FUEL AND FUEL CHANNEL BEHAVIOUR

by

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ABSTRACT

In this paper, the important aspects of fuel and channel behaviour in a large loss-of-coolant (LOCA) accident are discussed. These include the relevant design features of CANDU fuel, the physics of fission product release from the fuel during normal operation, the behaviour of fuel elements under LOCA conditions, effects of sheath strain on overall bundle behaviour, pressure tube deformation, the dynamics of pressure tube/calandria tube contact, and the heat sink capability of the moderator.

For the specific application of a large LOCA in a CANDU, it is concluded that sheath strain is limited by coolant pressure and very few fuel element failures would occur. Whatever sheath strain does occur will not unduly restrict coolant access through the channels. For a limited region of the break spectrum, pressure tube strain will occur, and will provide an additional heat removal path through the moderator.

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16.3.1 INTRODUCTION

This section discusses the important aspects of fuel and channel behaviour in loss of coolant accidents (LOCA). For the purposes of this discussion, I consider an accident in which all safety systems are available and electrical power is also available to power the main pumps. This has been a standard "design basis" for CANDU. Most of the important fission product nuclides are contained within the fuel, therefore, the transient behaviour of the fuel and channel is integral to the release of fission products, and hence, the safety of the reactor system.

The CANDU fuel design is in the form of 0.5 m long bundles of fuel elements (Figure 1). Each bundle is fabricated from seven basic components. The fuel elements contain UO_2 pellets (natural uranium) in a Zircaloy-4 sheath. A graphite layer (CANLUB) on the inside surface of the sheath reduces the effects of pellet-clad interaction (PCI). End caps are resistance welded to the ends of the sheaths to seal the element. End plates are also resistance welded to the end caps to hold the elements in a bundle assembly. Spacer pads are brazed to the elements at their mid-points, to provide the desired inter-element spacings. The bundle is spaced from the pressure tube by bearing pads brazed near the ends and at the mid-point of each outer element. Beryllium metal is alloyed with the Zircaloy to make the braze joints.

The features of this design which are particularly important to accident conditions are:

- thin walled Zircaloy-4 fuel cladding which is designed to collapse under the external coolant pressure. This provides good heat transfer out of the fuel pellet and lowers its operating temperature and the amount of fission product release from the fuel under operating conditions

- the UO_2 pellets are of relatively high conductivity material. Since the density is higher than in other reactor types, its operating temperature is lower.

- the short element is almost completely UO_2 filled such that the gas storage volume is small - typically, about 2 cm^3 cold and a third of that at operating conditions. The effect of this is that the gas pressure can be relieved by a relatively small amount of sheath strain.

- the induction brazing of spacer and bearing pads causes complex metallurgical changes in the fuel sheath at these zones along its length. These must be accounted for in predicting strain behaviour.

- the diffusion of the beryllium brazing alloy at the sheath bearing pad interface causes a failure mechanism at high temperatures (penetration of the alloy through the sheath wall).

- the bundle structure causes a flux depression in the centre, resulting in a large power gradient towards the centre. This means the inner elements have little fission gas and, therefore, will not strain in LOCA's.

These features are all important to the behaviour of the individual elements and, hence, the bundle as a whole. The pressure tube is also important to the accident behaviour. Firstly, it is an important radiative heat sink for the fuel, particularly for the higher powered outer elements. Secondly, the pressure tube may heat up during a severe transient and as a result, either expand diametrally to contact its calandria tube, if the pressure is relatively high (Figure 2a), or sag into contact for lower pressures (Figure 2b). This behaviour, if it occurs, brings in a heat removal path through the moderator.

The following sections will focus on the fission product behaviour within the fuel (Section 2), the failure mechanisms of the fuel sheath in accidents (Section 3), the overall bundle behaviour (Section 4), and the pressure tube behaviour (Section 5).

16.3.2 FISSION PRODUCT BEHAVIOUR IN FUEL

Fission product nuclides are created within the UO_2 fuel matrix during normal operation. There are many nuclides created which have widely varying half lives. Most of the nuclides remain bound within the UO_2 fuel matrix and will not be released in accidents which maintain the fuel at temperatures typical of LOCA's where the emergency coolant injection system functions. Nevertheless, during normal operation a fraction (typically no more than 10%) of the fission products are released out of the UO_2 matrix into the element gas volume or gap. This is known as the gap free inventory.

The physics of release of fission products from the UO_2 matrix into the fuel element space or free gap inventory is complicated. This is shown in Figure 3 and includes the following:

- Creation of the nuclide by fission.

- Diffusion through the UO_2 by either single gas atoms or as bubbles if the temperature for these processes is sufficient ($T \geq 1500^\circ C$).

- The simultaneous decay of the fission products into other nuclides and their subsequent diffusion.

- A process of release directly from the UO_2 matrix into the gap by recoil of the atoms from fission or knockout of the fission fragments - a small contribution to the free inventory.

- Diffusion of nuclide atoms to the grain boundaries, where some precipitate to form gas bubbles.

- The accumulation of fission products allowing the bubbles to grow and interlink.

- Tunnel formation on the grain boundaries which eventually interlink to the free surface of the UO_2 such that the fission nuclides are released. The tunnels are essentially closed bubbles but open when the gas pressure increases. When the gas is released, the tunnel re-closes. This makes the release a series of discrete events.

- The eventual opening up of grain edge tunnels at high burnups such that a continuous path forms between the grain to the gap free volume.

- The restriction of grain boundary bubble growth and tunnel development due to hydrostatic effects caused by fuel expansion and the coolant pressure acting through the sheath to the fuel.

- The migration of grain boundaries causing the grain sizes to increase. This has the effect of increasing the nuclide diffusion distance from the point of birth to the grain boundary. As the grain boundary moves through the UO_2, it sweeps fission products onto the grain boundary. This sweeping effect is only important in situations where the grain boundary movement is relatively fast compared to the diffusional release (i.e., at temperatures $\geq 1500^{\circ}C$).

- Bubble nucleation, growth and movement within the UO_2 matrix.

The UO_2 will behave mainly according to the temperature and the driving forces as they affect the above mechanisms. In normal operation, the fuel establishes a quasi-equilibrium, i.e., the grain structure is developed according to the temperature gradient across the fuel. Fission products are created, and diffuse out of the grains to reach the grain boundary bubbles, where they are eventually released to tunnels and the free element volume. Nuclides with short half lives obviously will rarely make their way to the free element volume. Longer half life nuclides will diffuse further and stable nuclides will have the best chance of making it to the free volume.

We can evaluate the gap free inventory in a reactor core by using the ELESIM fuel model[1,2], coupled with the ANS 5.4 release model[3]. We illustrate the main effects of the gap inventory in the whole core for one particular nuclide, I-131. As discussed above, the main parameter affecting release is the fuel temperature which depends on the element power. From the physics predictions of the element powers and burnups, we obtain the number of elements in a particular power group as shown in Figure 4. We can see a very sharp decline in the number of elements in the core at a power greater than 50 kW/m. Figure 5 shows the I-131 inventory as a function of element power. (For this example, 120 MWh/kgU burnup. The I-131 free inventory is not very burnup dependent because I-131 has a half life of about 8 days which prevents the accumulation of this nuclide over long times.) The element power must be high to achieve a significant release in short times (less than a few weeks). As can be seen from these two figures, the gap free inventory of I-131 in the core will be limited because generally, the element operating powers are lower than that necessary to give significant release to the gap inventory. The cumulative inventory in the core is shown in Figure 6.

The gas release inside the element creates pressure inside the element and is the driving force for sheath strain during LOCA transients. The sheath strain is the main cause of sheath failure. The elements which have a sufficient gas inventory to cause failure (more about this later), will be the elements which release the important nuclides like I-131. Note that high gas release elements do not necessarily mean that they also contain a high inventory of I-131.

Element failure does not mean that fission products are immediately released from it. The rate of release depends on the nuclide. Gaseous nuclides such as Kryton and Xenons will mostly be released on failure. Other elements are less volatile, particuarly if they are chemically combined. Iodine can be expected to combine with Cesium. This compound is much less volatile and a time dependent release out of the element can be expected.

16.3.3 FUEL ELEMENT BEHAVIOUR

The response of the fuel element in accidents is evaluated by first considering the heat transfer conditions from the fuel sheath to the coolant. Any detailed analysis is, of course, tied to the specific accident under consideration. We have chosen for illustrative purposes, the large loss of coolant accident.

Flow stagnation after a LOCA requires a specific break size and location. The duration of low flow depends on the reactor circuit but breaks can be found for which fuel elements are in a steam environment for a minute or so. The initial heat up phase is mainly due to a redistribution of the stored energy in the fuel. The heat up rate of the fuel sheath is very rapid during this phase, $100^{\circ}C/s$ to $300^{\circ}C/s$. The temperature rise terminates in a plateau or slow heatup period where the steam cooling removes a significant fraction of the decay power generated within the fuel. This plateau period continues until the emergency coolant injection turns the temperature around and cools the fuel - usually within a minute. Figure 7 shows a typical transient. For most of the core pass downstream of a break in the CANDU, we have found that under these conditions, fuel failures would be primarily from the incipient defects which were in the core prior to the accident.

The reason for the above is that most of the elements in the core have relatively low powers, as shown in Figure 8. This limits both the gas release and the plateau temperature reached in a LOCA. Let us first assume the sheath has no strength at all. Figure 8 shows the number of elements in the core which have a gas release above a certain value. Remember, the gas inside the element is the driving force for sheath strain. If the coolant pressure during the transient is about 1 MPa, the gas pressure inside the element must be greater than this value for strain to occur at all. For this LOCA transient, the volume in the element must be greater than 6000 mm^3 for strain to occur. We can see from Figure 8 that 98.5% of the elements cannot exceed the pressure of the coolant and, thus, will not fail. There is no known failure mechanism which could be expected in these transients if the gas pressure is not exceeded.

- stress corrosion cracking can also result from the corrosive fission products (e.g., Iodine). Sheath strains greater than 10% are required.

- oxygen embrittlement which is caused by diffusion of oxygen into the fuel sheath. Typically, more than 1000 seconds is required at 1200°C before this occurs. Thus, for transients which are terminated by emergency coolant injection, this mechanism will not cause failure.

The strain of the fuel sheath is an important method of accommodating the fission gas during the transient, and thereby limiting the number of fuel failures and the activity release. The fabrication of the fuel element results in three braze heat affected zones in the sheath. Metallurgically, these zones have large grain sizes which result in a lowering of the tensile strength but increasing the creep resistance of the material as compared with the as-received fuel sheath material. The as-received zircaloy is a high strength, partially recrystallized material. Thus, we have a small zone of annealed material of lower strength surrounded by two zones of higher strength. These differences are significant only in accidents which result in sheath temperatures below 1000°C. Above this temperature, all the zones become similar to the braze zone material. Below this temperature, the material differences do affect the sheath strain profile and are considered in assessing element-to-element interaction.

16.3.4 BUNDLE BEHAVIOUR

The factors which are important in determining bundle behaviour are: the strain response of the elements as it might affect the bundle behaviour, and the effect the bundle response has on the element.

The power distribution in the bundle is the single most important factor affecting the stored heat and the resultant initial heat up behaviour. Figure 10 shows the relative powers of elements in each ring of a 37-element bundle as a function of bundle burnup. It can be seen that the outer elements operate at a much higher power than the other elements. Remember, the fission gas release is a strong function of power so that even

The elements which do have gas pressures or volumes high enough to strain in a stagnation break behave as shown in Figures 9a and 9b as follows:

- the coolant pressure drops and the sheath temperature increases. At some time, the pressure inside the element will exceed the coolant pressure.

- depending on the sheath temperature, the sheath will strain diametrally

- any sheath strain dramatically increases the element volume, and so reduces the gas pressure within the element

- the coolant pressure may fall further, maintaining a differential pressure between the inside of the fuel element

- the strain in the sheath is thus controlled by the falling coolant pressure and the sheath temperature (which determines the sheath strength)

- if the gas volume inside the element is sufficient, the strain on the sheath may reach a point somewhere along its length where it fails. We have typically used a uniform strain criterion (e.g., 5%) below which, under non oxidizing conditions, ballooning and failure do not occur.

- in addition, other failure mechanisms are known to be imporant:

    - beryllium braze penetration (this is a small localized failure which occurs where the beryllium braze alloy is in contact with the fuel sheath). It requries both high temperature and a sheath hoop stress.

    - oxidation cracking can begin when there is a strain of about 1.8% on the oxide and eventually the crack propagates through by localized strain.

a very high powered bundle, which has a high gas release on the outer elements, will have virtually no gas in the other elements. For example, if the outer element were at a power of 55 kW/m and a burnup of 140 MWh/kgU, (much higher than any bundles in a 600 MWe reactor), it would have an outer element gas release of 16900 mm^3. The next highest power elements are the intermediate ring which would only have a power of 45 kW/m and, hence, a gas release of 3700 mm^3, This means that in CANDU bundles, only the outer elements are likely to strain at all during a loss of coolant accident. This is important because it means that coolant will always be able to flow through the central part of the bundles through the channels.

16.3.5 CHANNEL BEHAVIOUR

Generally, if the LOCA results in moderate pressure tube temperatures - less than about 650 ^{o}C, the pressure tube will not deform. Figure 11 shows the pressure tube deformation characteristics as a function of temperature and stress. Thus, considering the pressure and temperature transient for a LOCA, we can determine if the pressure tube will strain significantly. If it does strain into contact with the calandria tube, it will cool and regain strength. Generally speaking, most loss of coolant accidents will not result in pressure tube strain into contact with the calandria tube. However, for certain scenarios, it may occur.

Pressure tube strain, up to contact, is also affected by temperature gradients around the tube. The behaviour of the pressure tube with thermal gradients has been experimentally examined (at AECL's Whiteshell Nuclear Research Establishment (WNRE). The behaviour follows the expected material property behaviour so that we can use simple computer codes to estimate the amount of strain around the tube and whether the hot area of the tube will strain to the point where it might fail. Figure 12 shows the predicted failure strain decreases with pressure.

Channel integrity is assumed if the calandria tube is kept cool. When the pressure tube strains into contact with its calandria tube, the heat capacity of the pressure tube causes a high transient heat flux through the calandria tube and into the moderator. This behaviour has been investigated at WNRE. For conditions of interest, the calandria tube won't dryout and the stored heat of the pressure tube and calandria tube will be rapidly removed by nucleate boiling on the surface of the calandria tube. This is evaluated by assessing the conditions necessary to cause dryout. The heat flux on the calandria tube/moderator interface depends on:

- the pressure tube temperature at the time of contact
- the contact conductance between the pressure tube and calandria tube, and
- the moderator subcooling - its temperature and pressure within the moderator.

As discussed above, we can estimate the pressure tube strain behaviour and, hence, its temperature at contact as shown in Figure 13. The contact conductance is inferred from the experimental results - shown in Figure 14 to be about up to 11 kW/m^2C. This value is somewhat lower than would be predicted by considering the surface roughness and the contact pressure.

The moderator subcooling affects the critical heat flux at the surface of the calandria tube and is simply related to the boiling curve (Figure 15).

The heat removed from the channel by the moderator lowers the fuel temperatures by increasing the radiative heat flux to the pressure tube. A second important effect of pressure tube strain is the heat removed from the coolant through this new path. The fact that the pressure tube has strained also greatly increases the coolant flow area around the bundle and lowers the hydraulic resistance to flow.

16.3.6 SUMMARY

CANDU fuel is tolerant to loss of coolant accidents because most elements operate at temperatures which prevent significant gas release. Some fuel elements will operate at high enough power conditions that failure might be expected. The strain behaviour of the high power elements within a bundle and channel configuration will allow continued coolant access through the channel. Pressure tube diametral strain can provide an additional heat removal path prior to refill of the channel.

16.3.7 REFERENCES

1. M.J.F. Notley, "ELESIM: A Computer Code for Predicting the Performance of Nuclear Fuel Elements", Nuclear Technology 44 (1979)445.

2. M.J. F. Notley and I.J. Hastings, "A Microstructure Dependent Model for Fission Product Gas Release and Swelling in UO_2 Fuel", Atomic Energy of Canada Limited Report, AECL-5838, 1979 June.

3. "Background and Derivation of ANS5.4 Standard Fission Product Release Model", American Nuclear Society Working Group 5.4, NUREG/CR-2507, (1982).

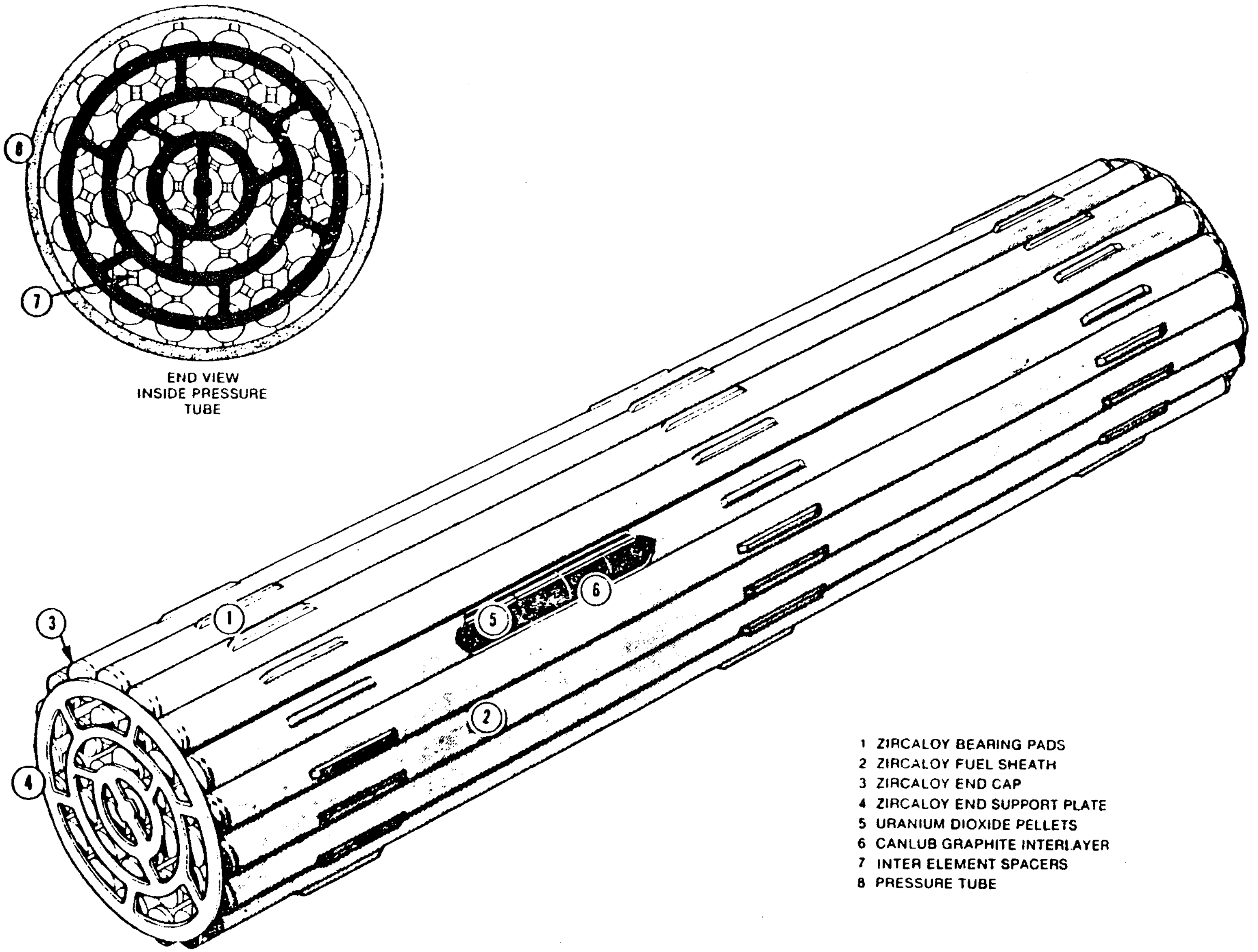


FIGURE 1 37-ELEMENT FUEL BUNDLE

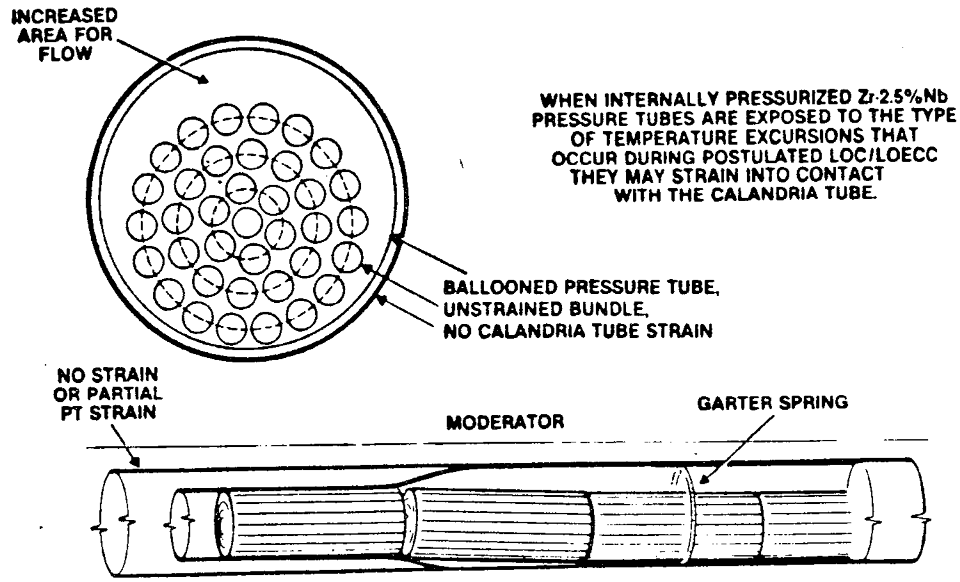


FIGURE 2a PRESSURE TUBE DIAMETRAL STRAIN

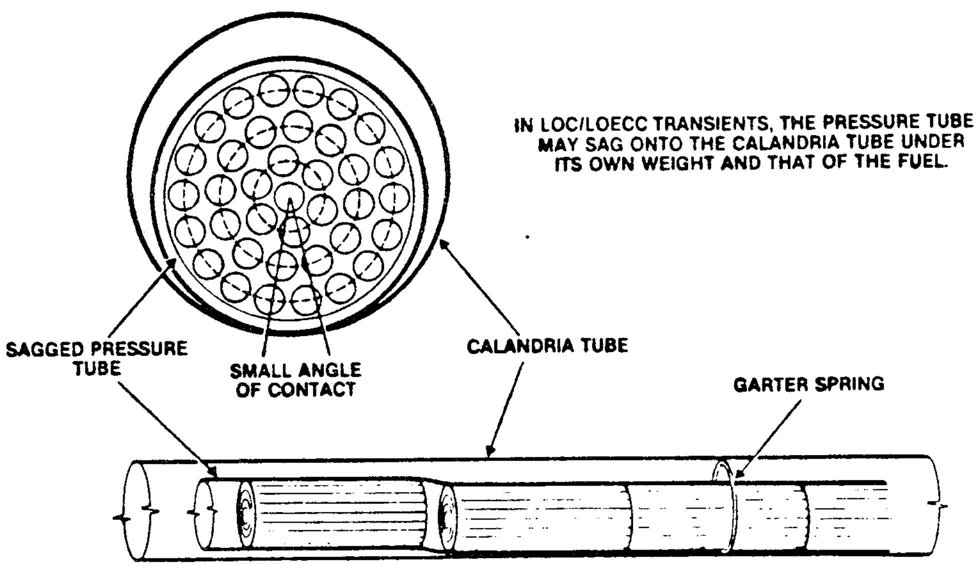


FIGURE 2b DIAGRAM OF CHANNEL IN THE CASE OF PRESSURE TUBE SAG

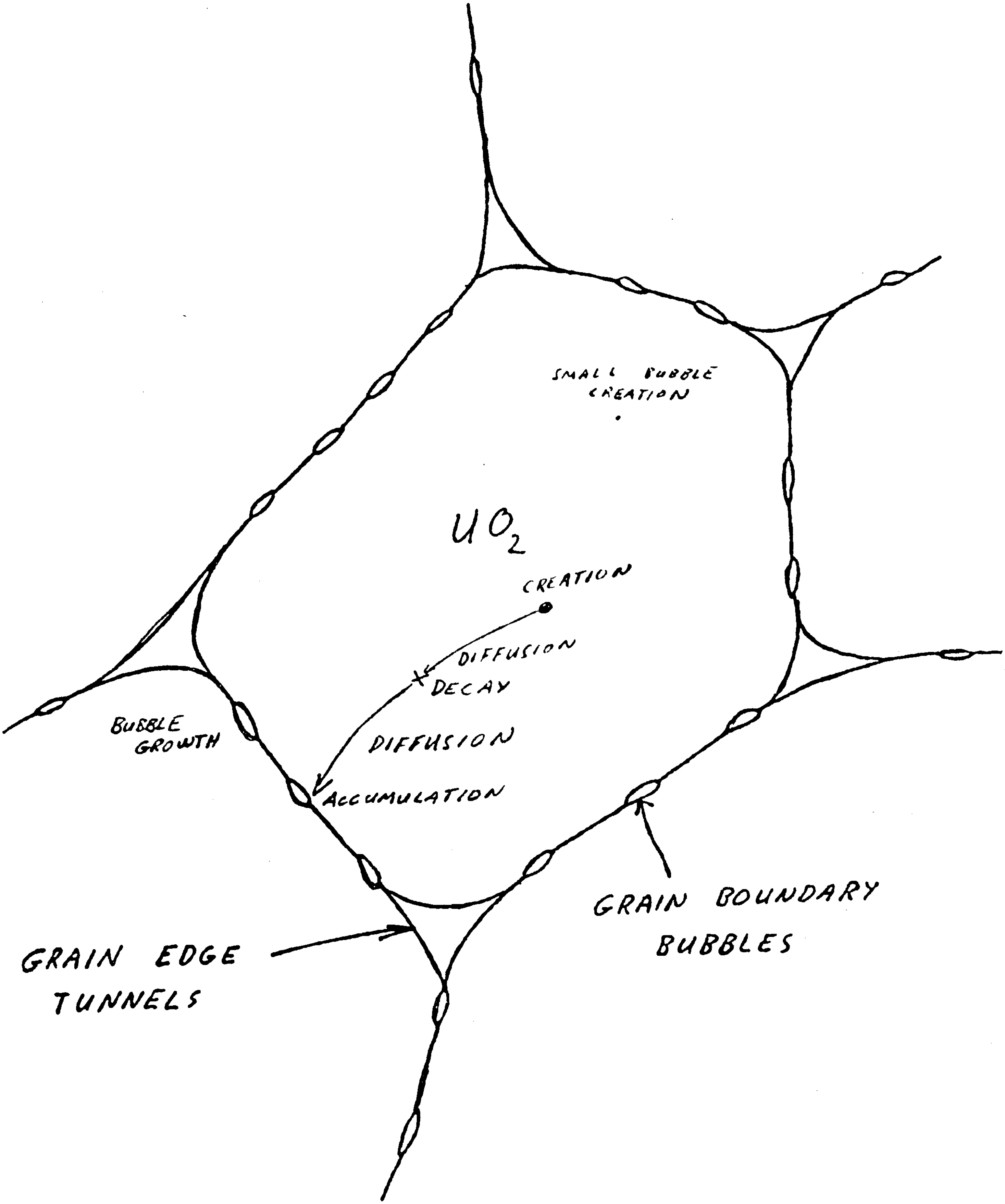


FIGURE 3 FISSION PRODUCT RELEASE PROCESS

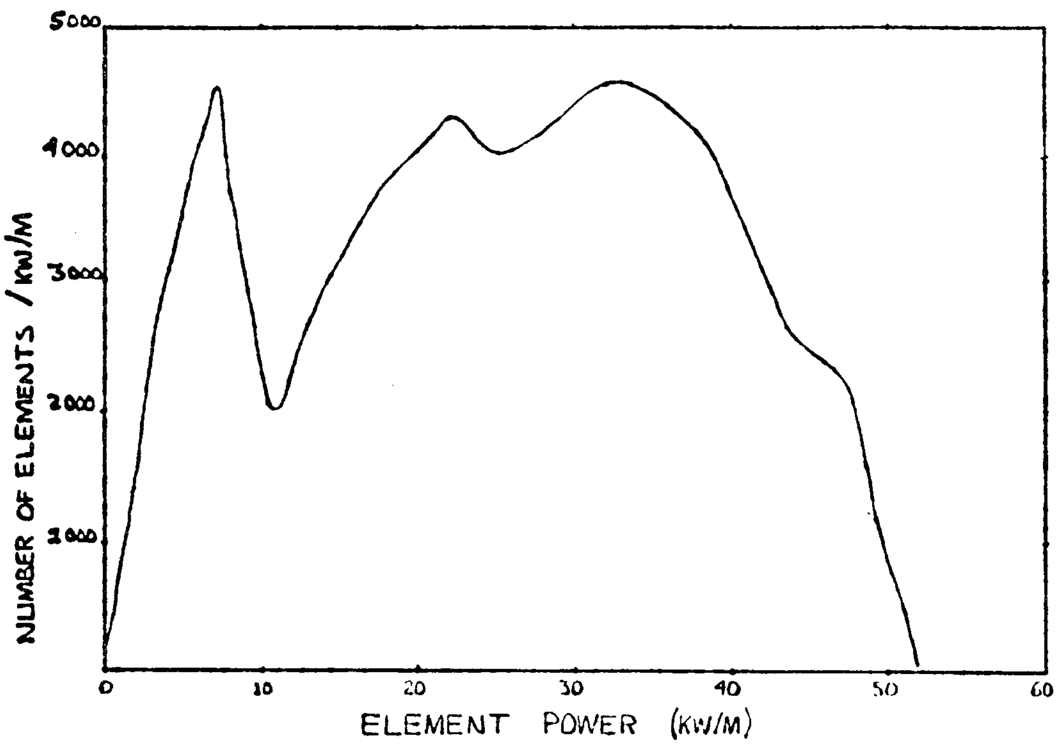


FIGURE 4   NUMBER OF ELEMENTS IN CORE AS A
FUNCTION OF ELEMENT POWER

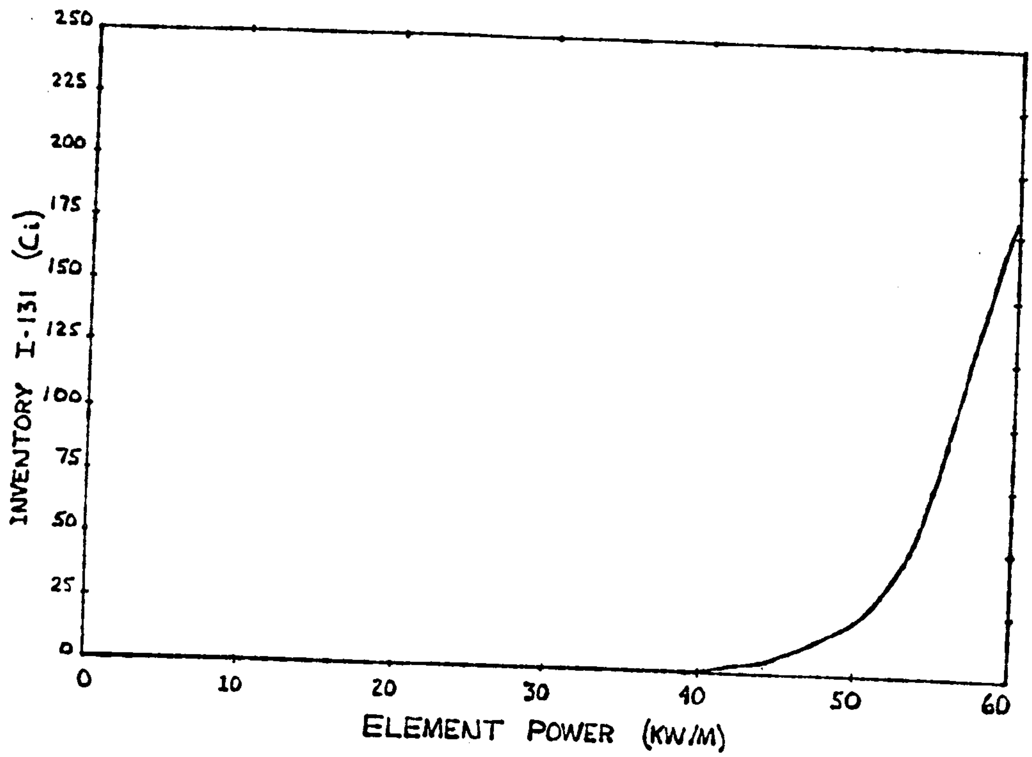


FIGURE 5: ELEMENT FREE GAP INVENTORY VERSUS
ELEMENT POWER

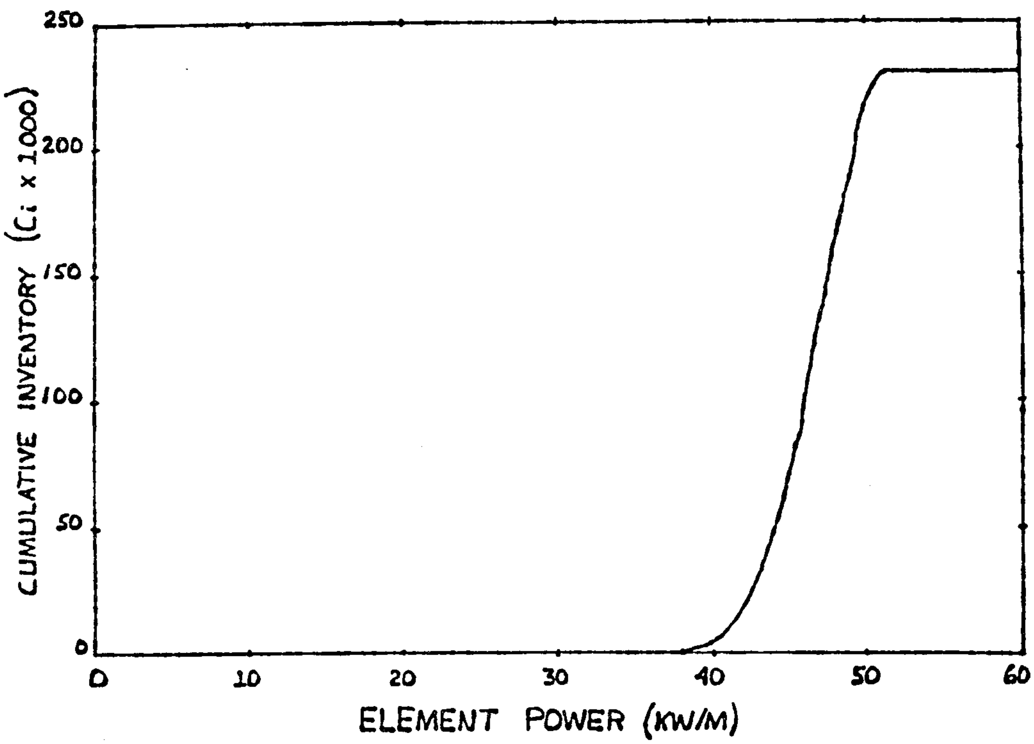


FIGURE 6: CUMMULATIVE GAP FREE INVENTORY OF I-131 IN CORE AT POWERS LESS THAN THE GIVEN VALUE

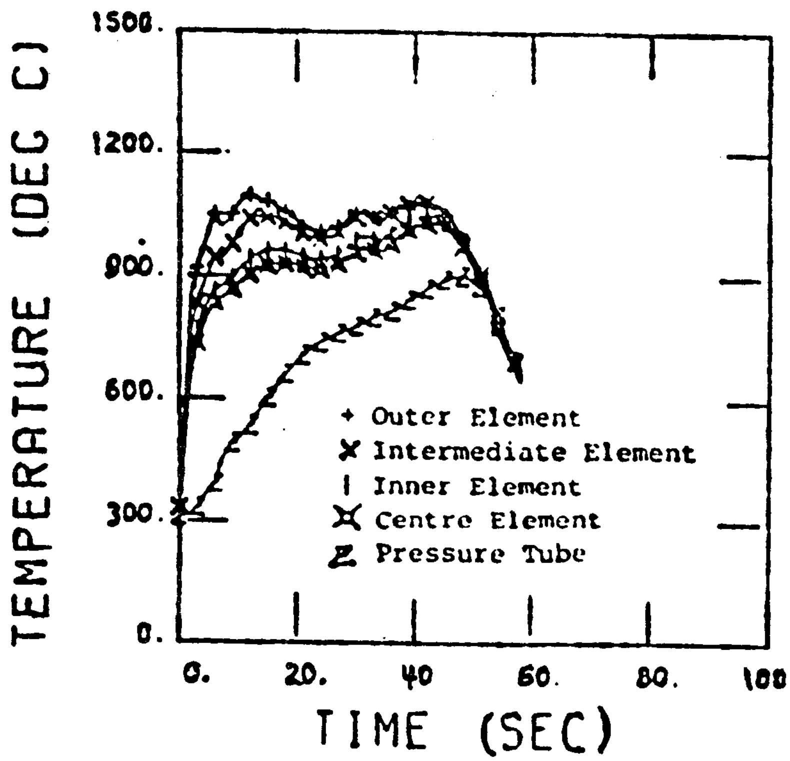


FIGURE 7 TYPICAL "CANDU" FUEL CLADDING TEMPERATURES FOLLOWING CRITICAL LOCA

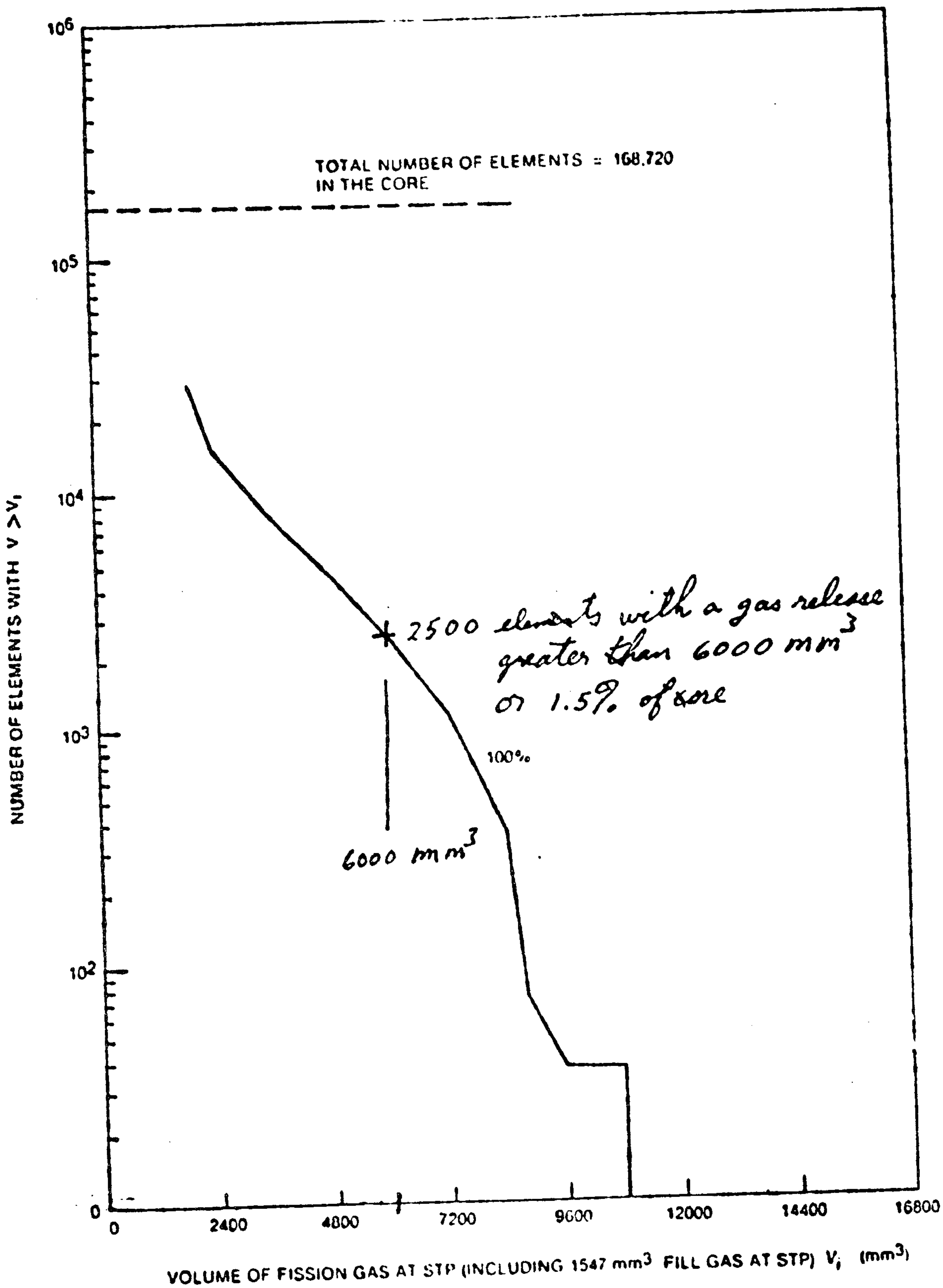


FIGURE 8 NUMBER OF ELEMENTS WITH A VOLUME OF FISSION GAS GREATER THAN V_i

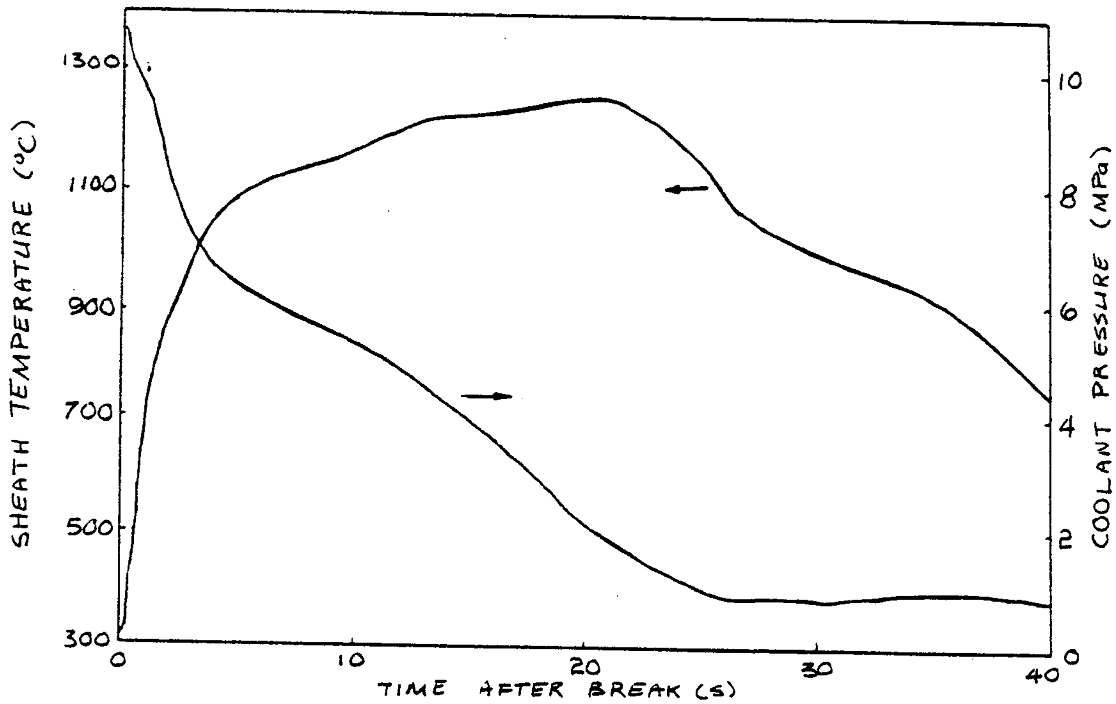


FIGURE 9a: TYPICAL SHEATH TEMPERATURE AND COOLANT PRESSURE TRANSIENTS FOR A LARGE STAGNATION BREAK LOCA

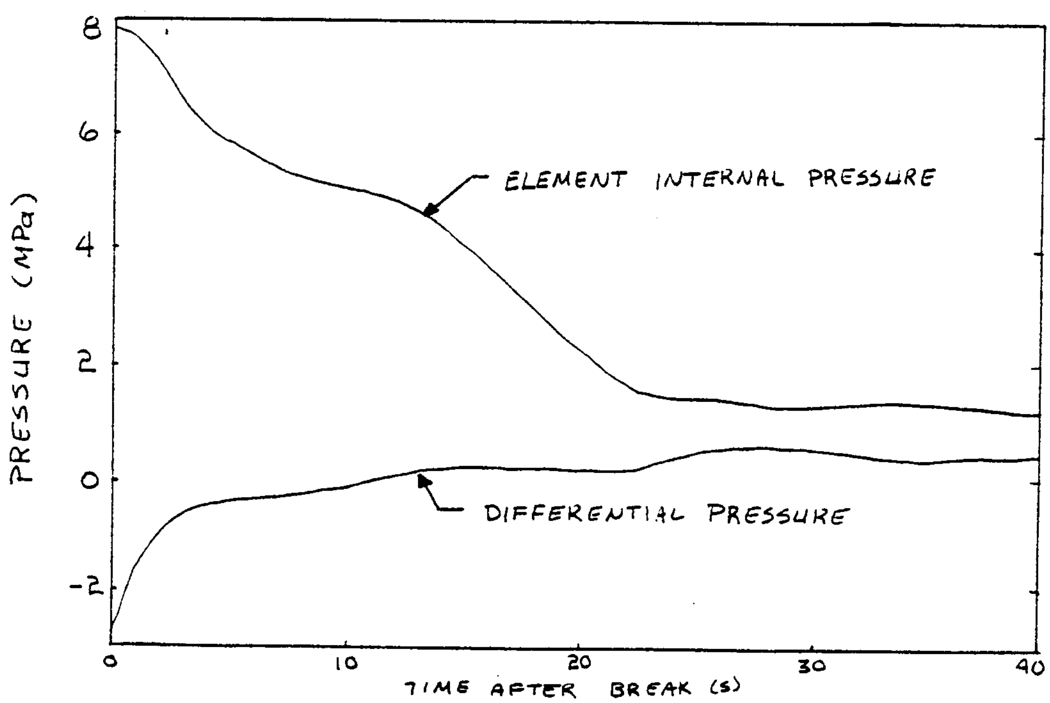


FIGURE 9b: ELEMENT INTERNAL PRESSURE AND DIFFERENTIAL PRESSURE TRANSIENTS FOR CONDITIONS SHOWN IN FIGURE 9a

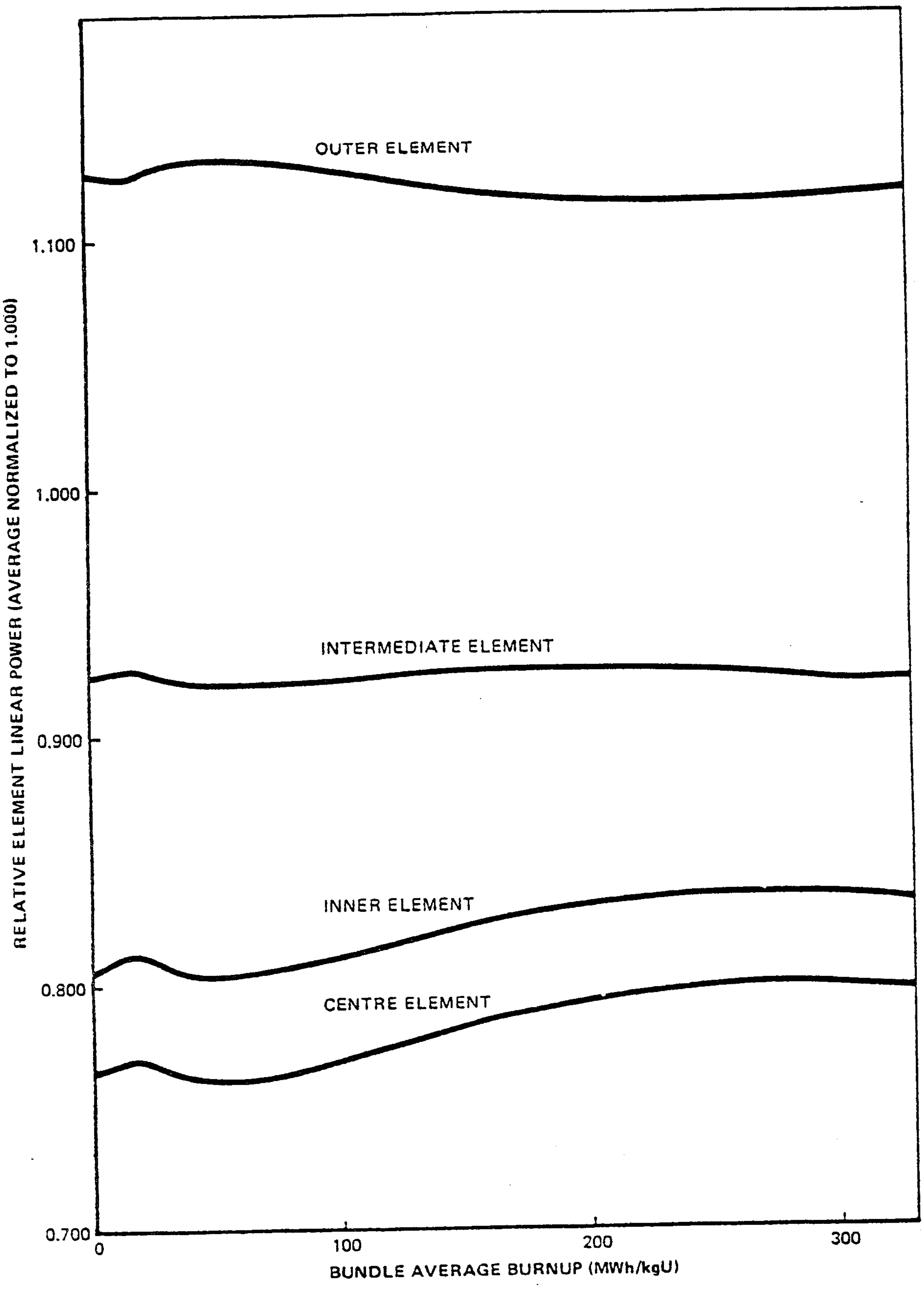


FIGURE **10** VARIATION OF RELATIVE ELEMENT LINEAR POWER WITH BUNDLE AVERAGE BURNUP IN CONSTANT FLUX

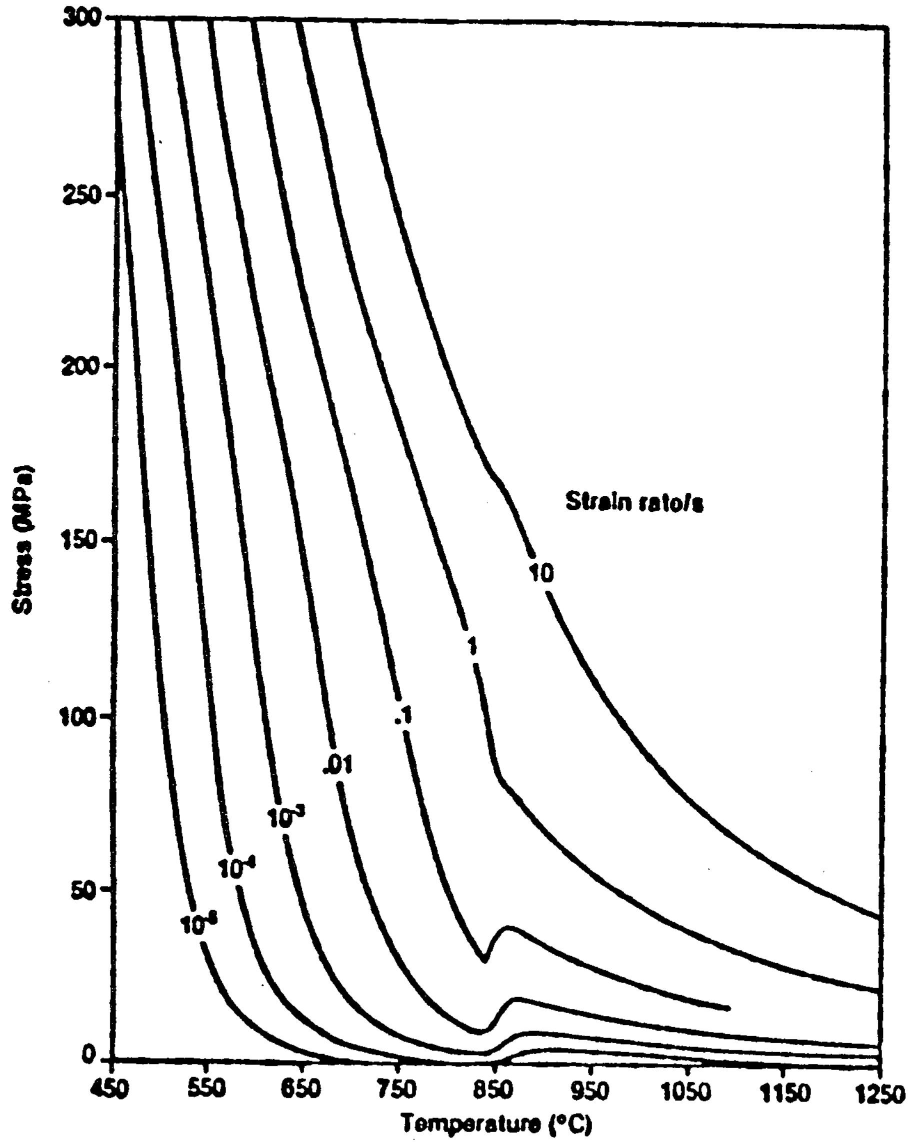


FIGURE 11 PRESSURE TUBE STRAIN RATE AS A FUNCTION OF STRESS AND TEMPERATURE

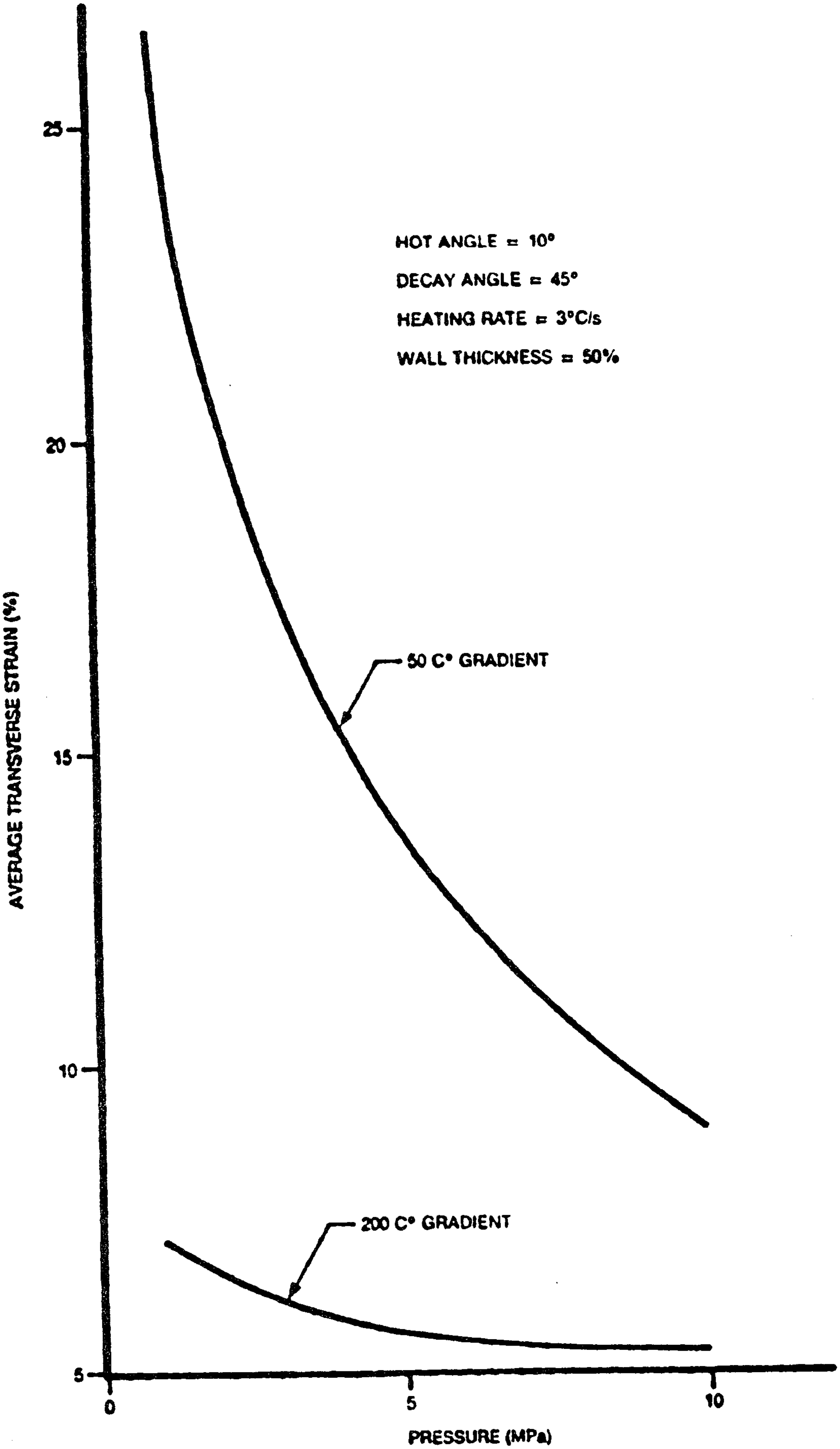


FIGURE 12 EFFECT OF PRESSURE ON AVERAGE TRANSVERSE STRAIN

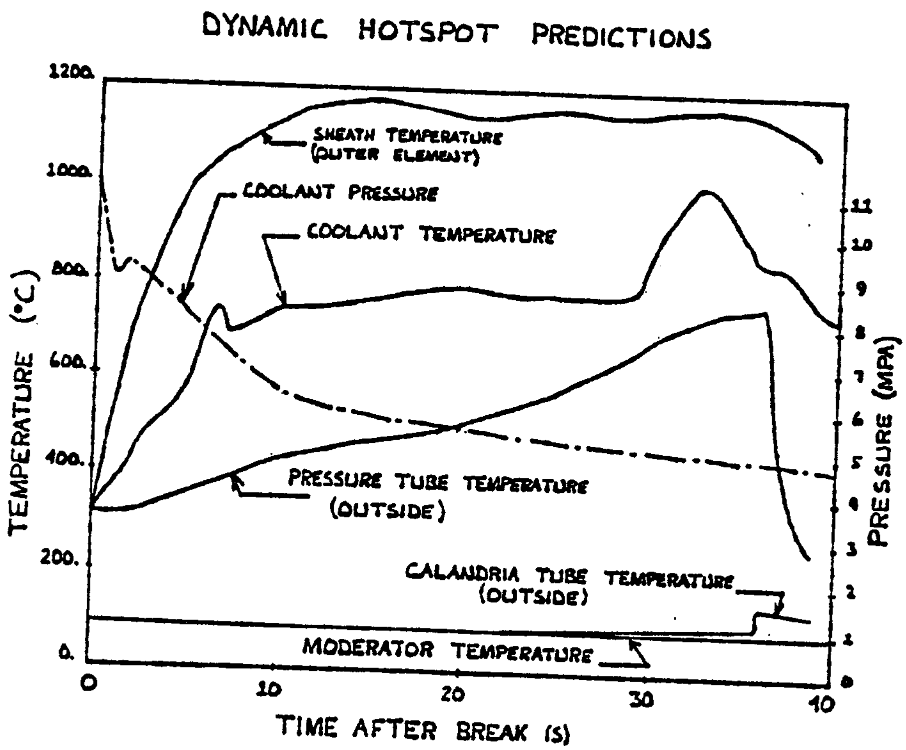


FIGURE 13: LOCA TRANSIENT PREDICTIONS IN
HOT NODE OF CHANNEL

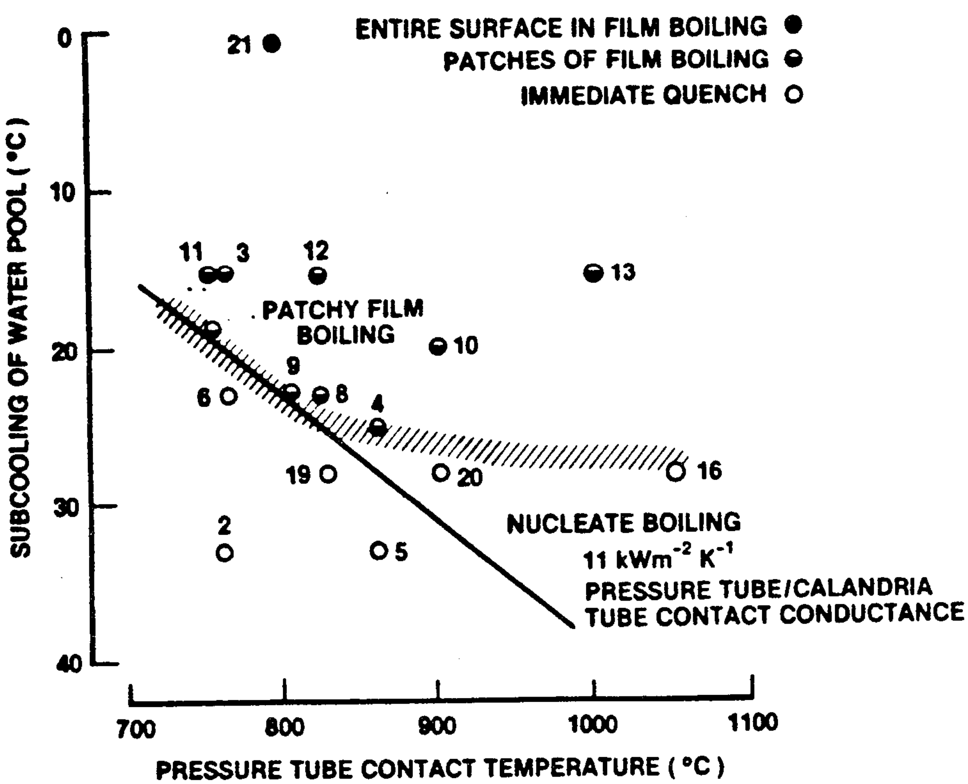
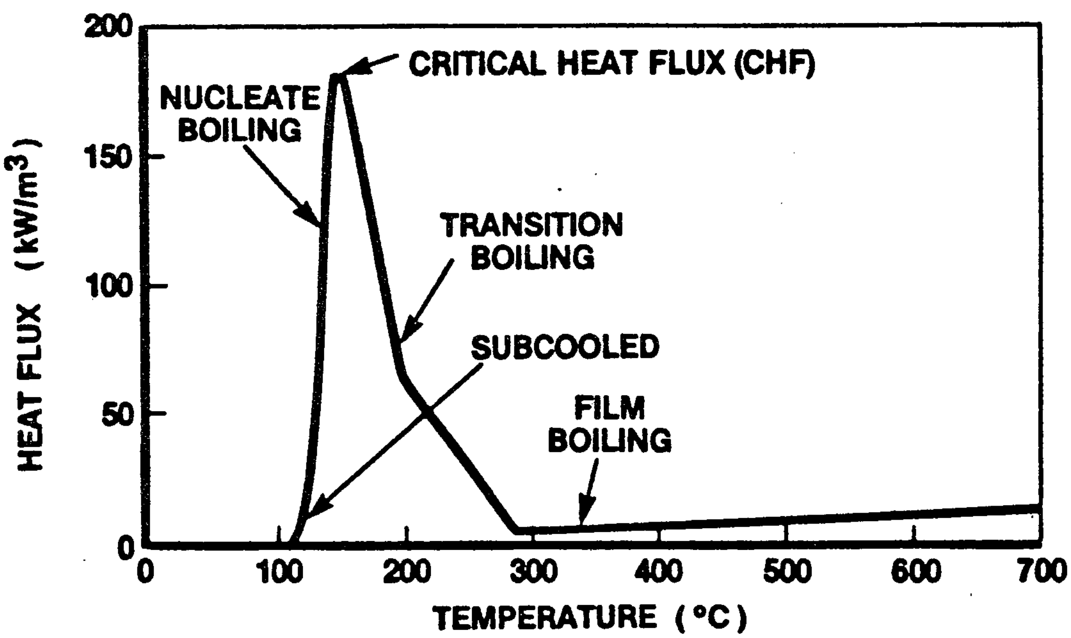


FIGURE 14 RESULTS OF CONTACT BOILING EXPERIMENTS



Based on: T_{MOD} = 90°C T_{SAT} = 103°C

FIGURE 15 BOILING CURVE