ATOMIC ENERGY OF CANADA LIMITED
Power Projects, Sheridan Park, Ontario

Lecture 8

Nuclear power symposium

SYSTEMS ANALYSIS

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ATOMIC ENERGY OF CANADA LIMITED
Power Projects

NUCLEAR POWER SYMPOSIUM

LECTURE NO. 8: SYSTEMS ANALYSIS

by

D. B. Primeau

1. INTRODUCTION

In earlier lectures you have heard about various reactor types and their subsystems and also about nuclear fuel. I would like to describe some aspects of analysis required to support various heat transfer and fluid dynamic aspects of water cooled CANDU designs. In particular I shall be comparing the D_2O cooled systems, which operate with a small amount of boiling in the primary system, and the H_2O cooled systems, which operate with significant boiling.

First of all let's compare the basic primary system arrangements. In Figure 1 the D_2O cooled system is shown in its simplest form. The reactor is horizontal, there is very little boiling in the primary system, if at all, and the system is indirect. The secondary side coolant is light water and both primary and secondary coolants are circulated by pumps.

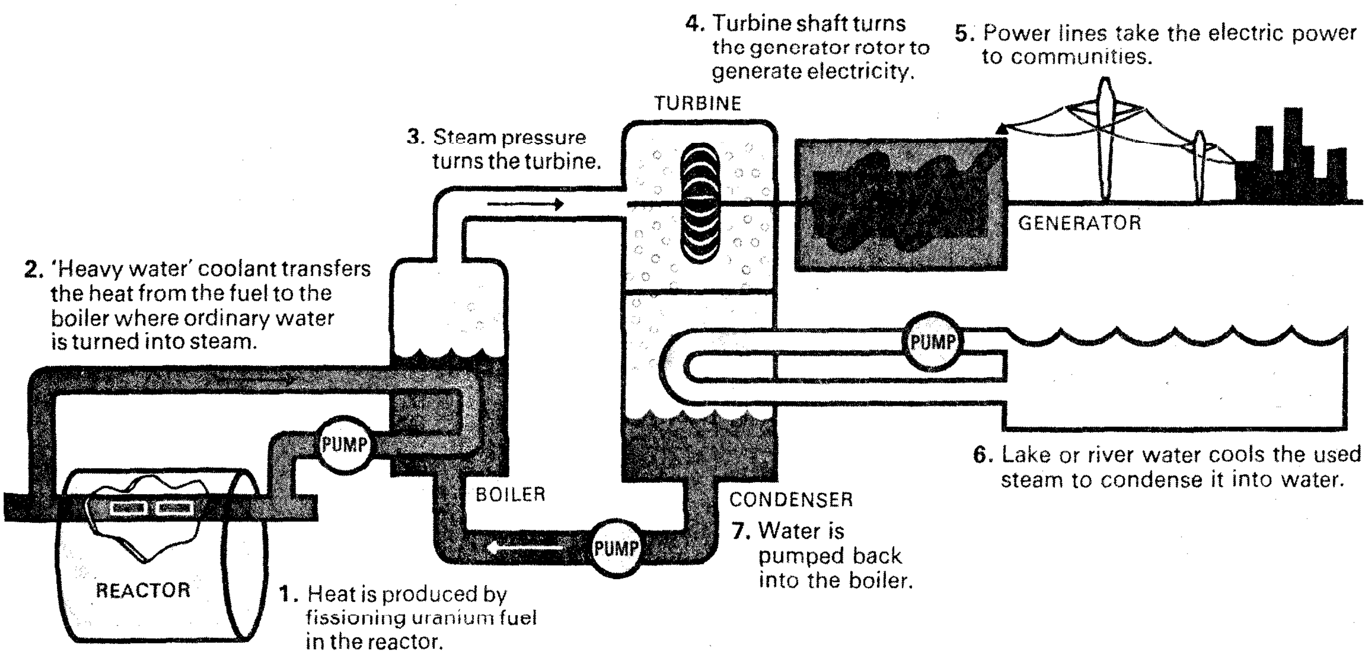

Figure 1 CANDU Pressurized Heavy Water Nuclear Power Process

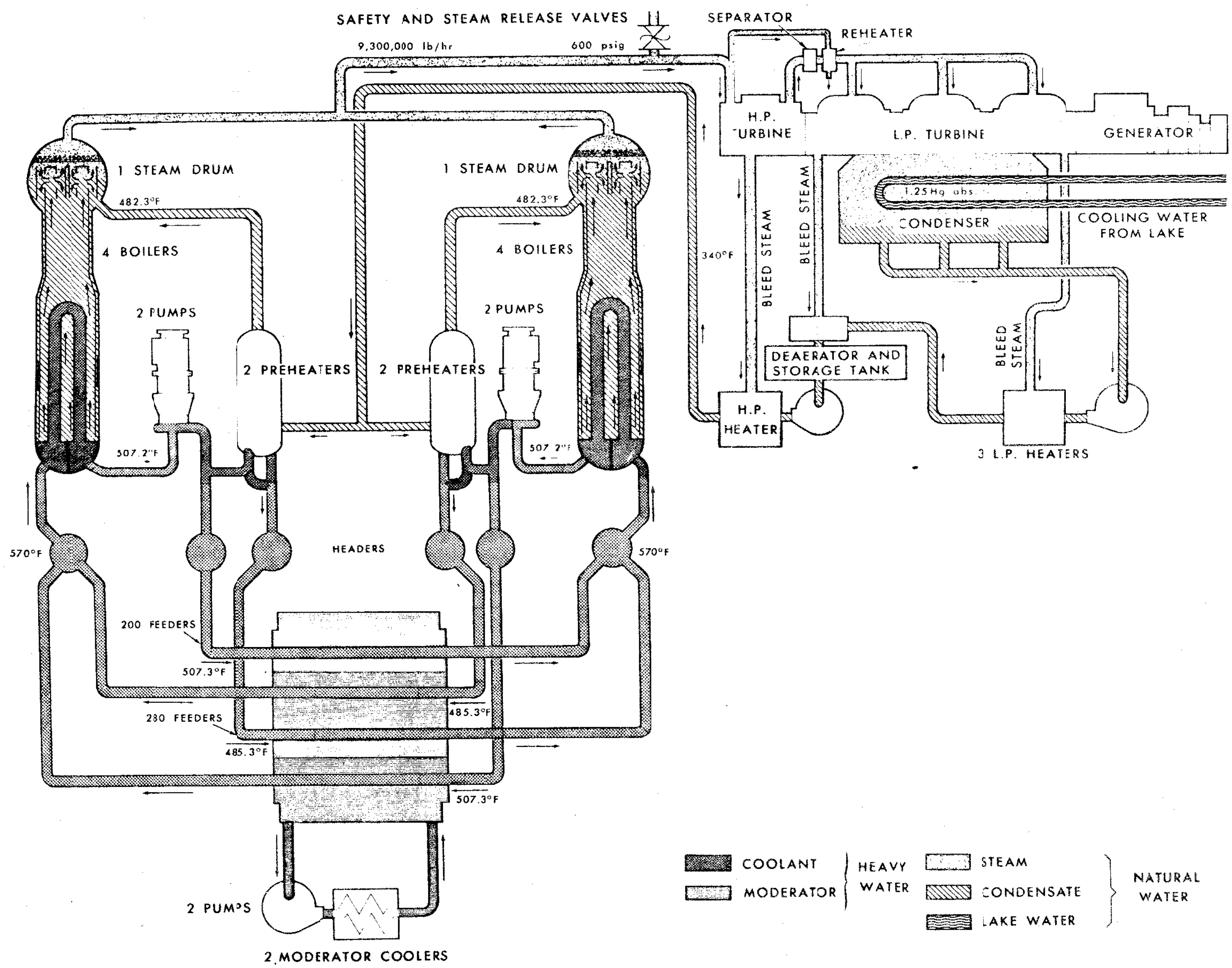

Figure 2 Bruce - Flow Diagram

In Figure 2 another standard feature is shown. This is the "figure of eight" layout of the primary system. It will be noted that a particle of fluid passing through the core does not return to the same core channels immediately. Instead it passes through a distinct set of boilers and pumps and through the core in the opposite direction before it returns to its starting point.

This figure also shows the reactor core as the low point in the system. This is very desirable since it leads to good thermosyphoning characteristics. Hence even without the pumps operating the fuel will be cooled to quite high power levels. Being the low point of the system also ensures that as long as there is water in the system the fuel will be covered.

The boiling light water system, as shown schematically in Figure 3, is a direct cycle. This means that the steam which drives the turbine is generated directly in the reactor core. This is only economically possible because the coolant is light water. It will also be noted that the fuel channels are vertical and flow is upwards in all channels. As the coolant passes through the core it changes from single phase liquid to two phase steam-water. The rate at which steam is generated is approximately 20% by weight flow of the total primary flow. The steam is separated from the water in the steam drum. The turbine feedwater system returns the condensed steam to the drum where it is mixed with the coolant coming from the core.

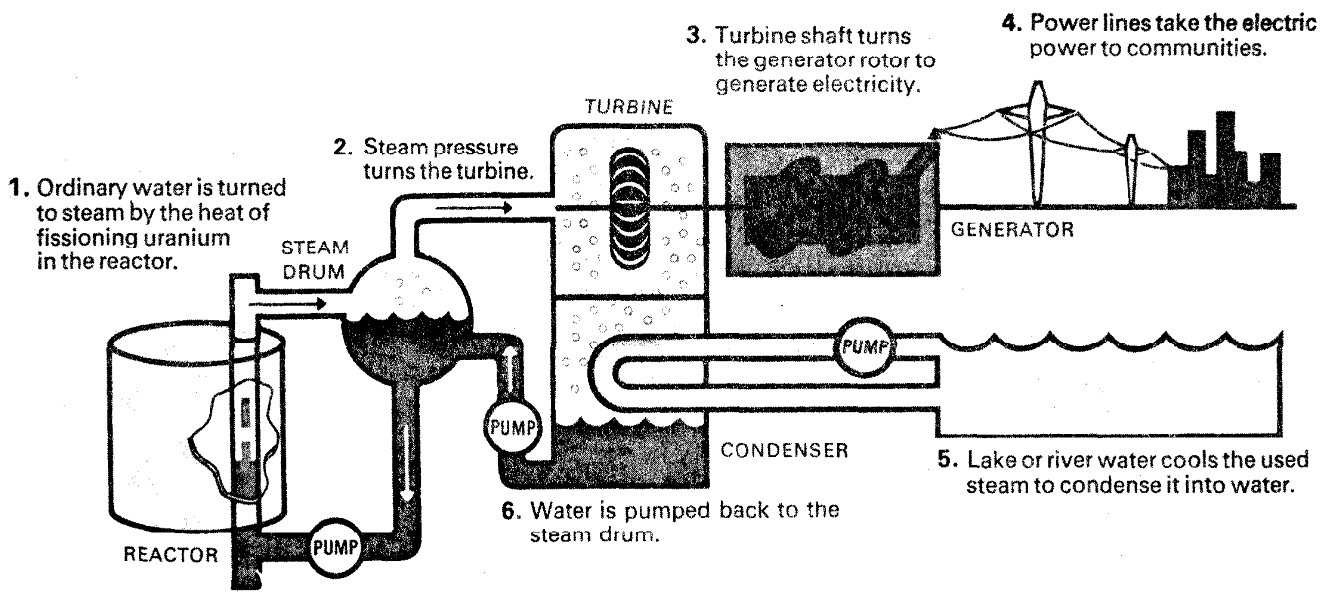

Figure 3 CANDU Boiling Light Water
Nuclear Power Process

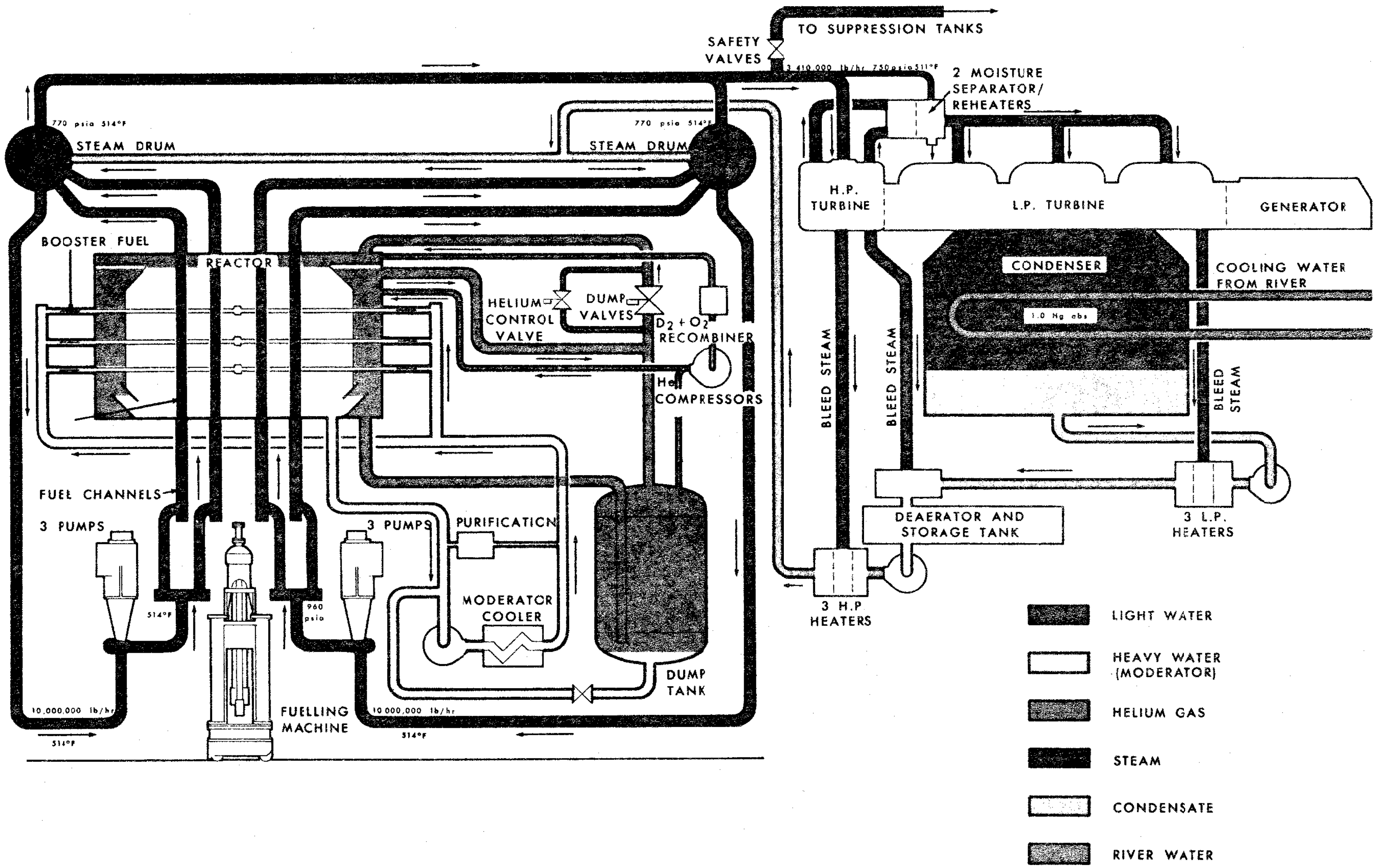

Figure 4 Gentilly - Flow Diagram

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Figure 4 shows the Gentilly system and in particular identifies the two separate primary loops decoupled hydraulically except at the turbine. The channels associated with each loop are also uniformly mixed in the reactor core. These features were largely dictated by safety considerations. In the event of a pipe rupture the neutronic influence is restricted to one half the potential effect.

There are other inherent differences between the systems as shown in Figure 5. In the D_2O system the operating pressure is much higher because it is an indirect cycle. Because it is indirect there is a small but important temperature drop across the heat exchanger tubes. This leads to the higher pressure. It is, of course, possible to operate the primary side at higher temperatures and hence achieve comparable steam pressures, however, at this stage in the development of D_2O cooled reactors the steam pressure has been limited by the amount of boiling and/or temperature that has been permitted.

	BRUCE	GENTILLY
STEAM PRESSURE (BAR)	43	52
PRIMARY PRESSURE (BAR)	86	53
PRIMARY SYSTEM QUALITY (%)	0	20
CORE LENGTH (m)	6	5
FUEL SUBDIVISION	28 OR 37	18

Figure 5 Bruce-Gentilly Comparison

In Figure 6 the fuel geometries currently in use in Gentilly, and proposed for Bruce, are shown. They are all designed to fit inside a 10 cm ID pressure tube. The linear power that can be extracted is lower for the Gentilly design mainly because of limitations with the 18-element fuel.

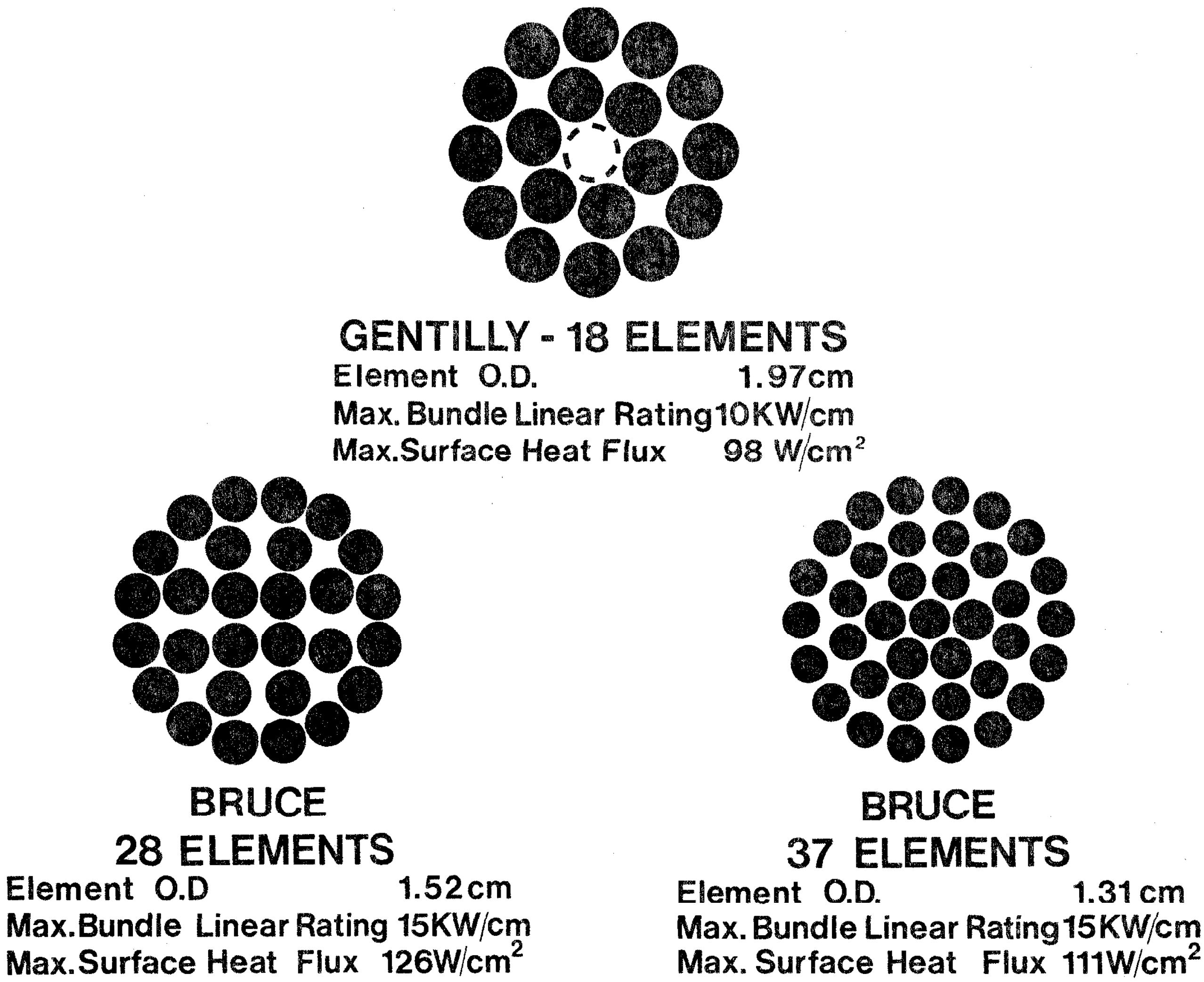


Figure 6 Fuel Bundles Cross Sections

It is optimum to operate Gentilly with large amounts of boiling since the neutron economy of a light water cooled natural uranium system is lower than with heavy water. The more boiling the lower the fuel operating costs. The limit on boiling is probably set by heat transfer considerations.

2. PRIMARY SYSTEM PRESSURE LOSSES

There is an upper limit to the flowrate because of increased pressure losses. In Bruce and Gentilly the pressure loss (psig) of the primary circuit is roughly 14 bar which is a large fraction (~20%) of the operating pressure. The pressure loss in a pipe is made up of three components as shown in Figure 7.

PRESSURE LOSS = FRICTION + MOMENTUM + GRAVITY

$$\text{FRICTION} = \frac{1}{2} \rho_L V_L^2 \left[\frac{fL}{D}\right] \phi^2$$

$\frac{1}{2} \rho_L V_L^2$ = KINETIC ENERGY OF THE EQUIVALENT LIQUID

$\frac{fL}{D}$ = FRICTIONAL AND FORM LOSS COEFFICIENT

ϕ^2 = TWO-PHASE MULTIPLIER

Figure 7 Pressure Losses

Frictional losses are caused by surface roughness or inefficient flow passage shapes that can cause energy dissipation in the formation of turbulent eddies. A momentum component exists when steam is being formed in the flow passage, mainly through heat addition or removal. Since more volume is required per unit mass of coolant, an acceleration force is required. This causes a pressure loss. The gravity component is caused by elevation or coolant density differences. The friction loss is the main loss which influences the sizing of pumps and is a strong function of the fluid conditions. The friction loss is normally expressed as follows:

$$\Delta p = \frac{1}{2} \rho_L V_L^2 \left(\frac{fL}{D}\right) \phi^2$$

where it is standard practice to define the losses in terms of the number of velocity heads, $\frac{1}{2}\rho V^2$. fL/D is a loss coefficient that represents the combined surface friction and form loss. $\emptyset^2$ is a coefficient that accounts for boiling and depends on the fluid conditions. In the case of liquid flow, $\emptyset^2$ is unity.

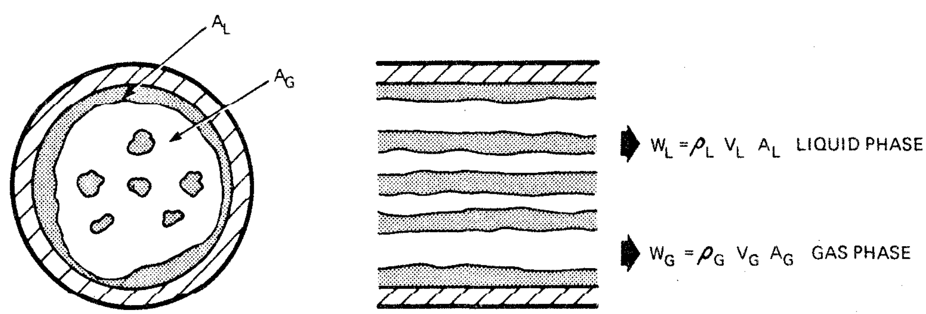
Note that $\rho_L V_L^2$ is defined in terms of the density and velocity of the equivalent all liquid flow.

The factor $\emptyset^2$ is called the two phase multiplier. However, before discussing $\emptyset^2$ let us define certain terms related to two phase flow.

Figure 8 shows the idealized flow of steam and water in a pipe. Each phase is given its own velocity V, flow area A, and density ρ. The phase velocities are generally not equal in two phase flow, and the term slip is used for the ratio of their velocities.

The quality is defined as the fraction of the total mass flow passing a fixed plane that is steam. The symbol X is usually reserved for quality.

One other quantity that is very important is the void fraction, usually denoted by α. It is defined as the fraction of the total flow area that is occupied by steam. It is particularly important from a reactor



BASIC DEFINITIONS

SLIP = V_G/V_L

MASS FLOW RATE = $W_T = W_G + W_L$

QUALITY = $\frac{W_G}{W_T}$

VOID FRACTION = $\frac{A_G}{A_G + A_L}$

Figure 8 Two-Phase Gas-Liquid Flow

physics point of view since the reactivity of the core is dependent on the amount of coolant in it and a change in the void fraction can alter the rate of power production.

Let us return now to the two-phase multiplier and its dependence on quality and pressure.

In Figure 9 ϕ^2 is shown as determined by Martinelli and Nelson for steam-water mixtures. It shows that pressure and quality are both very important. Since we have defined the frictional pressure drop in terms of the velocity of all liquid flow, ϕ^2 incorporates an allowance for the higher two-phase mixture velocity. Since the difference in velocity is greater the lower the pressure, ϕ^2 is largest at low pressure. As the pressure approaches the critical pressure the multiplier approaches unity.

At Gentilly conditions ϕ^2 is about 10, meaning that the pressure drop in certain portions of the primary system would be about an order of magnitude lower if the coolant were all liquid.

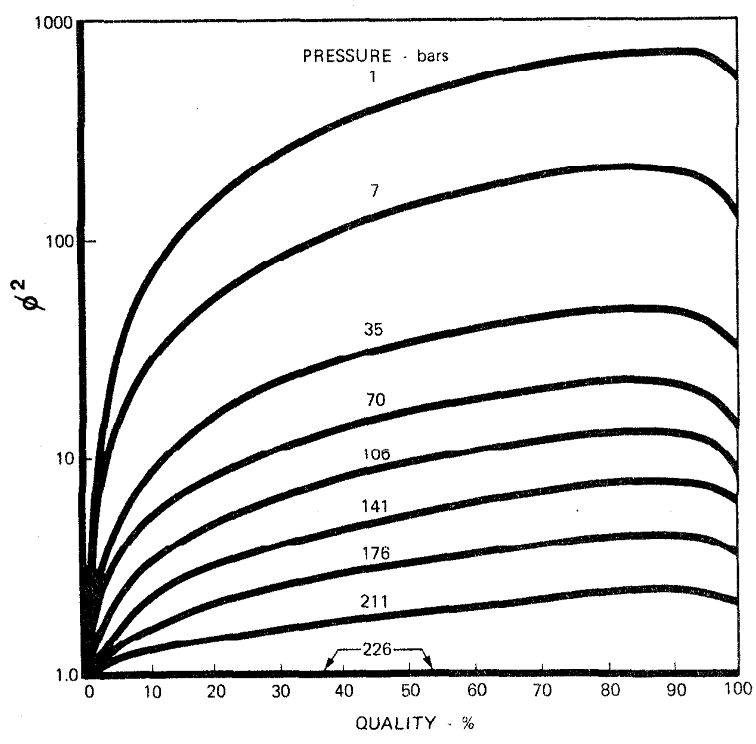


Figure 9 Two-Phase Friction Multiplier

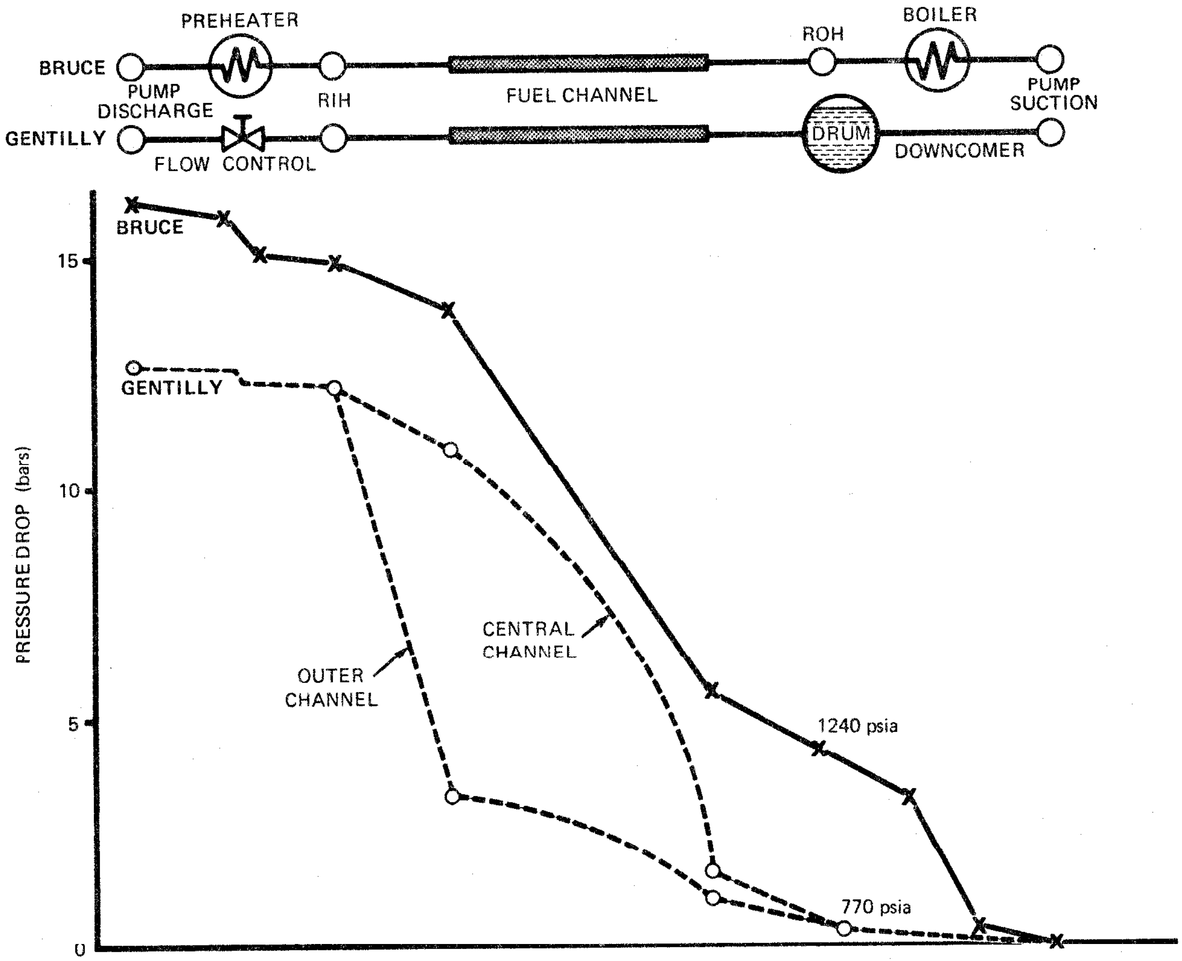


Figure 10 Pressure Distribution in Bruce and Gentilly

Figure 10 shows the calculated pressure distribution in Bruce and Gentilly. This figure shows the pressure decrease as the coolant moves from pump outlet to pump suction. The pressure in Bruce is controlled at the outlet header to an absolute value of 85 bar by means of a surge tank. In Gentilly the steam drum pressure is controlled to a nominal pressure of 53 bar. This is controlled by steam flow control.

It will be noted that the pressure drop in Bruce is larger than in Gentilly. This arises primarily because the optimum flow in a D_2O cooled CANDU system is higher than in a H_2O cooled system. The maximum coolant velocity at the core inlet to Bruce is approximately twice as high as it is in Gentilly. Because of the velocity squared dependence of pressure drop, one would expect the loss in Gentilly to be almost one quarter that in Bruce but we can see from Figure 10 that it is almost the same. This apparent anomaly is explained by the high steam qualities in Gentilly.

The loss in the path through the central channels set the required pump head, and the largest part of that loss is in the fuel channel. Designing fuel channels for low pressure drop receives considerable effort.

There was a basic difference in the design approach regarding the flow per channel. In Bruce the flow is equal in all channels. In Gentilly there was an incentive to operate all channels at as high a quality as possible as limited by heat transfer. This led to the addition of extra flow resistance to the outer channels. In those channels a major part of the pressure loss is across this resistance.

3. FLOW REGIMES

A few words should be said about the various flow regimes that may occur in two-phase flow. It is important to establish which regimes may occur in an operating reactor since certain regimes would be undesirable from both a heat transfer and a control point of view.

Figure 11 shows schematically the important regimes. It shows the transition from single phase water to single phase steam. As heat is added and the water approaches the saturation temperature, bubbles will form at the surface of the heater and be carried away. This is called bubble flow. This can be followed by slug flow which is characterized by a large time variation in the void fraction passing any given point. Slugs of vapour will alternate with slugs of water. This is a very undesirable regime that must be avoided by design.

Normally reactor conditions are such that there is a transition directly from bubble to annular flow. This transition can vary from about 1 to 10% quality depending on the pressure.

In the annular regime the water phase tends to flow as a film on the available surfaces, somewhat preferentially on the cooler ones. This regime normally persists only at low flows.

At higher flows the liquid is also found in the form of droplets dispersed in the steam phase. This is called the annular dispersed flow regime.

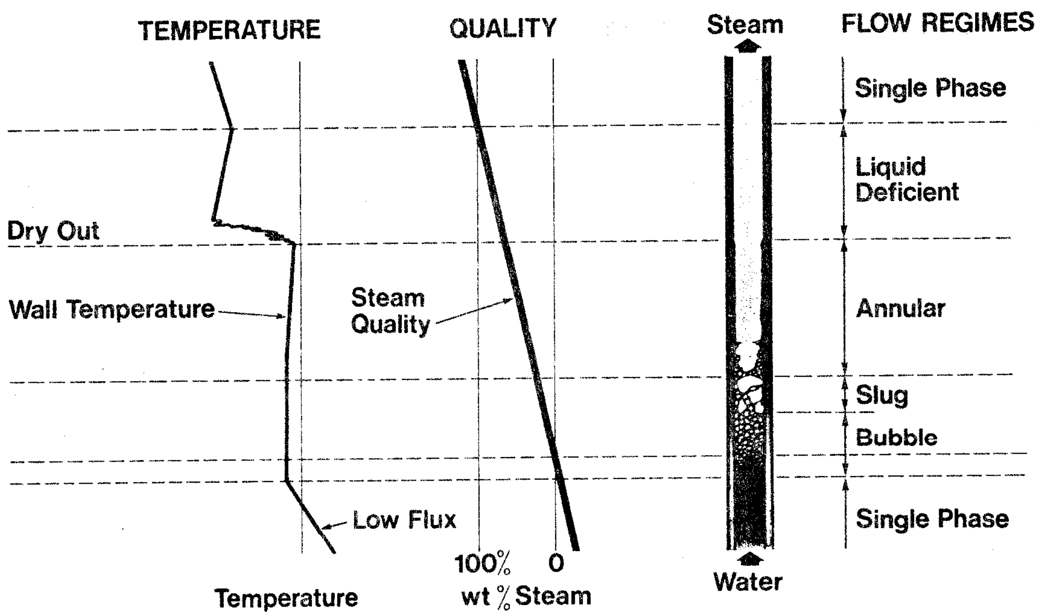


Figure 11 Thermo-Hydraulic Regimes
in Vertical Upward Flow

The next regime is called liquid deficient since the surface film has been disrupted. When this occurs it is often referred to as the dryout point and the heat flux at this point is called the critical heat flux. It will be noted from the figure that at this transition there is a sudden increase in the wall temperature. More will be said about this later. As quality increases beyond this point the heat is transferred from the wall surface either by steam or by the occasional evaporation of water droplets that may strike it.

Figure 12 shows the range of flows and qualities that define these flow regions for pressures typical of CANDU systems. The boundaries are not nearly as well defined as might be interpreted from the figure since pipe geometry and orientation can also influence them. However, it indicates that for the low quality D_2O systems, the normal flow regime will be bubble flow, whereas in the H_2O systems both bubble and annular dispersed flow regimes can co-exist.

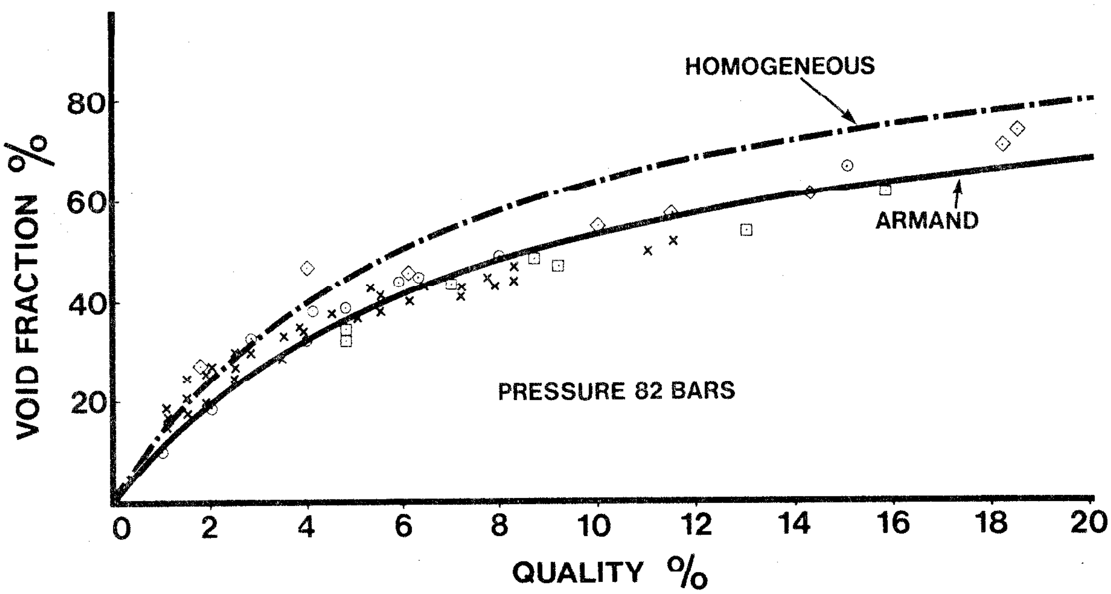


Figure 13 Void Fraction vs. Quality

conditions at core exit. Averaged over the full channel the void fraction is about 50%.

Also shown on this figure are predictions by two theoretical correlations, the homogeneous correlation and the other, the modified Armand correlation. The latter shows reasonable agreement when due allowance is made for the scatter in the experimental data, and is used in much of our hydraulic analysis.

The basic assumption of the homogeneous model is that the steam phase velocity is equal to that of the liquid phase, or in other terms, there is a slip of unity. In the low quality region this assumption appears to be justified intuitively and also from a comparison of the correlation to the experiments. However, at the higher qualities the model overpredicts α and the discrepancy is ascribed to the slip assumption.

In order to give an idea of the magnitude of slip let us look at Figure 14 This shows estimates of coolant velocity for a flow rate typical of that in Bruce and as a function of steam quality. The maximum quality in Bruce is approximately 3%.

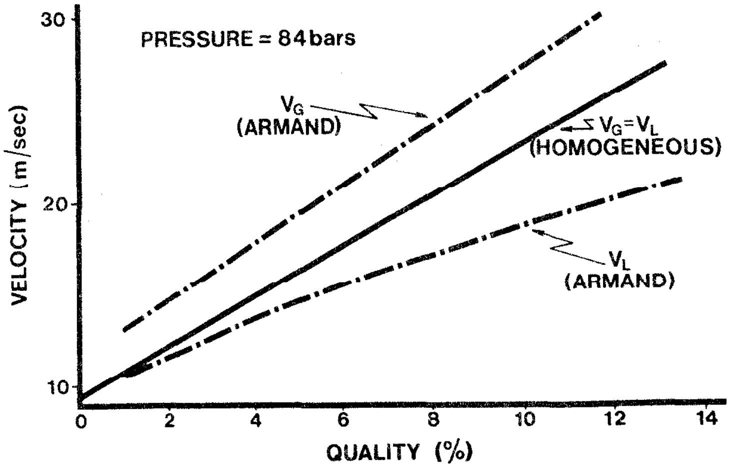


Figure 14 Liquid and Gas Phase Velocities vs. Quality

As the quality increases the average fluid velocity increases. Since the homogeneous model is probably quite good up to 3% quality we would expect velocities at the exit end of a channel that are approximately 50% higher than at the inlet.

At a quality of 10%, Armand's correlation would suggest a slip of about 1.5.

5. FUEL AND CHANNEL CONDITIONS

Let us now consider the operating conditions of a Bruce and Gentilly channel. Figure 15 shows the neutron flux distribution calculated for the Bruce reactor across its diameter. It is similar in most CANDU type reactors. The central zone is flattened by irradiating the fuel in that zone longer so that it is less reactive. The control of the shape is maintained by the refuelling program. To a first order of approximation the channel power distribution follows the same shape. Note that the reflector helps to keep the power up even in the outer channels. The average to maximum channel power ratio is between 0.8 and 0.9 for CANDU stations.

In the axial dimension the operating power distributions are also very similar in all CANDU reactors. They are nearly sinusoidal in shape, but due to boiling, the axial distributions of sheath and coolant temperatures differ in Gentilly from those in Bruce.

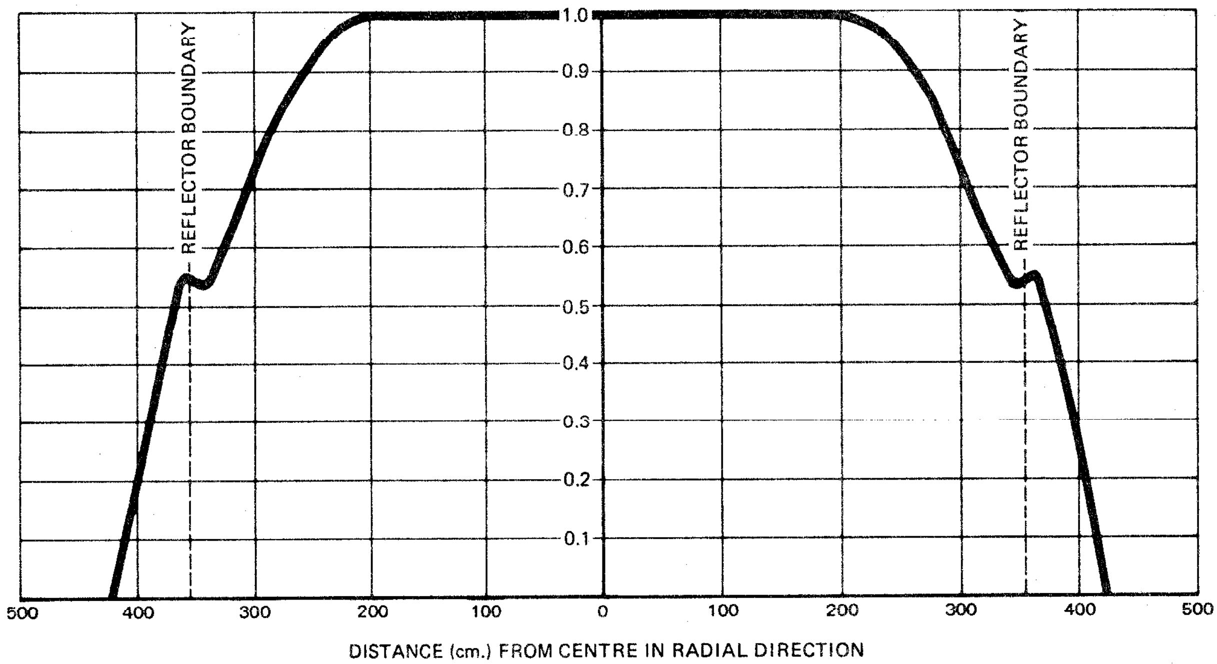


Figure 15 Bruce Radial Neutron Flux Distribution

Figure 16 shows Bruce axial distributions. This is typical of a central channel. Note that the average exit coolant temperature equals the saturation temperatures so that there is no net quality at outlet. The coolant temperature rises with position along the channel, but because of different hydraulic characteristics of the subchannels within the fuel, the coolant temperature varies from subchannel to subchannel. The differential can be enough to lead to net boiling in certain subchannels. In the example shown the small outer subchannel in the centre of the triangular element array has the highest temperature rise.

The fuel sheath temperature follows the coolant temperature fairly closely with only 20^{o}C maximum difference in the example shown. This rather small difference is characteristic of the very good heat transfer that exists in CANDU reactors.

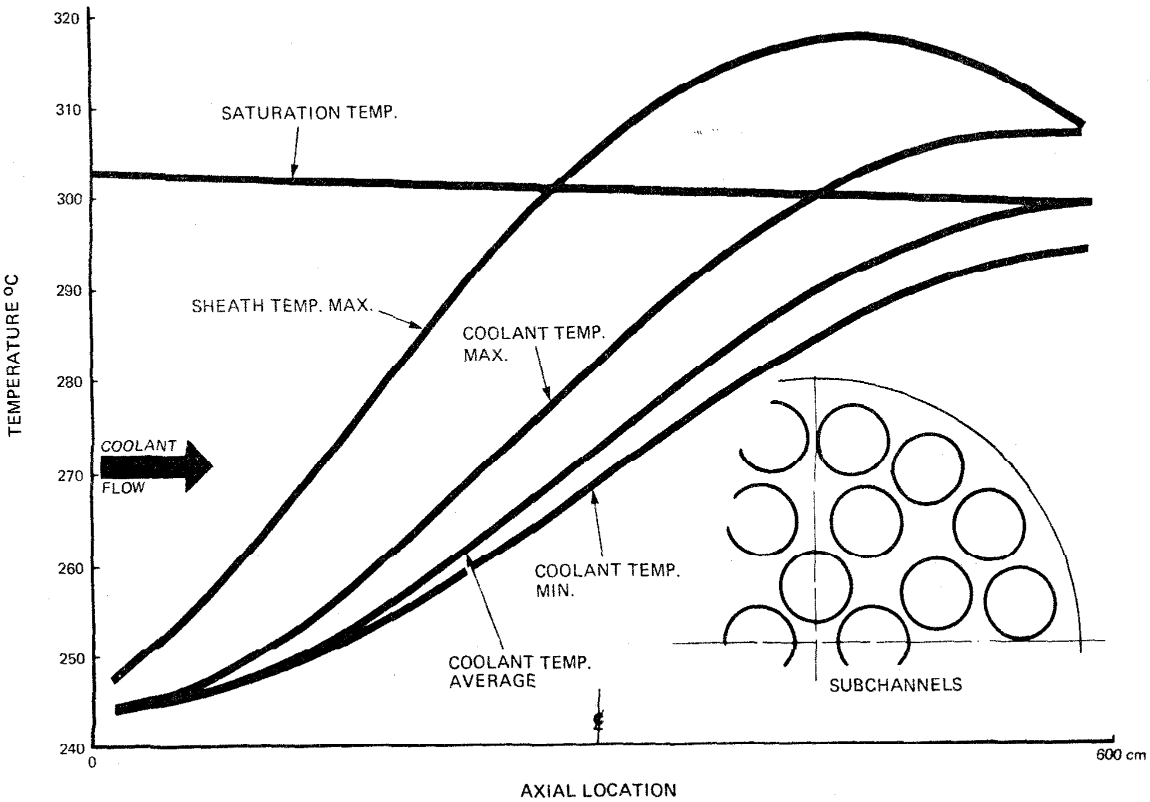


Figure 16 Axial Sheath and Coolant Temperatures in Bruce

The axial variations are somewhat different for a boiling system. Figure 17 shows results for a Gentilly central channel. In this example the power distribution has been assumed to take the ideal sinusoidal shape. The exit coolant quality is 17% and the onset of boiling is initiated about 1/4 of the way up the channel. It should be recalled that the Gentilly channel is vertical. The bulk coolant temperature rises until boiling begins and then follows the saturation temperature corresponding to the pressure. As in the previous example, the sheath temperature closely follows the coolant temperature.

As has been mentioned earlier there is a strong variation of the coolant density along the channel, this is a characteristic peculiar to a boiling channel, and this example shows that the density decreases dramatically with quality reaching a value at the core exit that is only 30% of that at the inlet.

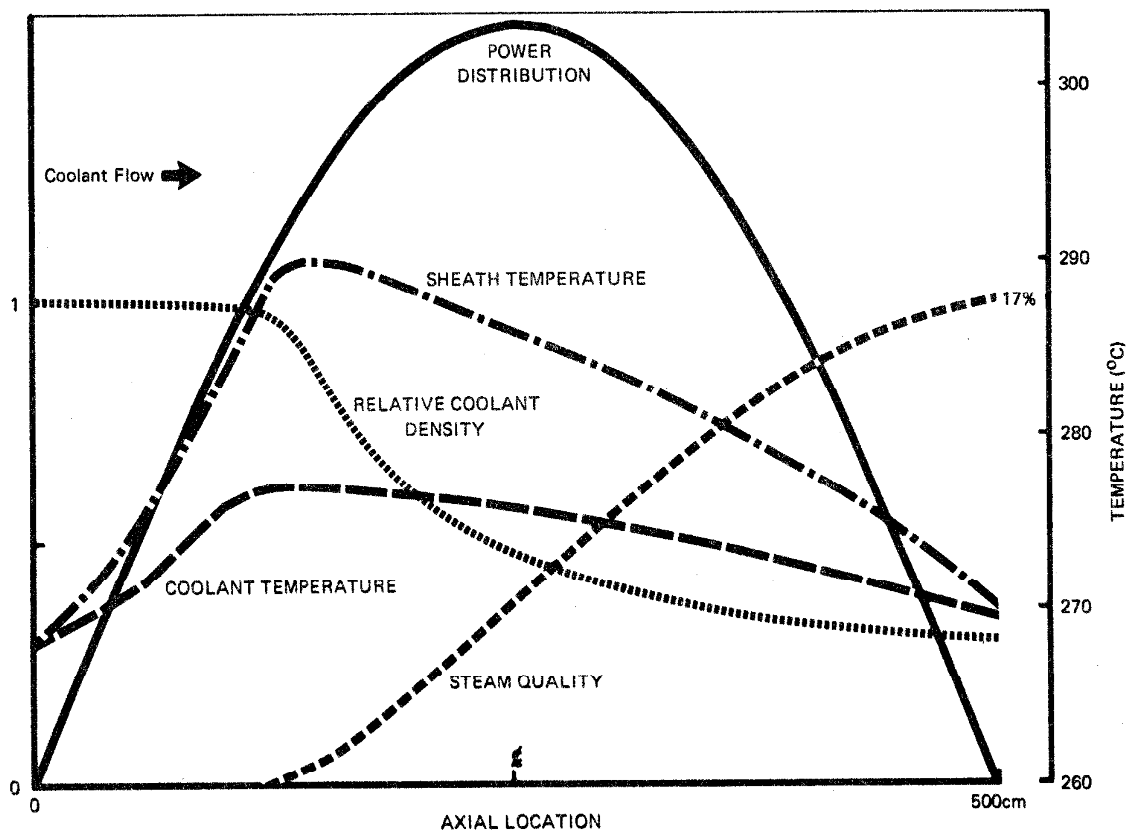


Figure 17 Axial Distributions in Gentilly

The axial temperature variations that have just been shown for both systems are very small, relative to the large variations that exist inside the fuel itself. This arises because the conductivity of the uranium oxide fuel is extremely low, hence a very large temperature differential is necessary to support the flow of heat. Compared to the highly conductive material aluminum, the oxide conductivity is 75 times lower. Figure 18 shows the calculated temperature distribution inside the fuel pellet for the Bruce 28-element fuel. The distribution would be very similar in the Gentilly fuel except for the radius. There are temperature drops across the coolant film, sheath and the fuel-to-sheath interface, but these can be seen to be small, relative to that in the fuel itself. The peak central temperature is approximately 2100°C. If the power were increased by about 45% the central portion of the fuel would begin to melt.

The fuel temperatures shown were for the hottest pencils of the bundle in the hottest region of the core. The fuel temperature over the majority of the core is well below those shown.

For example, Figure 19 shows variations along the axial dimension of the hottest channel.

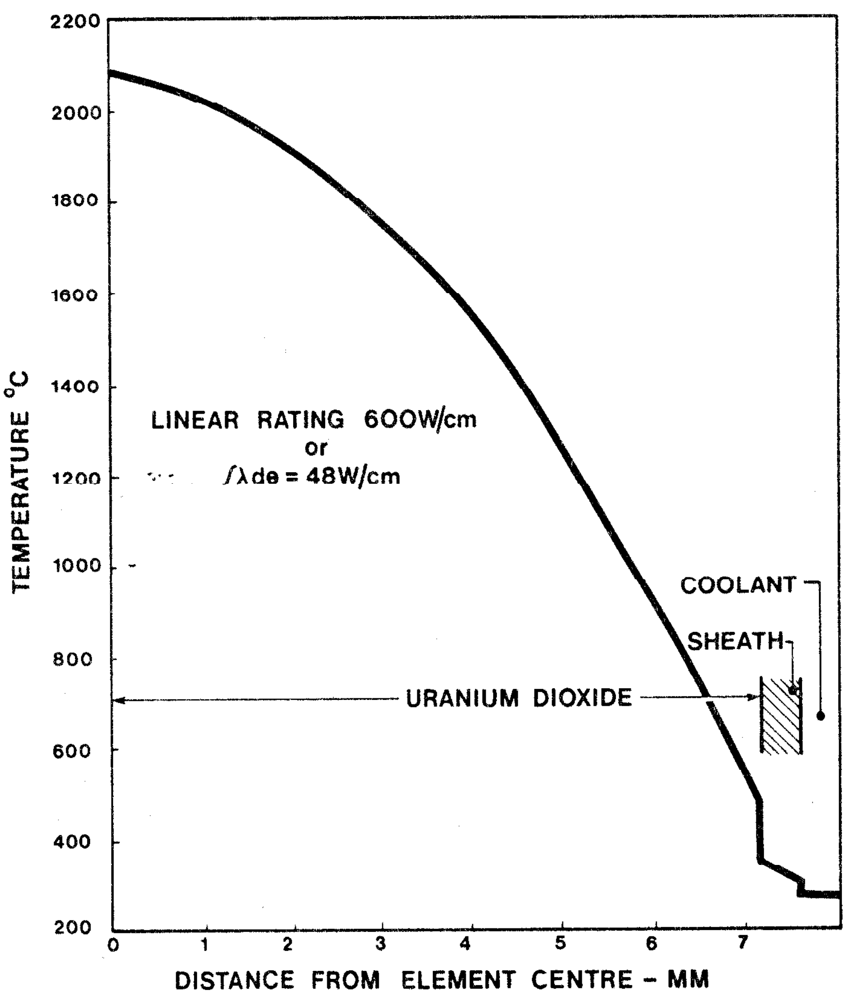


Figure 18 Radial Fuel Pellet Temperature

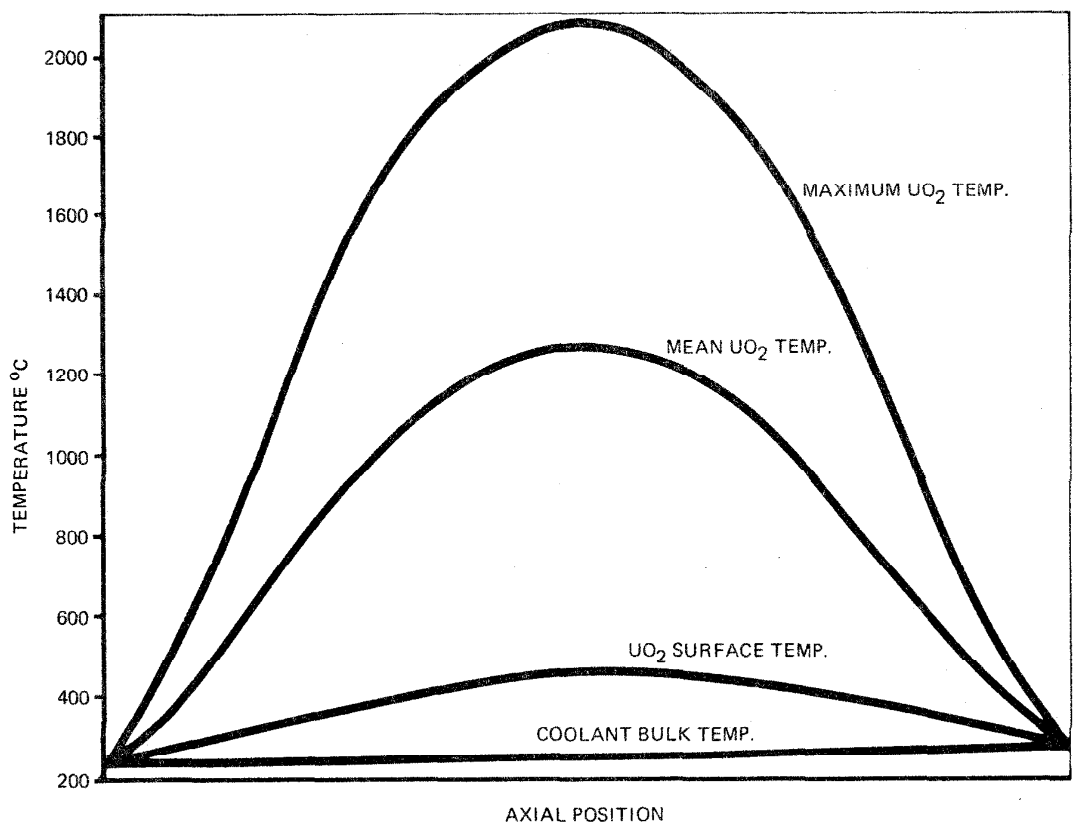


Figure 19 Axial Fuel Pellet Temperatures

6. <u>HEAT TRANSFER LIMITS</u>

The previous discussion has shown that under normal operating conditions, the cooling provided is extremely good, and the fuel sheath temperature run only slightly above coolant temperatures.

I would now like to discuss what eventually happens to the cooling process as the power is raised and/or the flow is decreased. Let us first of all define some of the terms. Figure 20 shows that in the single phase flow regimes the heat transfer is called "convective". As soon as bubbles begin to be formed on the heated surface the heat transfer mode is known is nucleate boiling. In the annular or annular dispersed flow regime, the heat transfer is also convective but this time to a water film. In all of these heat transfer regimes the cooling is extremely good.

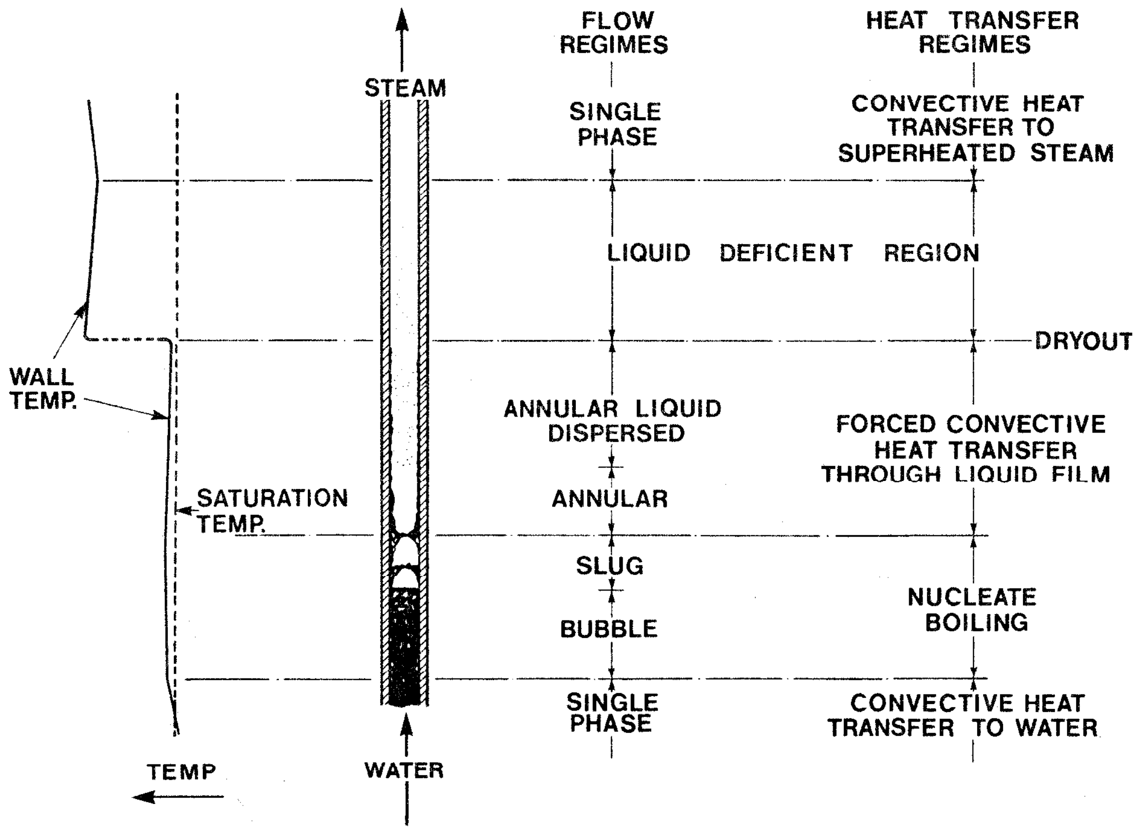


Figure 20 Heat Transfer Regimes

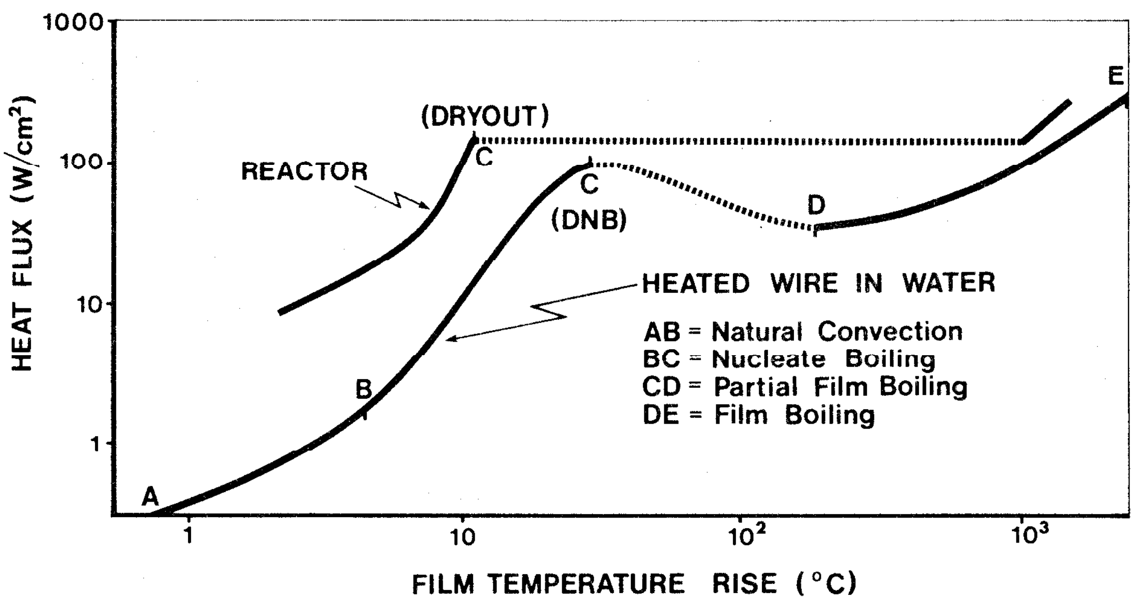


Figure 21 Boiling Crisis

In the liquid deficient region the coolant film breaks down and we have a very much less efficient heat removal process. The transition is called dryout and is characterized by a large increase in the temperature rise across the surface film. This phenomena sets the heat transfer limit of the reactor channels.

The dryout phenomena can also be illustrated in a slightly different manner. Figure 21 shows the relation between surface heat flux (the power/unit area of surface) and the film temperature rise. Typical results are shown for reactor conditions and for a wire heated in a pool of water.

In the case of the wire there is no forced convective cooling and the crisis, labelled as "C", is called DNB, the Departure from Nucleate Boiling. It is called this because the crisis occurs in a nucleate boiling flow regime. At DNB the surface becomes steam blanketed and the temperature must rise to transfer the power.

Under reactor conditions the point "C" is called dryout, although this is rather an arbitrary distinction since the net result is very similar, that is, a sharp rise in sheath temperature.

To determine the heat transfer limits of a reactor channel, the dryout or critical heat flux (CHF) must be determined experimentally. Because of the complications of real reactor fuel it is impossible at this stage to analytically predict CHF, at least to the accuracy that is needed.

Figure 22 shows typical CHF experimental data for two fuel geometries, one being similar to that used in Gentilly. The one dependence found in all CHF experimental data is the decrease of CHF with steam quality. Second order dependencies have been found with flow and pressure but these effects are not at all well understood. In the figure, pressure is the unlabelled third variable. The measurement and prediction of CHF is a major heat transfer field.

Using critical heat flux data of the type we have just shown, it is possible to calculate the critical power of a reactor channel. The ratio of the critical power to the nominal power is often called the critical power ratio. Figure 23 shows the results of calculations for the Gentilly central channel. The nominal power distributions and quality distributions are shown together with the critical power and quality distributions. The crisis is predicted to occur when the heat flux, at any point along the channel, equals the critical heat flux predicted for the conditions which exist at that point. From these results we would say that the Gentilly central channel had a critical power ratio of approximately 1.5.

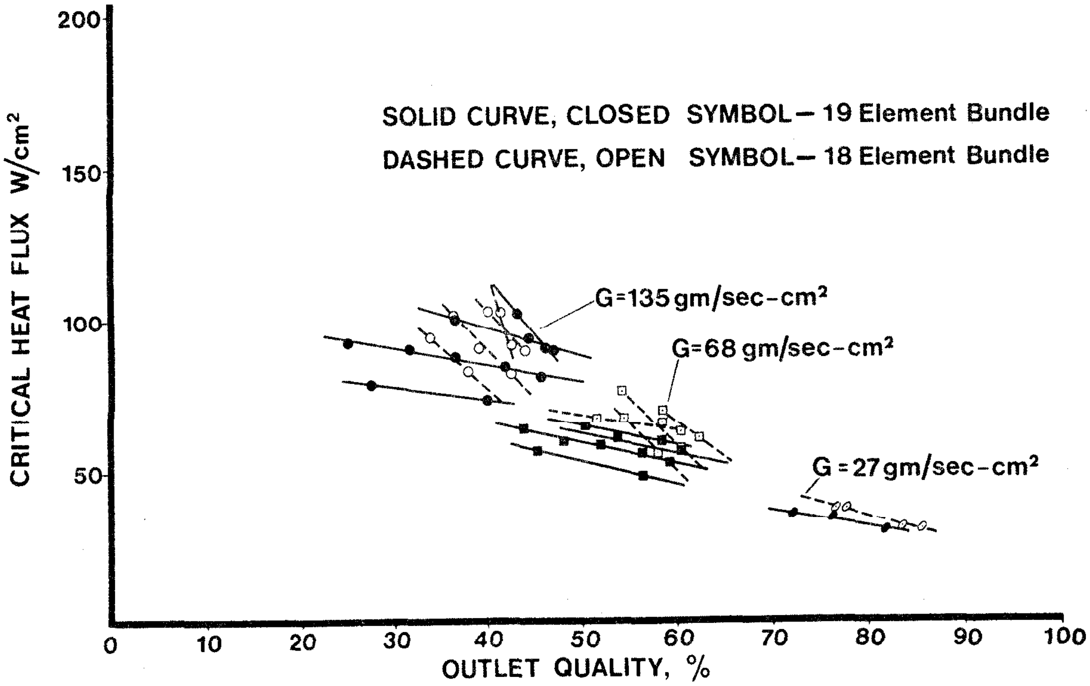


Figure 22 18-Element and 19-Element Bundle CHF Data

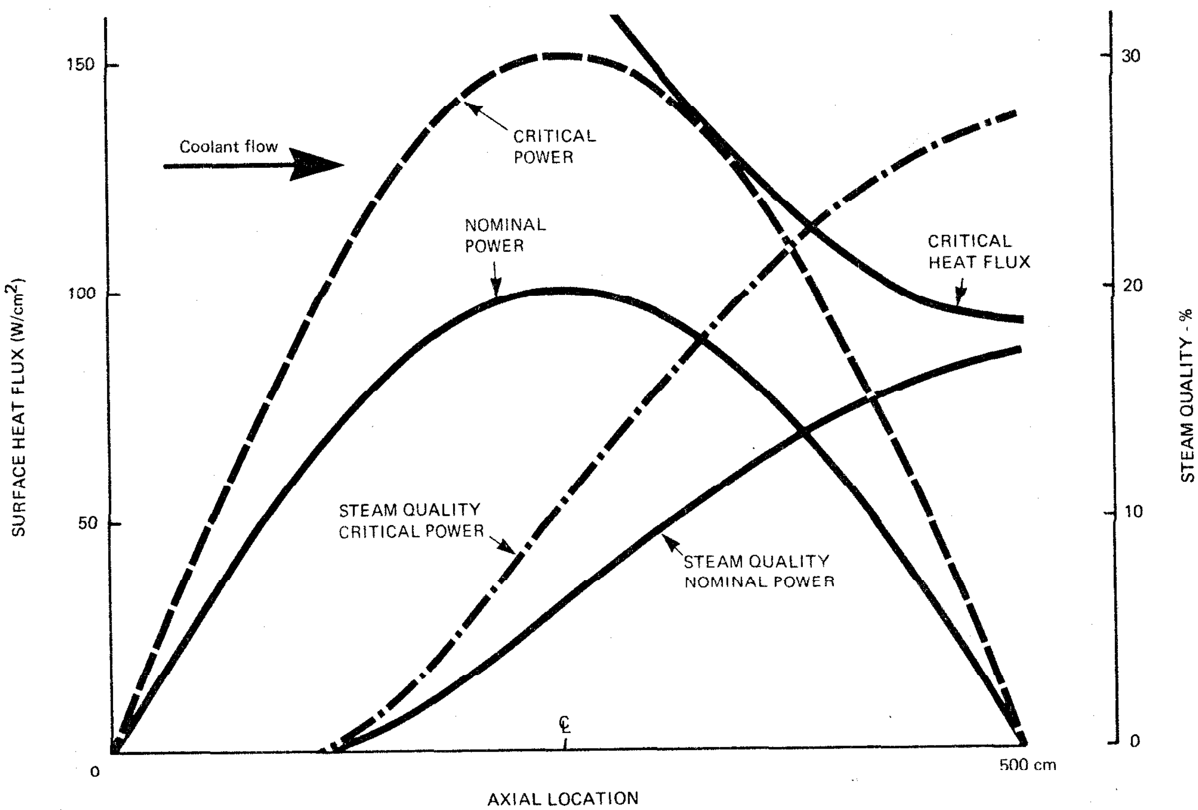


Figure 23 CPR Calculation for Gentilly

When a crisis occurs, the sheath temperature increases. Figure 24 shows the predicted sheath temperatures just before and just after critical conditions have been reached. It can be seen that below the critical conditions the sheath temperature varies little along the channel but on exceeding the critical power the sheath temperatures will rise by several hundred degrees over at least part of the channel. The estimated heat transfer coefficients of the film, labelled "h", are also shown. The temperature rise across the film are inversely proportional to this quantity.

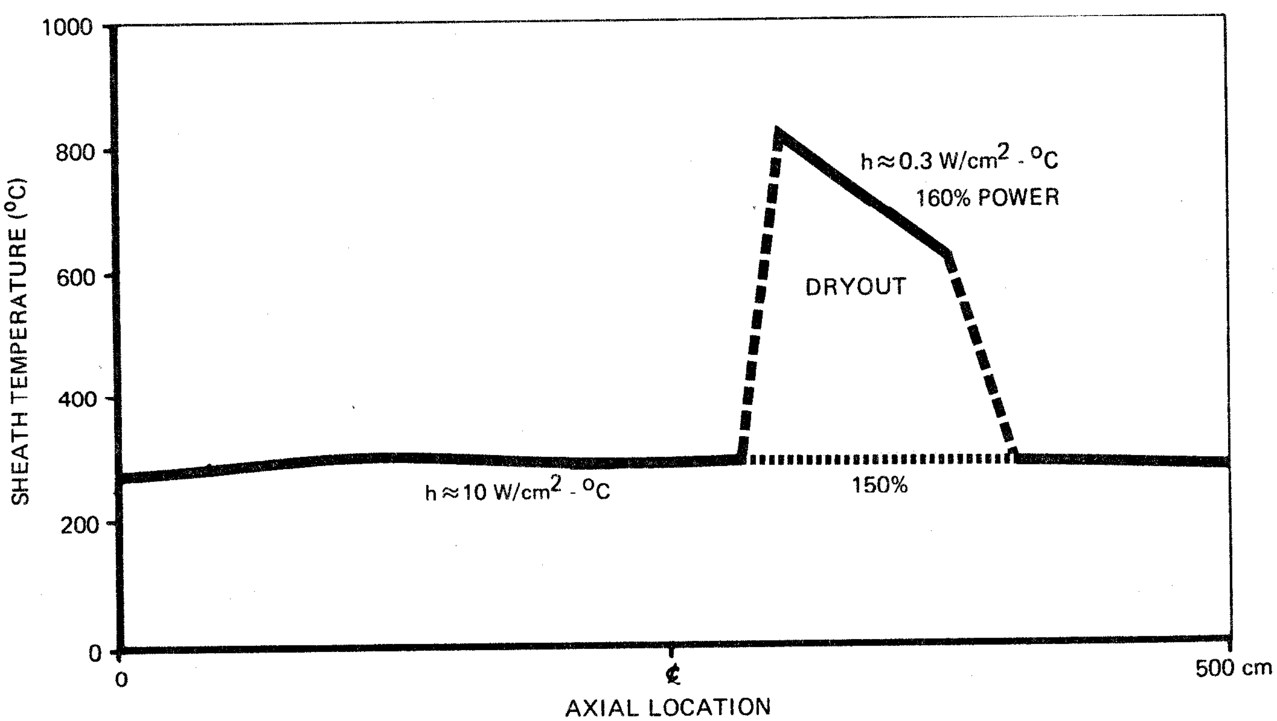


Figure 24 Axial Sheath Temperature in Gentilly

7. DRYOUT PROBABILITY

The critical power ratio will vary from channel to channel and from day to day. Consequently a probability approach to dryout was found necessary to establish suitable design targets. If a channel were heavily instrumented operating conditions would be known at any time and a much closer approach to the critical conditions would be permissible. In general, though, this is not practical and margins are chosen to cover the lack of this kind of information.

By isolating the important operating parameters, by establishing the variation that can be expected to occur in each, and by combining their influence in a statistical manner, the probability that dryout will occur can be estimated. Of course, the accuracy of the estimate depends on some rather ill-defined quantities. These can only be firmed up by operating experience.

Figure 25 lists the major parameters that must be considered. The study has been carried out for Gentilly, the parameters are listed in order of importance. The first three were by far the most important, with the channel power and power distribution being the single most important parameter of those three.

The final results of the Gentilly study are shown in Figure 26. The estimated percent probability that dryout will occur is shown as a

- CHANNEL POWER AND POWER DISTRIBUTION
- CHANNEL FLOWRATE
- CHF CORRELATION
- COOLANT PRESSURE
- INLET TEMPERATURE
- FUEL ELEMENT SPACING
- PRESSURE TUBE DIAMETER
- FUEL ELEMENT DIAMETER

Figure 25 Dryout Probability Allowances

function of the nominal critical power ratio. When the ratio is unity there is a 50/50 chance that a channel will be in dryout.

It will be noted that the probability decreases rapidly as the critical power ratio increases. At the Gentilly design value of 1.5 the possibility is so small that it is considered negligible.

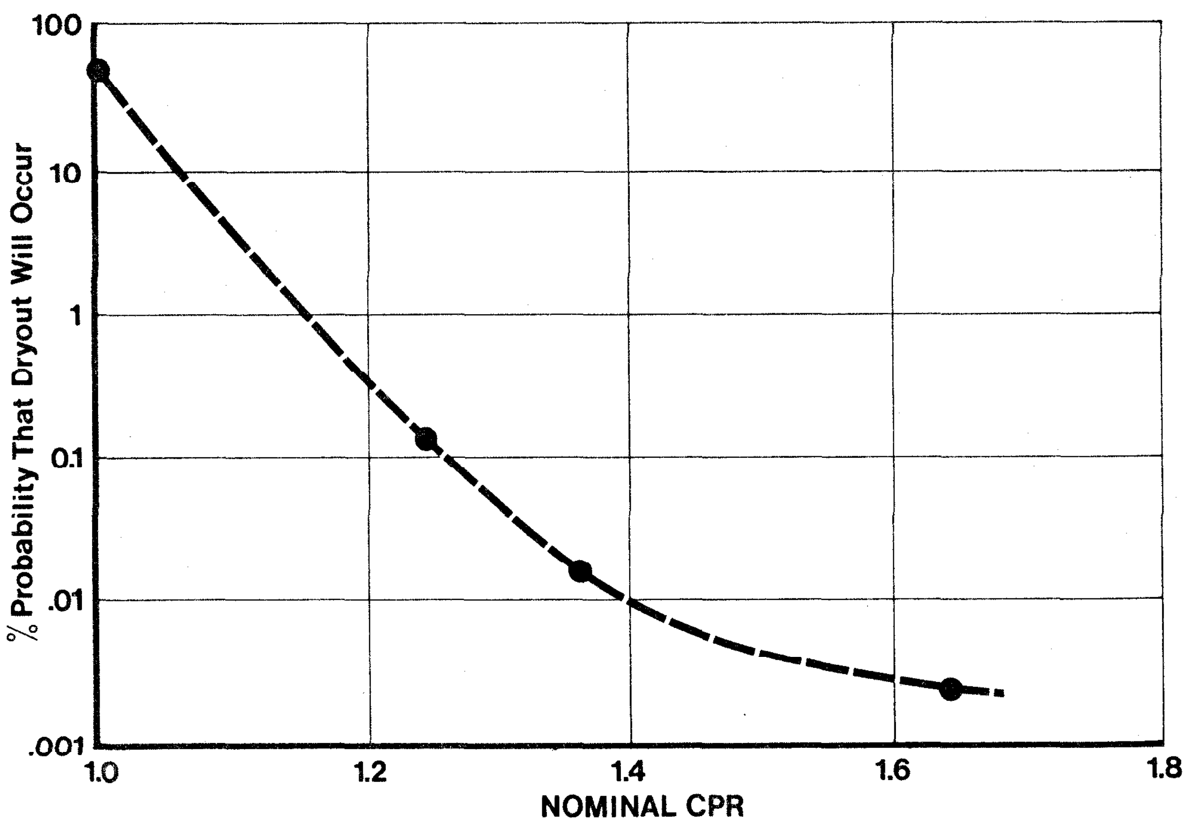


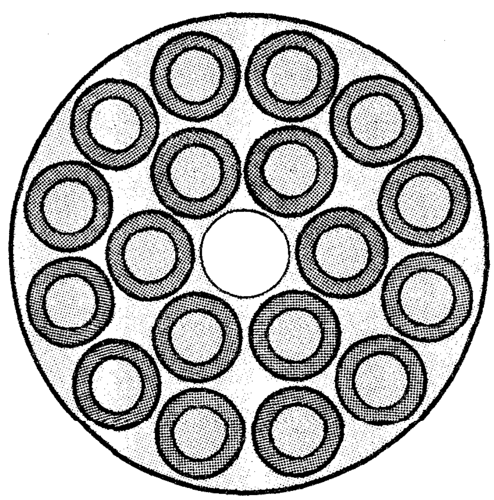
Figure 26 Probability of Dryout in Gentilly

8. BOOSTER HEAT TRANSFER

In CANDU systems there are a few highly enriched fuel assemblies used for control purposes. They are used to counteract reactivity changes associated with the buildup of Xenon-135 in the core. Xenon-135 is a strongly neutron-absorbing isotope.

Since boosters produce significant power when used, the same considerations of heat transfer apply as with the reactor channels.

Figure 27 shows a cross-section of the fuel. Its design differs in several ways from the reactor fuel. Each element is hollow with both internal and external cooling. It is made of a uranium-zirconium alloy extruded between sheaths of Zircaloy. Since zirconium has higher conductivity than uranium oxide (by about a factor of 5) the operating fuel temperatures are considerably lower.



ELEMENT ID = 1.33 cm
ELEMENT OD = 2.08 cm
MAXIMUM BUNDLE RATING 14 KW/cm
MAXIMUM HEAT FLUX 80 W/cm^2

Figure 27 Bruce Booster Fuel Cross-Section

Perhaps the most important difference from a heat transfer point of view is the use of a low pressure coolant. However, before I discuss that, let us look at the channel design and their locations in the core. I shall use the Bruce design as an example.

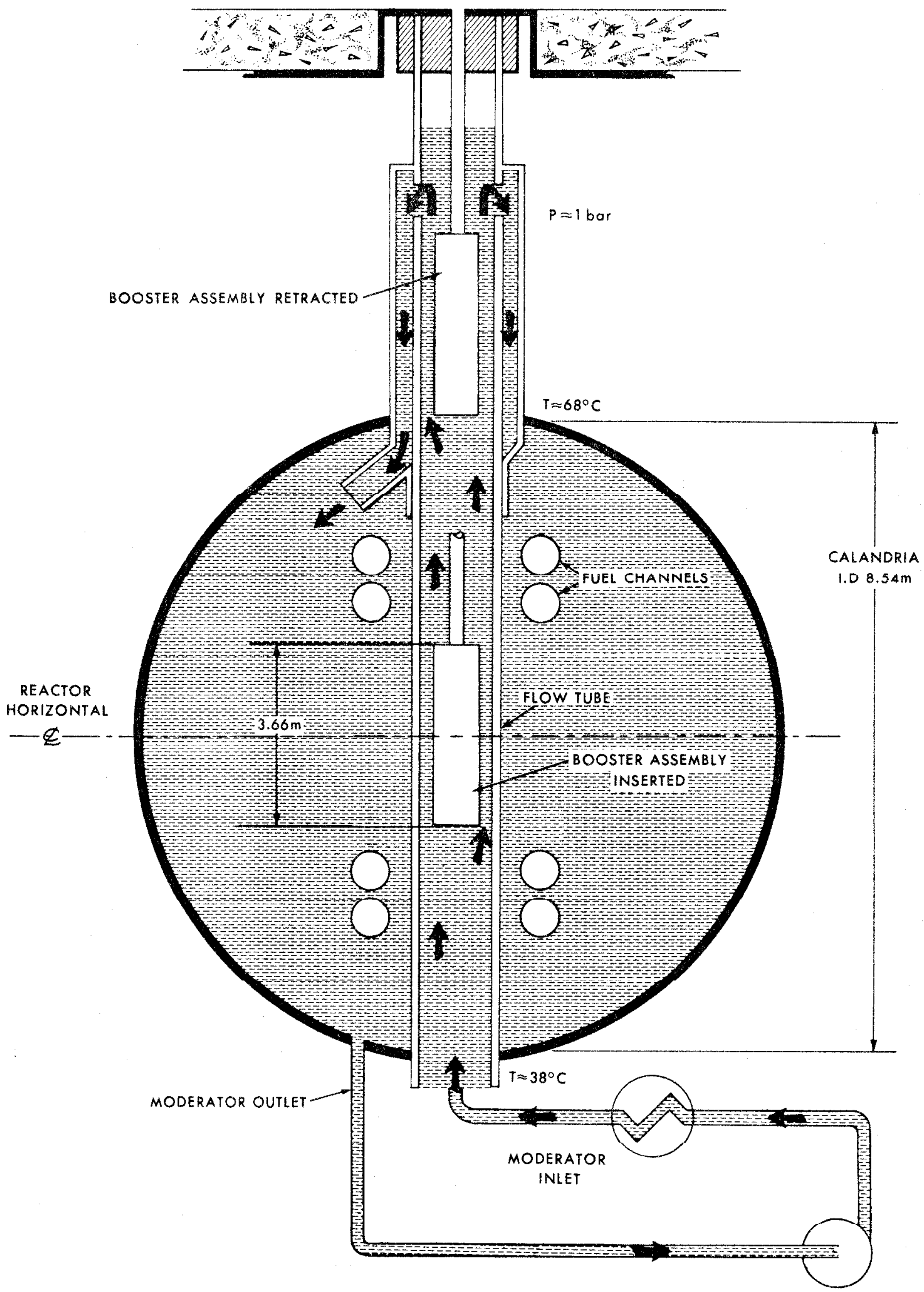


Figure 28 Bruce Booster Channel Schematic

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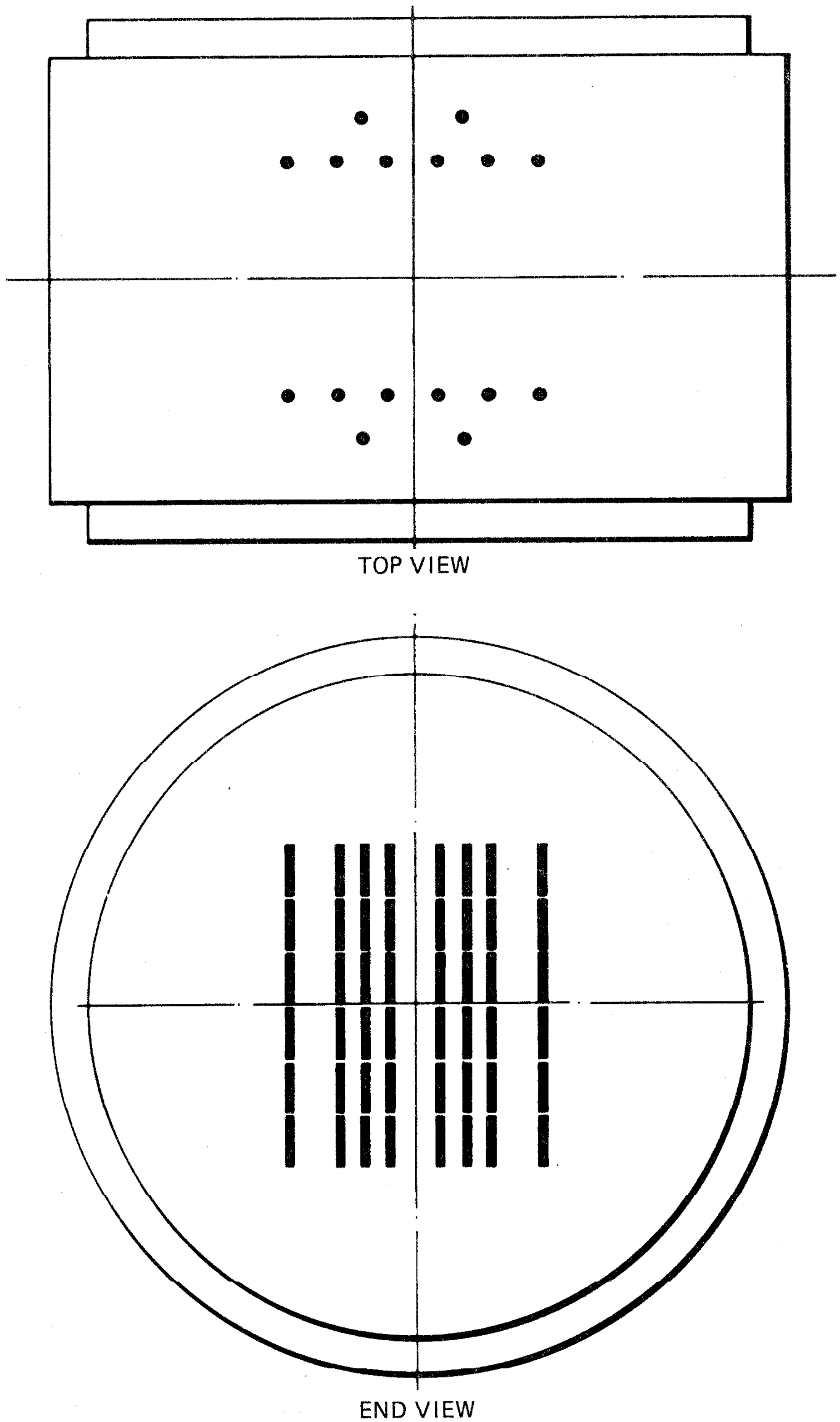

Figure 29 Booster Rod Layout in Bruce

A schematic of the booster channel is shown on Figure 28. The channel is vertical and perpendicular to the reactor channels. The booster fuel is normally stored out of the core at the top, and when it is needed is driven into the core against the coolant flow. It is seen that at the top of the channel the coolant pressure is atmospheric and at power there is no boiling. After passing through the channel the D_2O coolant goes into the moderator.

Only a few of these rods are needed, 16 in Bruce. Figure 29 shows their locations when in service. It should be stressed that booster usage is relatively infrequent, but even so their provision can be justified on economic grounds.

The special feature of booster heat transfer is the low coolant pressure. When water boils, the change in specific volume is very large, particularly at low pressure. The ratio of the specific volume change to the enthalpy change on vapourization is shown on Figure 30. Relative to its value at primary circuit pressures the ratio increased by a factor of about 50 at atmospheric pressure. If there is boiling or surface bubble generation (often called subcooled boiling), the steam

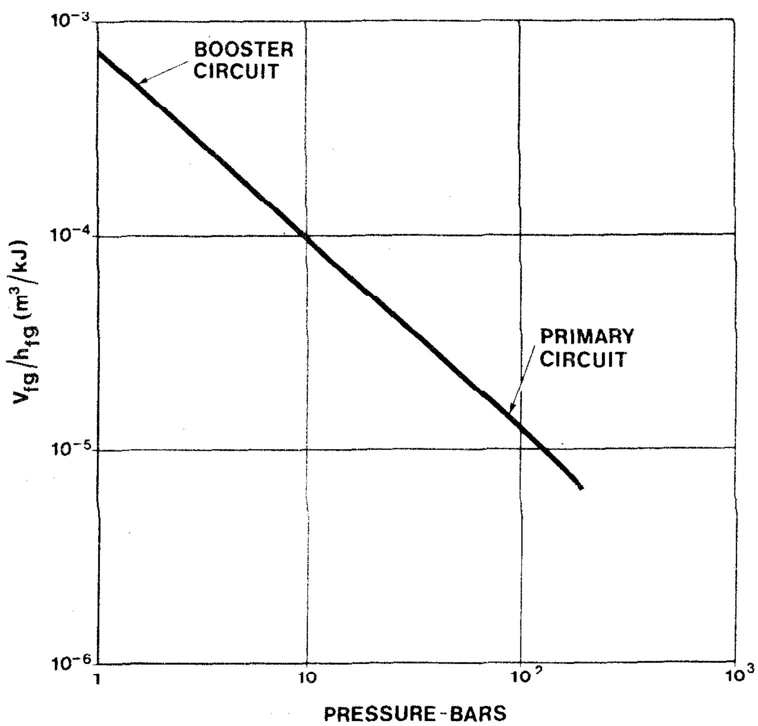

Figure 30 v_{fg}/h_{fg} vs. Pressure

generation can cause a large pressure drop that will tend to reduce the flow. The power which will lead to the onset of significant void (OSV) is determined by correlations that are based on experimental data of pressure drop.

Figure 31 shows results calculated for the Bruce booster. The critical parameter is the coolant subcooling, the lower the coolant temperature, the higher the critical heat flux. A dependence on coolant velocity has also been found.

These results indicate a critical power ratio of about 1.7. To be conservative, it is assumed that at OSV the flow decreases enough to cause DNB. From the figure it is noted that if the flow did not decrease, the DNB critical power is much higher.

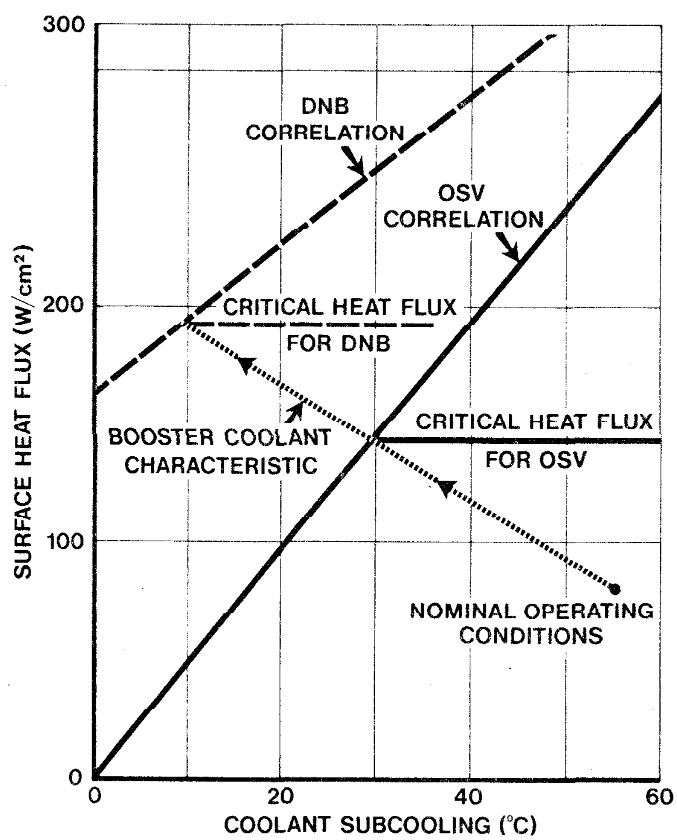

Figure 31 Onset of Significant Void (OSV) and Departure from Nucleate Boiling (DNB)

9. CHANNEL INSTABILITY

A further consideration which can set a fuel channel power limit is channel instability. This is a fluid dynamic problem which is characterized by unsteady flow. Under the conditions in which the problem is met in reactor design, the phenomena is exhibited by periodic flow oscillations. The oscillations by themselves do not impose a limit, but if their magnitude is large enough, premature dryout can occur. This is the concern.

In a system like CANDU, there are always many channels connecting common headers. This is shown schematically in Figure 32. The hydraulic and thermal characteristics of the channels may vary considerably. Because of the very low resistance to flow in the header, the pressure drop across all channels is equal.

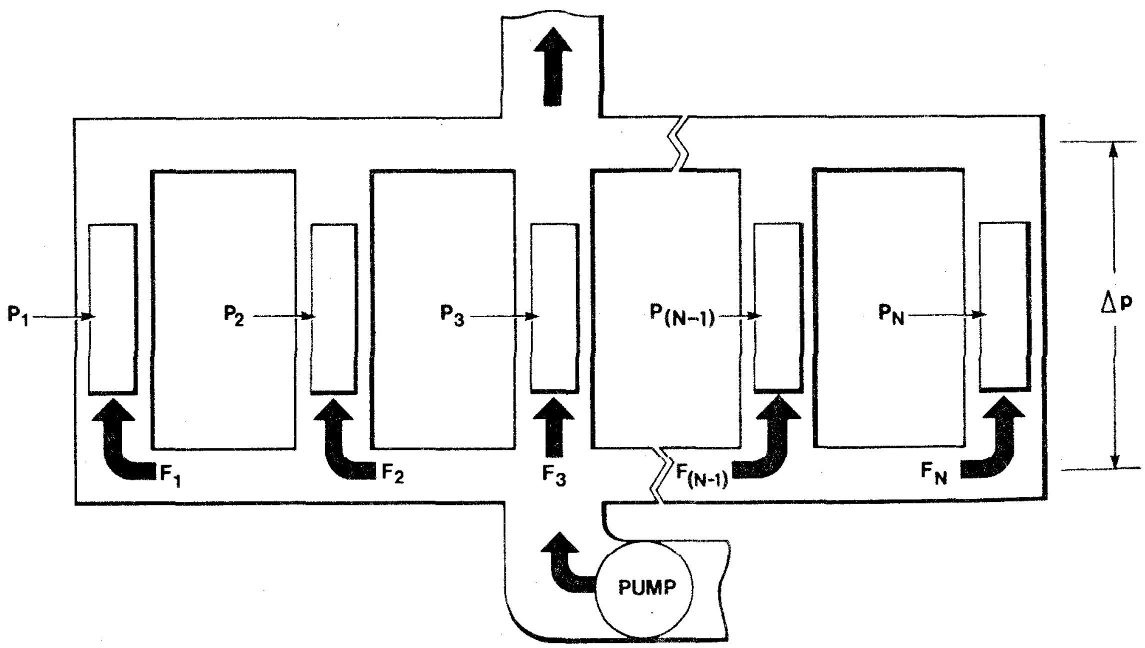

Figure 32 Schematic of Parallel Channel System

The analytical problem is to determine the channel power which will lead to the onset of flow oscillation, or in its most dramatic form, flow divergence. Analytical models have been developed and checked against experiments for this prediction. In terms of the physical plant, the problem corresponds to the local overpower of a few channels without a corresponding increase in total power.

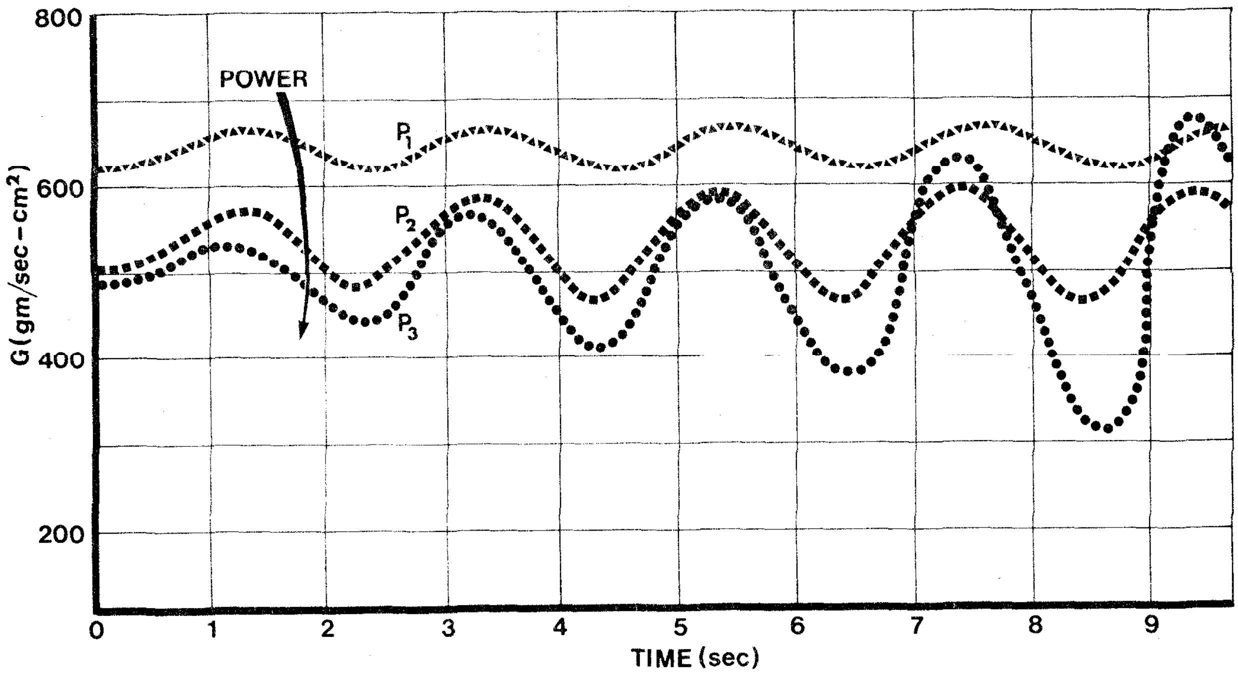

Figure 33 Typical Flow Oscillations

Figure 33 shows typical results as predicted by an analytical model. As the power increases more steam is generated, and the pressure drop would increase if the flow remained constant. Since the pressure drop is held constant the flow must decrease. Eventually channel conditions are reached when the flow oscillates, with ever increasing magnitude. The frequency of the oscillation is typically around 1 Hz although channel characteristics can affect this.

There are well established relationships between flow instability and certain parameters. Figure 34 indicates that increasing heat input, steam quality and channel exit flow resistance all decrease the stability margin. On the other hand, increasing the channel inlet flow resistance or the coolant pressure increase the margin.

In attempting to understand the phenomena it has been found that when the two phase pressure drops are sufficiently large and exhibit sufficient phase lags relative to the single phase pressure drops, then the flow becomes unstable, at least in a small signal sense. In current CANDU designs the two phase pressure drop that is most important is that due to friction.

PARAMETER	STABILITY MARGIN	
	Increases	Decreases
Increased heat input		X
Increased exit quality		X
Increased exit resistance		X
Increased inlet resistance	X	
Increased system temperature	X	

Figure 34 Effect of Various Parameters on Stability Margin

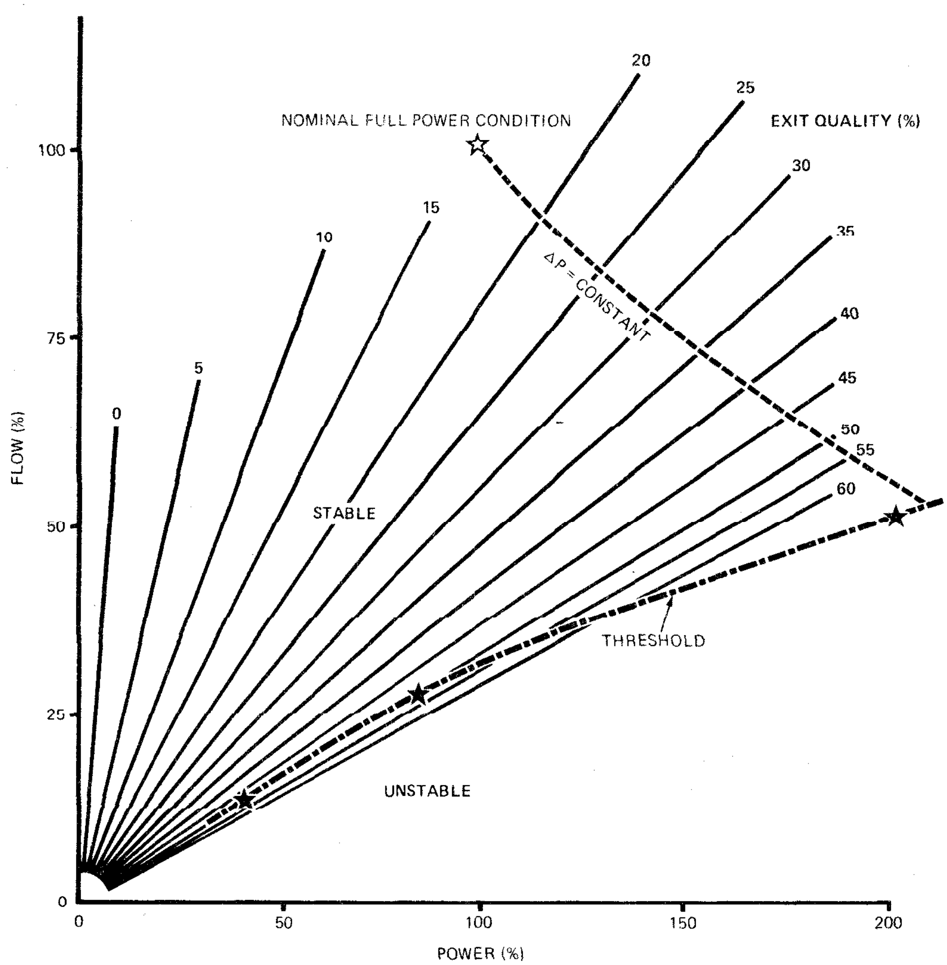

Figure 35 Flow Stability Map for Gentilly

Figure 35 shows calculations for the Gentilly central channel. The threshold line follows approximately a constant quality line. This stability map can be interpreted in several ways. If the channel power is increased then it must reach over 200% of nominal before the instability threshold is reached.

On the other hand, if the total flow were decreased to about 30% of nominal, at nominal power, it would also reach an instability threshold. These margins are quite adequate and impose no operating limitations.

Because the margin to flow instability in Gentilly is very large, there is no interaction with the dryout heat transfer limitation. However, we have found that in other designs both phenomena can occur at roughly the same overpower. In these cases the interaction can lead to an earlier onset of dryout, often called periodic dryout, a feature that must be recognized in the design.

An example of how significant the effect can be is shown by the experimental results on Figure 36. The ratio of the power for periodic to steady-state dryout is plotted as a function of the ratio of outlet to inlet pressure drop. Increasing the latter parameter leads to a reduction in the flow stability threshold. At the lower values, dryout occurs prior to the onset of flow oscillation, but at the higher values periodic dryout occurs first.

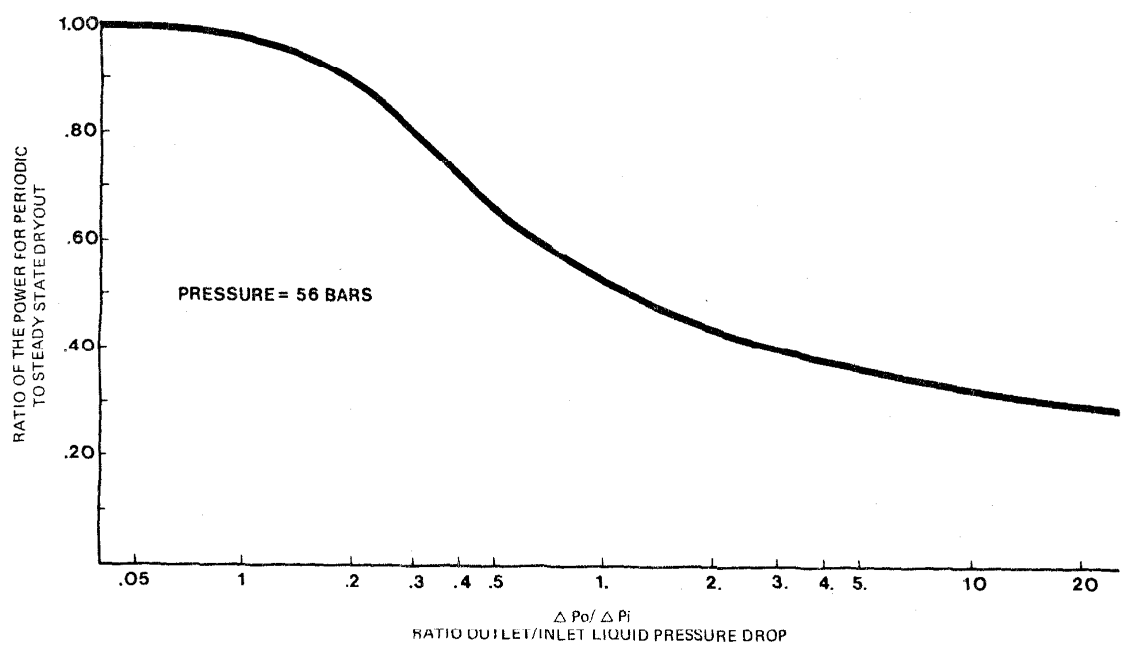

Figure 36 Periodic Dryout

10. CONCLUSION

I have touched on only a few of the areas where analysis is provided in support of the engineering design. Perhaps the most notable trend that is being noticed, and one that I would like to point out in conclusion, is that the heat transfer and fluid dynamic behaviour of CANDU reactor systems is rapidly becoming a very well understood field. This makes it possible to provide the analysis required to guide engineering towards ever improving designs.