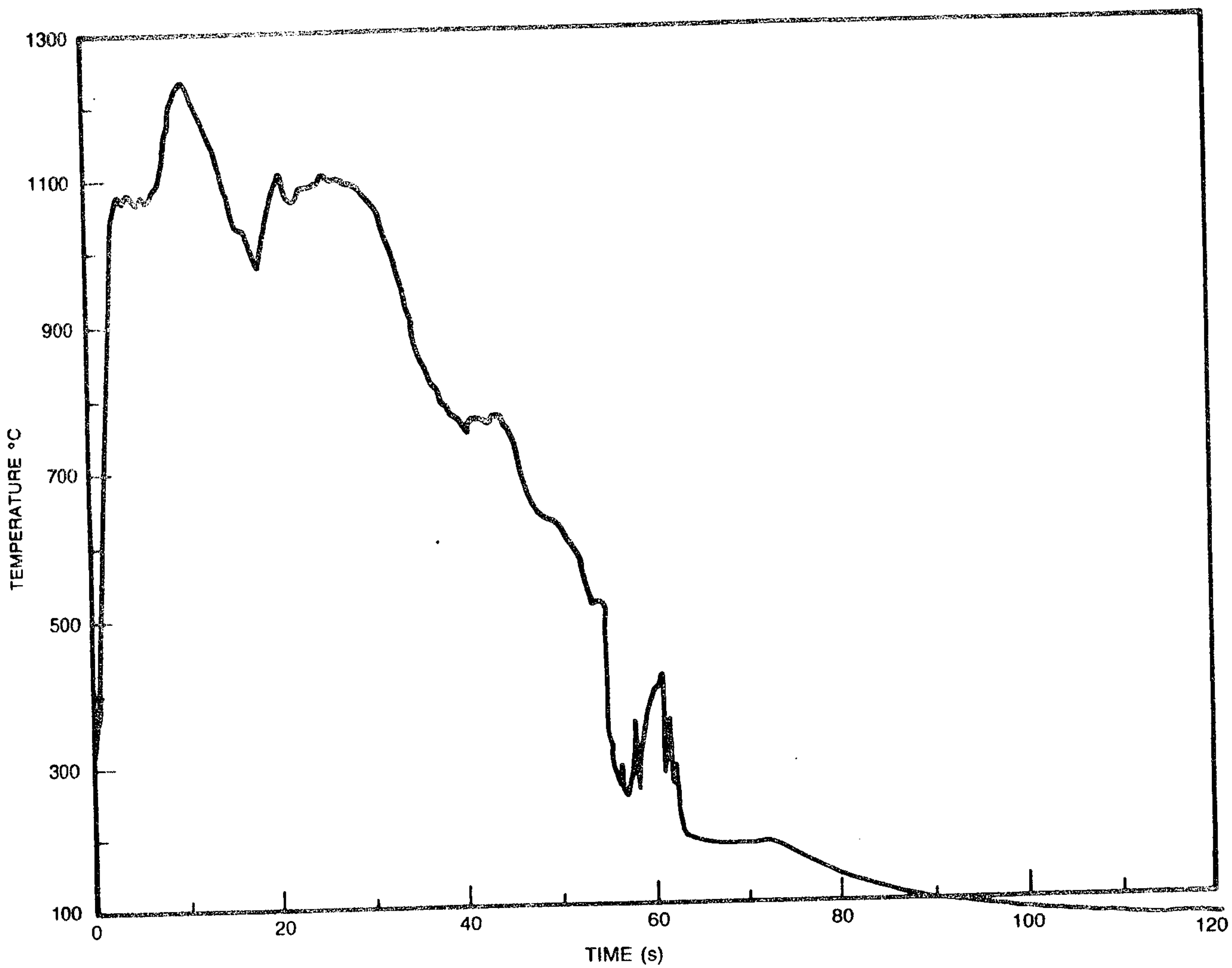


FIGURE 9 HOT PIN SHEATH TEMPERATURE IN THE AVERAGE CHANNEL OF CRITICAL PASS

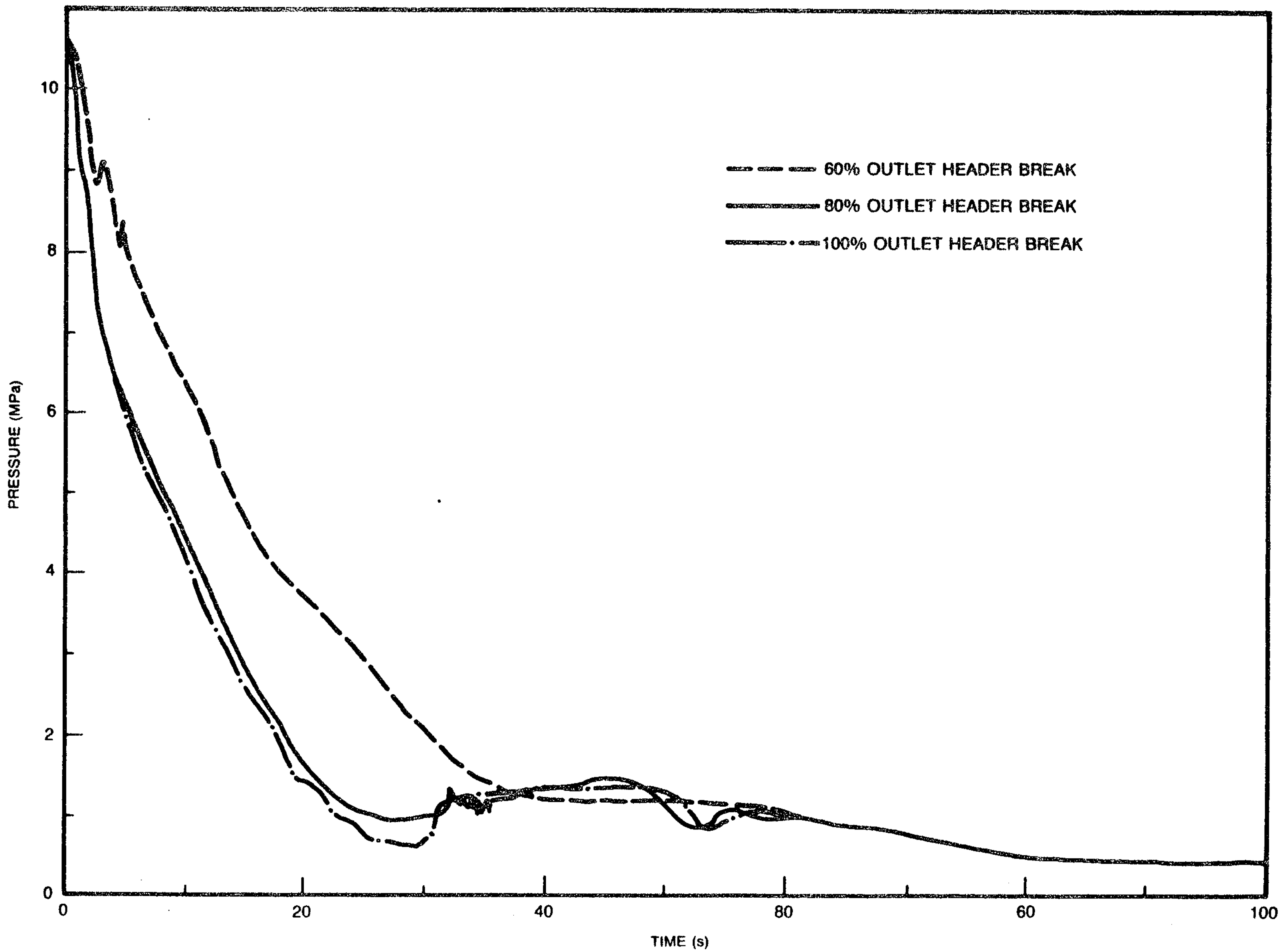


FIGURE 10 600 MW 80% ROH BREAK CENTRAL CORE PRESSURE OF CRITICAL PASS

4 POINT INJE ON

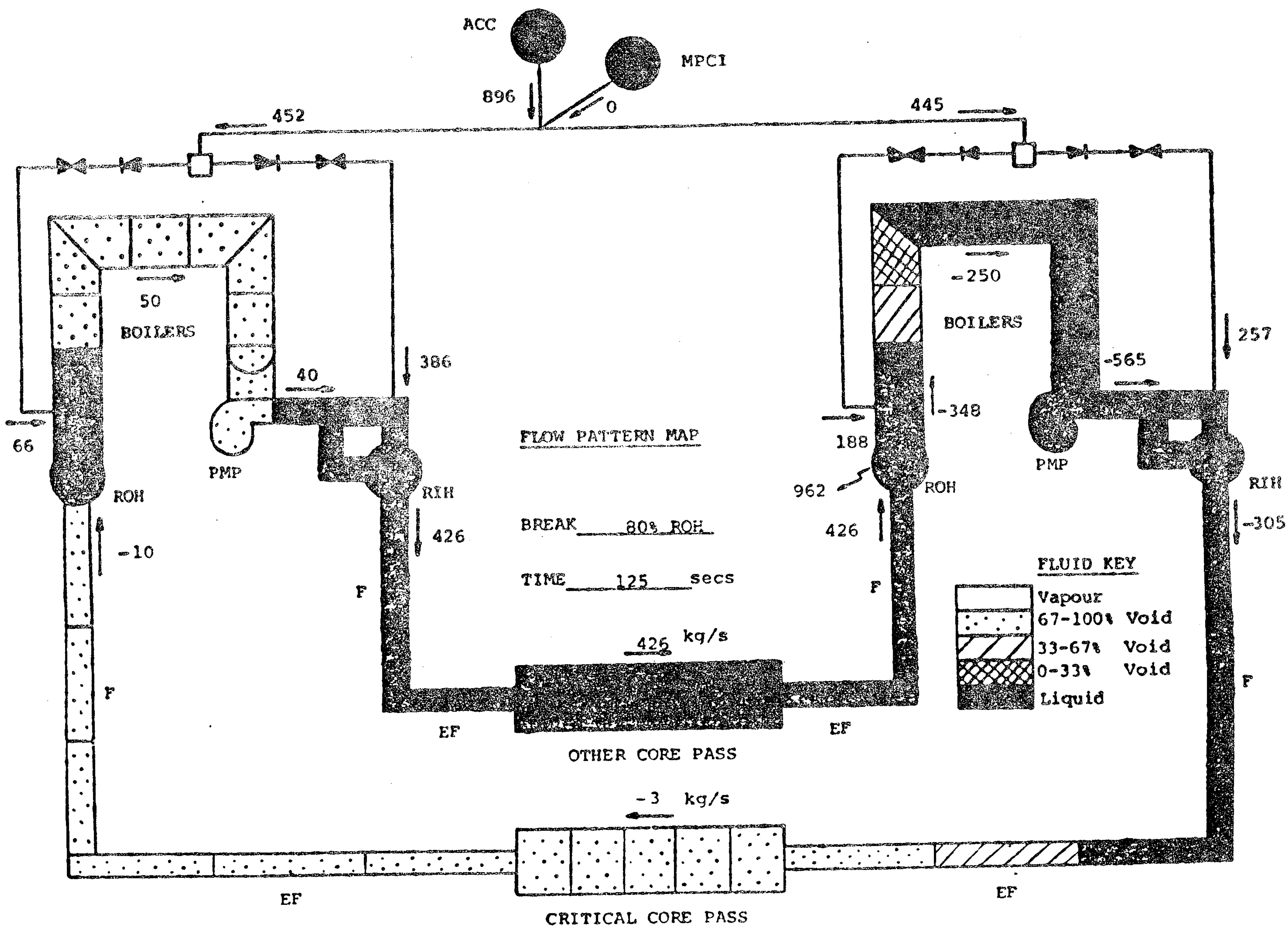


FIGURE 11 4 POINT INJECTION

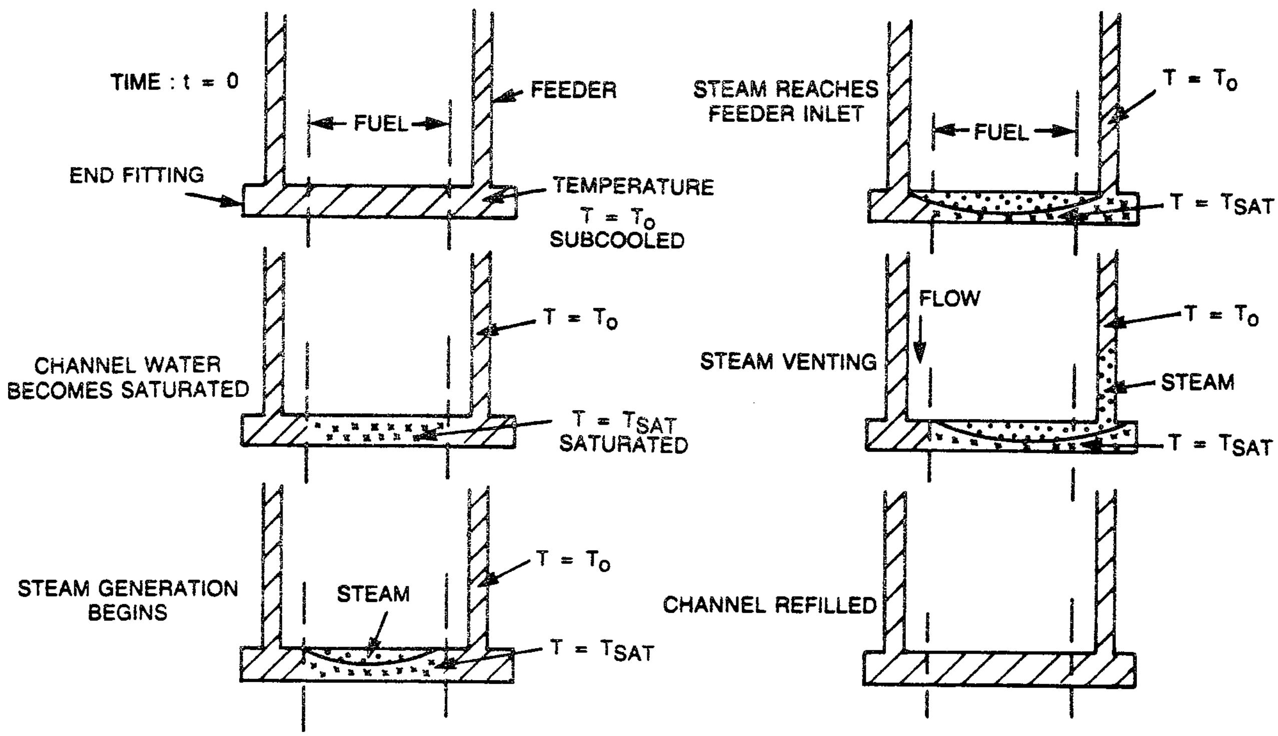
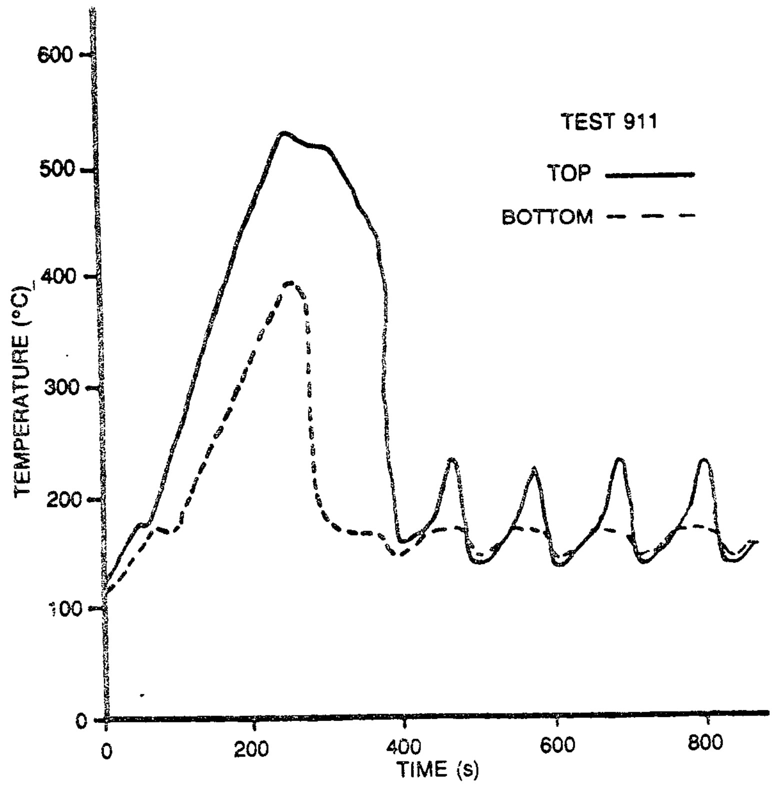
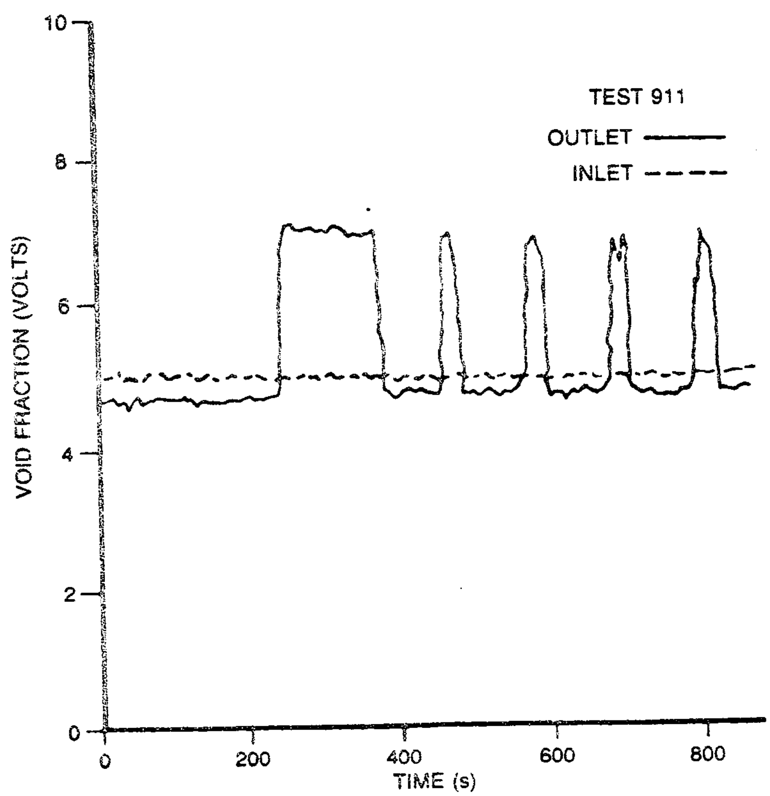


FIGURE 12 CHANNEL-REFILLING SEQUENCE OF EVENTS



TOP AND MIDDLE PIN SURFACE TEMPERATURES



INLET AND OUTLET FEEDER VOID FRACTIONS

FIGURE 13

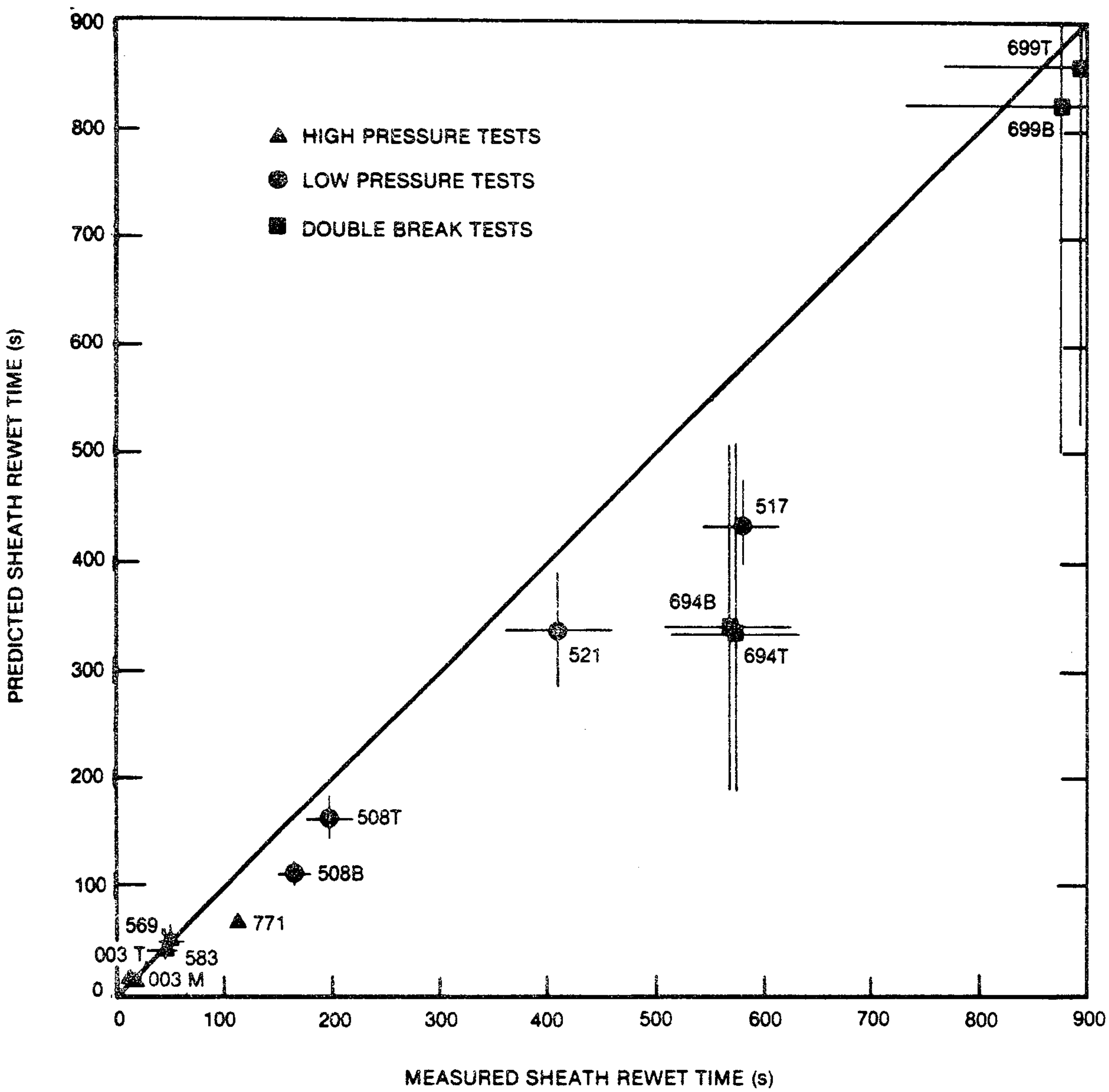


FIGURE 14 COMPARISON OF REWET TIMES FOR TOP ELEMENTS

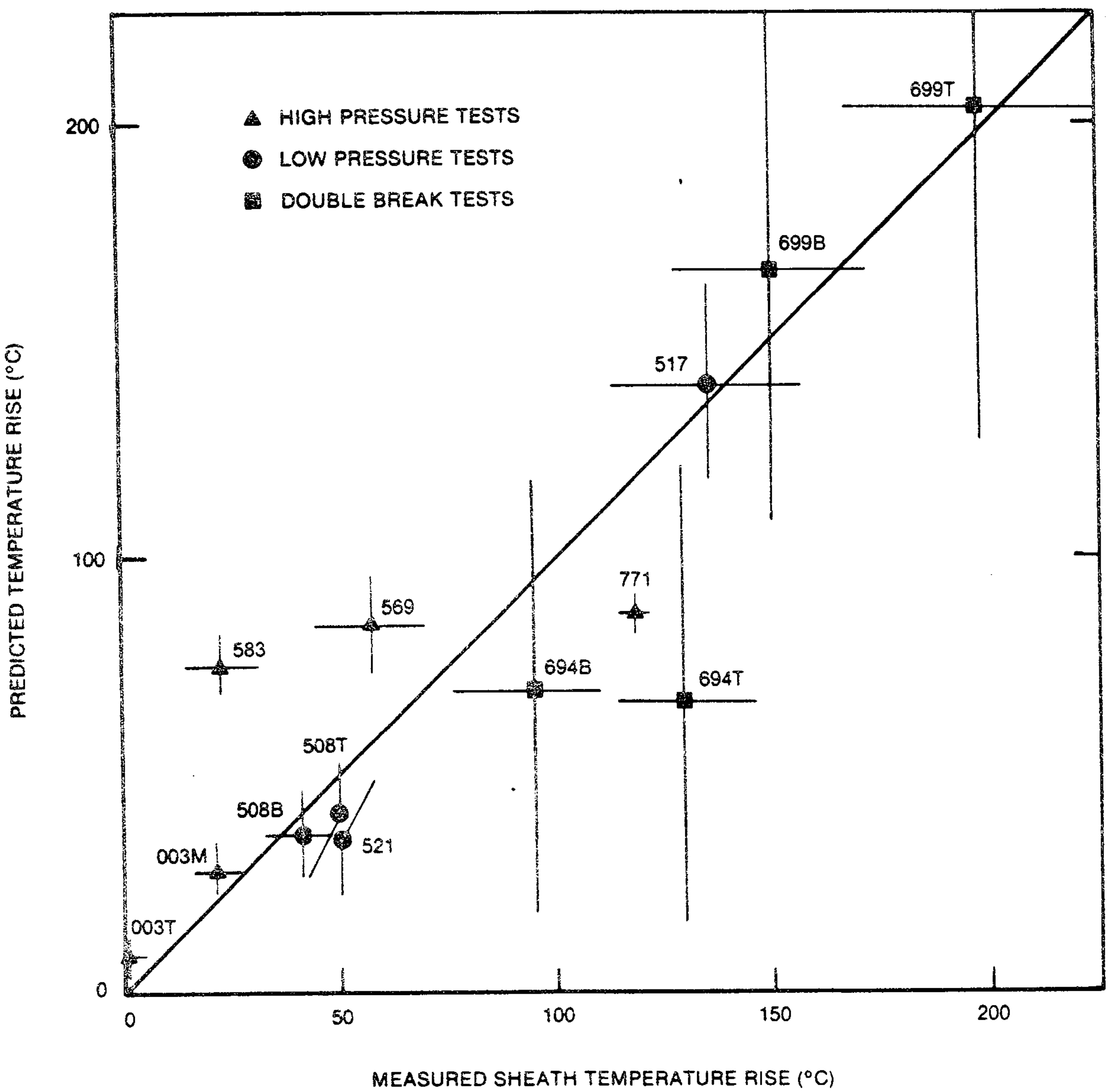


FIGURE 15 COMPARISON OF SHEATH TEMPERATURE RISES

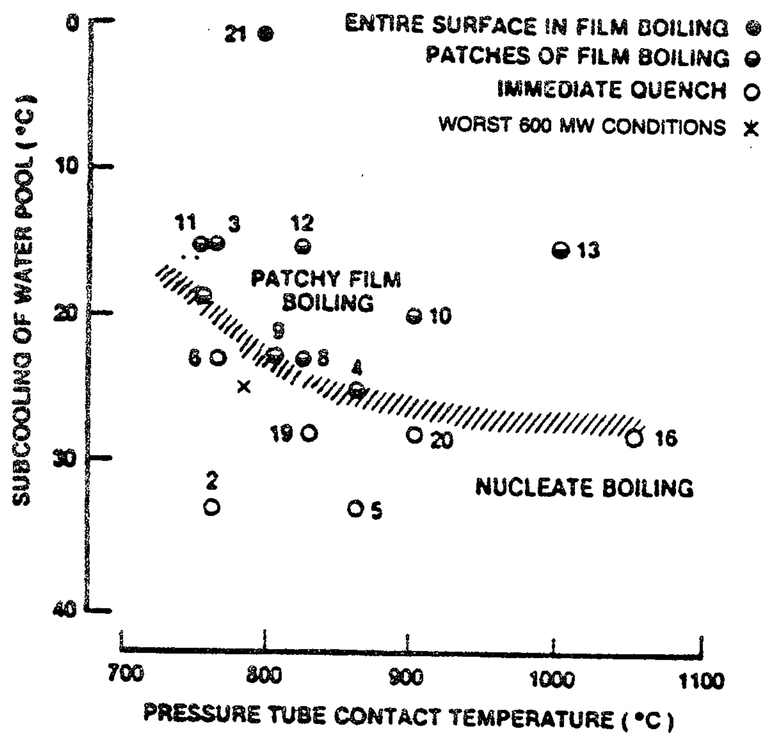


FIGURE 16 RESULTS OF CONTACT BOILING EXPERIMENTS

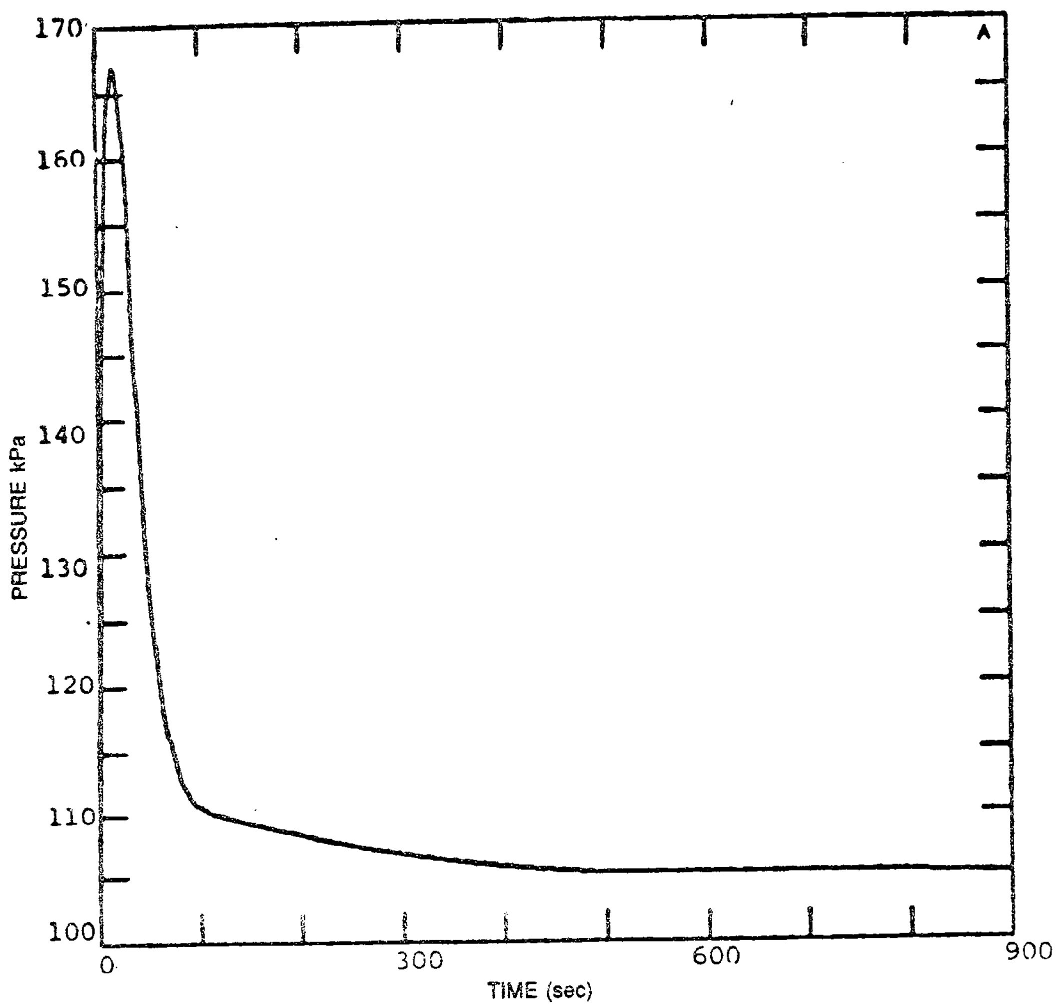


FIGURE 17 CONTAINMENT PRESSURE TRANSIENT FOR 80% ROH BREAK — INTACT CONTAINMENT