NUCLEAR SAFETY AND RELIABILITY

WEEK 7

TABLE OF CONTENTS - WEEK 7

1. Introduction

This segment of the course presents some of the basic technical characteristics of power reactors under accident conditions. It must be remembered, however, that safety in operation depends more on human behavior than on the details of equipment design. The best design cannot compensate completely for weaknesses in operating staff; on the other hand, a first-class operating staff can make up for many weaknesses of any given plant design.

The overall goal is to achieve defense in depth. This concept is based on the recognition that mechanical failures and human errors will occur in all aspects of the project. Design, construction, and operating policies are structured to achieve a high degree of tolerance to such errors, as well as to minimize their frequency of occurrence.

The technical aspects of reactor safety involve the behavior of the plant under failure conditions. The failure conditions chosen for study cover a wide probability range from those expected "every day" to failures that might be expected once in a billion years. The chosen failure scenarios do not pretend to represent all of the actual failures that might occur - rather, they are chosen for the express purpose of testing the capabilities of the installed safety systems - to test the defenses. Actual failure sequences are virtually certain to be different from the test scenarios in many respects.

2. Technical Aspects of Plant Safety

Aside from the unique way in which heat is produced (i.e. the fission process), a nuclear power reactor is quite a simple system. It has, however, a few weak safety characteristics that must be considered. The main ones are:

- Wastes (the dangerous radioactive materials) are contained in the fuel through its life.
- If fuel is severely overheated, the fission products will be released.
- The rate of energy release in fuel is very rapid if the reactor is near to or above prompt critical.
- Some subsystems are "fragile"; that is, they operate quite near to their strength limits.

These characteristics lead directly to a definite requirement for high grade, competent safety systems in the plant. The specific safety design needs are different for each plant type; some plants rely less on engineered safety systems and more on characteristics intrinsic to the design. Generally, plants with greater intrinsic safety are considered to be "safer", but this claim usually is true over only part of the accident spectrum.

Table 7.1 shows some of the characteristics of current power reactor designs. Thermal conversion efficiency is indicated by the ratio of electrical to thermal core power. Core height is the dimension parallel to coolant channels; the equivalent cylindrical core diameter is calculated from the total cross-sectional area of fuel lattice cells. The last column is normalized to the FBR output of 1200 megawatts electric. It represents approximately the core volume required to produce 1200 MWe from a reactor of the given type. Power density gives a very rough indication of the decay heat removal required per unit volume after shutdown in order to prevent core damage. Several other factors must be considered, such as the heat capacity of the core structure (e.g. moderator graphite in GCR's), heat transfer between fuel and structure or coolant, and the margin to phase change of fuel, structure, and coolant. Detailed accident analysis must be performed to obtain the response of any given reactor to various potential failures that might lead to release of fission products.

Table 7.1- Typical design parameters

Plant Type	Output	Output	Power Density	Core Diameter	Core Height	Volume	Normalized Volume
	(MWt)	(MWe)	(MWt/m^3)	(m)	(m)	(m^3)	(m^3)
Magnox	1880	590	0.86	17.4	9.2	2190	4455
AGR	1890	625	3.4	9.3	8.2	557	1069
RBMK	3200	1000	4.2*	11.8	7.0	766	919
THTR	738	300	5.0	5.6	6.0	148	592
HTGR	837	330	6.3	6.0	4.7	133	484
PHWR	2060	625	11.7*	6.3	5.9	183	351
BWR	2570	907	50.0	4.2	3.7	51.3	68
PWR	3500	1140	100.0	3.4	4.0	36.3	38
FBR	3200	1200	500	3.0	0.92	6.5	6.5

* This is the power density averaged over the whole core volume. For comparison with other concepts, this value makes sense only as a scale for the ultimate heat sink capability. A better measure for the power density of a PHWR or RBMK core, from the safety point of view, is that relative to the volume of the pressurized portion of the core. These values are 107 MW/m^3 and 44.1 MW/m^3 respectively

Inherent Safety Characteristics

Inherent safety characteristics are those basic features of a particular design that make it either easier or more difficult for the designer to achieve three basic safety objectives -- (a) protection of the public, (b) protection of the operating staff, and (c) protection of the plant. A "good" characteristic is one that makes engineered safety easier to achieve; a "bad" characteristic makes it more difficult. Combinations of characteristics also are important in some cases. The major characteristics are listed in Table 7.2.

It is obvious that many of these "good" characteristics conflict with the goal of sound economics. The designer then must choose a reasonable balance between safety objectives and economic objectives. The best design from a safety point of view probably never will be built because it will be too expensive. At this point the society must choose the appropriate balance between risk and economic benefits; alternative energy supplies carry different safety and economic pros and cons. Demands for perfectly safe energy supplies result in no energy supply at all; demands for extremely cheap energy supplies likely will lead to high levels of health risk.

Table 7.2 - Safety-Related Features of a Reactor Design

FEATURE	PREFERRED VALUE
Fuel heat flux	low
Stored heat in fuel/unit mass	low
Added energy margin to fuel damage threshold	high
Reactivity change due to fuel heating	negative
Coolant voiding reactivity change	Small positive or negative
Power coefficient of reactivity	small negative
Prompt neutron lifetime	long
Spatial stability of power density	high
Alternate heat sink on loss of coolant	1 or more
Coolant pressure	low
Coolant pressure rise with energy to coolant	small
Primary containment vessel size	small
Access to reactor core for detection & shutdown	direct
Fuel/coolant/ moderator chemical reactions	none

Fuel Behavior

All of the designs listed in Table 7.1 employ solid fuel pins surrounded by metal sheaths. A general safety objective for solid-fuel reactors is to **keep the fission products inside the fuel sheath** following "credible" accidents. A credible accident is one that the people consider is important enough to demand safety design action, both in terms of the size of consequences and the chance of their occurrence.

It is necessary to look first at fission product behavior during normal operating conditions. All fission products are, of course, formed within the fuel grains; they are trapped there (with the exception of a few which recoil directly into inter-grain and gas gaps) until they are released by diffusion. The volatile fission products that are released form a gas mixture inside the fuel sheath. Figure 7.1 indicates the distribution of fission products. It is useful to categorize these in three groups: (a) bound inventory, (b) grain boundary inventory, and (c) gap inventory.

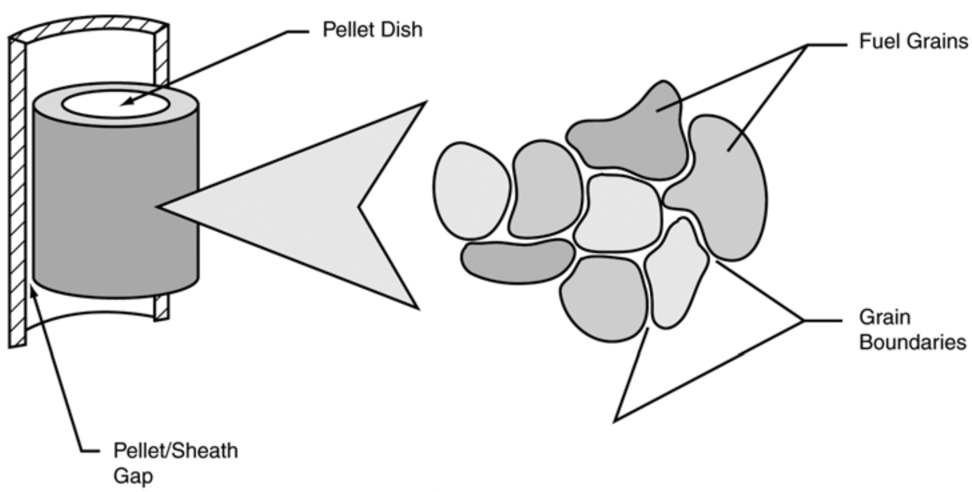


FIGURE 7.1
LOCATIONS OF FISSION PRODUCT GASES

The gap inventory includes pellet dishes, pellet/clad gap and sheath end cap inventories. If the fuel sheath fails the gap inventory escapes quickly, the grain boundary inventory much more slowly, and the bound inventory even more slowly. At relatively low fuel temperatures the diffusion processes are very slow, so that almost all isotopes remain in the grains. (An exception to this occurs when uranium dioxide is exposed to air at moderate temperatures; in this case oxidation to higher states takes place. The fission products are then released.) As temperature and fuel burnup increase the amount of fission gas in the gap increases, as shown in Figure 7.2 for oxide fuel.

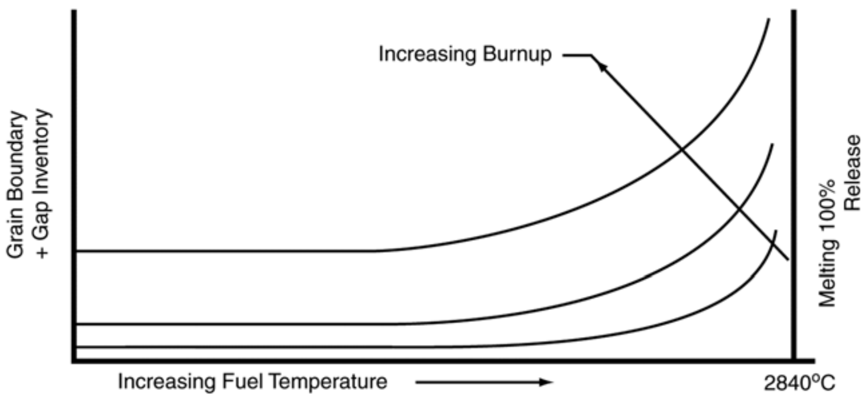


FIGURE 7.2
DEPENDENCE OF FREE FISSION PRODUCT INVENTORY ON BURNUP AND TEMPERATURE

At any given burnup, a larger fraction of the fission product gases are released near the center of the pellet, where the temperature is highest. All volatile fission products tend to migrate down the temperature gradient toward the outside of the pellet. Their diffusion is assisted by the fact that the pellet cracks under the influence of temperature gradient; this cracking increases with fuel burnup and center temperature. At high burnup (and relatively low initial sintered fuel density) a hole develops at the center of the fuel pellet.

In summary:
(a) at low burnup or temperature, nearly all fission products are trapped in fuel grains,
(b) fraction of total f.p. located at grain boundaries increases with temperature and burnup,
(c) the gas gap inventory increases steadily with fuel temperature and burnup.

Fuel Element Temperatures - A typical radial distribution of temperatures within an oxide fuel element is shown in Figure 7.3.

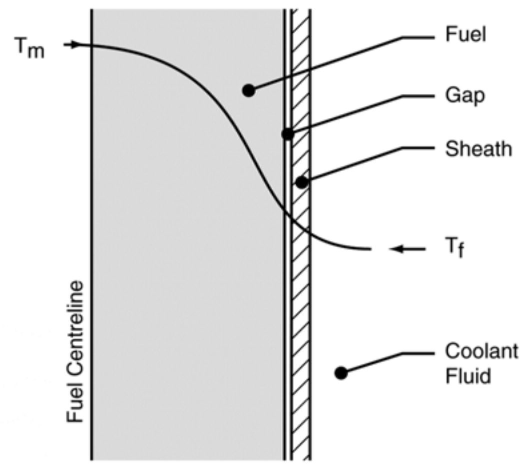


FIGURE 7.3
FUEL ELEMENT RADIAL TEMPERATURE DISTRIBUTION

If we assume for simplicity that the thermal conductivity is independent of fuel temperature in uranium oxide and Zircaloy, the temperature distribution in the fuel (assuming constant heat generation) is parabolic. The relationship between fuel and bulk fluid temperatures and linear heat rating of the fuel is given by the equation:

$$q'' = \frac{T_m - T_f}{R_{mf}}$$

where the thermal resistance is given by:

$$R_{mf} = \frac{1}{4\pi k_f} + \frac{1}{2\pi h_{gap} R} + \frac{1n\left[1 + 3/(R+g)\right]}{2\pi k_g} + \frac{1}{2\pi h_{film}(R+G+s)}$$

and: k_f = fuel conductivity
k_g = sheath conductivity
h_{gap} = gap heat transfer coefficient
R = fuel radius
g = gap thickness
s = sheath thickness

Typical values in CANDU fuel are (in order of appearance above): 0.004 and 0.017 kW/m.°C, 7 and 60 kW/m^2.°C, 6.071,0.04, and 0.42 mm. Gas coolants have much lower, and liquid metal coolants have much higher, film heat transfer coefficients. This is the basic reason for the variations in design power density seen in Table 7.1. Heat capacity does not enter the steady-state equations, but is important in transients. The values for uranium dioxide and Zr-4 are, respectively, 0.5 and 0.4 J/g.°C. The corresponding heats of fusion are 27 and 42 J/g.

The Integral Fast Reactor (IFR) concept created at Argonne National Laboratory utilizes zirconium-stabilized uranium-plutonium metal fuel in place of oxide. The thermal conductivity of this alloy is .04 kW/m.^{o}C, an order of magnitude larger than that of oxide. Melting temperature of the metal fuel is, however, much lower - 1100 ^{o}C versus 2800 ^{o}C. The irradiation stability characteristics of metal fuel are not as good as those of oxide; pins swell and distort at relatively low burnup. This problem has been solved by leaving a large gap between fuel and sheath. The gap is filled with sodium to improve thermal conductance. In this way, metal fuel can be designed to operate at linear heat ratings similar to oxide designs.

Gas Pressure Inside Sheath - CANDU fuel has a "collapsible" fuel sheath, which creeps down onto the pellet during irradiation due to the excess external coolant pressure. The small, enclosed gas space in the element results in a high sensitivity of gas pressure to fuel sheath geometry. A small amount of fill gas is added to this space on assembly, so as to achieve the proper sheath stress distribution during operation. The dependence of internal gas pressure on burnup is shown in Figure 7.4.

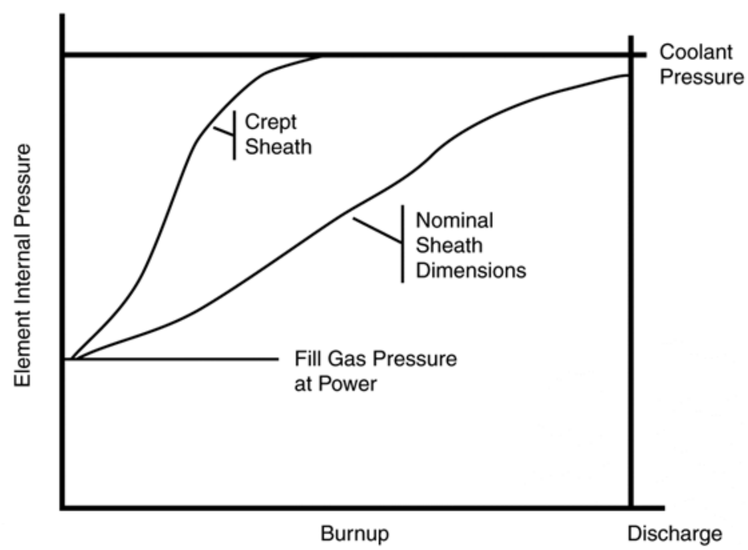


FIGURE 7.4
INTERNAL GAS PRESSURE vs BURNUP

If the internal pressure exceeds coolant pressure, the gap heat transfer coefficient decreases because the sheath creeps away from the fuel pellet. This decrease leads to higher peak fuel temperature, greater fission gas release from the fuel, and finally higher gas pressure. A new equilibrium point is reached.

In reactors with high burnup, the internal gas pressure is controlled by adding a free space or "plenum" above the fuel stack. This requires thicker sheaths in order to control creep stress in the plenum region. All fuel sheaths interact with the pellet to some degree during operation; this interaction was the cause of some sheath failures during early CANDU operation. The pellets tend to form a "hourglass" shape as shown in Figure 7.5. The pellet corners "bind" against the sheath and induce higher than normal stresses. This effect increases with burnup and fuel temperature. In CANDU, this sheath failure mechanism has been controlled by addition of a thin lubricating layer on the I.D. of the sheath, using graphite or silicon.

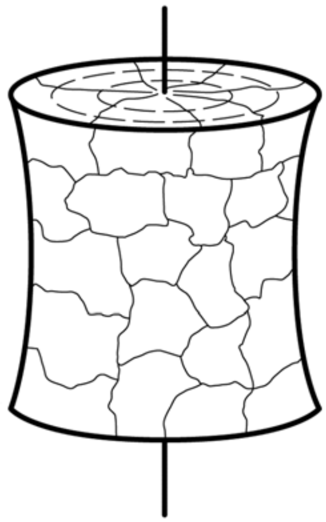


FIGURE 7.5
IRRADIATED FUEL PELLET SHAPE

Fuel Sheath Characteristics - Irradiated Zircaloy-4 is much harder and less ductile than the annealed material, due to neutron bombardment during operation. This effect is very similar to cold-working. The coolant flowing past the elements also contains some free oxygen (from radiolysis). This produces a brittle oxide layer on the O.D. of the sheath. The accompanying free deuterium diffuses into the sheath and further decreases ductility. (It is important to maintain very low hydrogen content inside the element - if hydrogen is present the I.D. can become sufficiently brittle to produce sheath failure.) Zircaloy anneals and softens at about 700 ^{o}C; above this temperature it will distort freely to high strain before failure.

3. Accident Conditions

During normal operation a CANDU-600 core contains 4560 fuel bundles, each with its unique conditions of location, burnup, and power level. On initiation of accident conditions each will behave differently depending on these parameters and the specific circumstances of the accident sequence. The end objective of in-core accident analysis is to estimate the quantity, characteristics, and timing of fission products released to the containment space. Obviously, a number of approximations will be required to obtain an estimate of this release. To introduce this topic, consider the effects on fuel of three abnormal conditions which might be encountered during an accident; (a) overpower, (b) low coolant flow, and (c) loss of coolant.

(a) Overpower

Sheaths might fail due to internal gas pressure or impingement of molten fuel, if the overpower is severe. The equation for the steady-state power level at which centerline melting begins is:

$$\dot{q}\, melt = \left[\frac{T_{melt} - T_f}{R_{mf}} \right]$$

If a sudden power increase is imposed, so that there is no time for additional heat removal, the melting point is reached when an amount of energy is added given by:

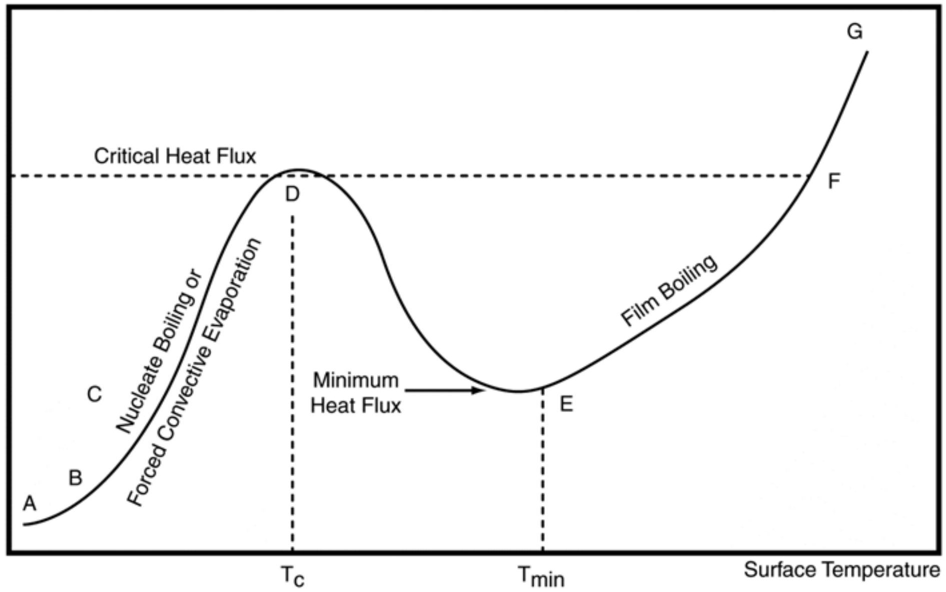
$$\delta\left(\dot{Q} / m\right) = \delta \dot{q}\, t / m = C_p \left(T_{melt} - T_m\right)$$

where:

m = mass of fuel per unit length C_p = specific heat of fuel

Experiments in pulsed test reactors have been conducted to determine the fuel stored energy levels which lead to element rupture; in general, these are somewhat lower than those required to produce gross melting, especially for highly irradiated fuel.

Dryout - This term is used loosely to identify a sudden drop in sheath to coolant heat transfer coefficient that occurs when a "critical heat flux" is exceeded. This phenomenon occurs, of course, only in liquid cooled reactors; gas reactors have no corresponding characteristic. Figure 7.6 shows the characteristic shape of the convective heat transfer curve for water as a function of surface heat flux; this curve is drawn for saturated liquid conditions in the fluid. On dryout, the sheath surface temperature jumps suddenly to the value at which the temperature drop is sufficient to transfer the heat under the new film boiling flow regime. In water under the conditions of pressure, enthalpy, and flow typical of a modern PWR this temperature rise can be very large - dryout is roughly equivalent, under these conditions, to "shutting off" the heat transfer from fuel to coolant. Recalling Figure 7.3, with zero heat transfer the temperature distribution adjusts rapidly to a constant value approximately equal to the average fuel temperature before dryout. This average is about 1000°C. A sheath temperature at this level probably produces immediate failure.



Forced Convection Boiling Curve

PPI649 7-6

FIGURE 7.6
BASIC CHARACTERISTICS OF THE "HEAT TRANSFER CRISIS" IN TWO-PHASE COOLANTS

The effect of dryout is strongly dependent on the heat transfer regime of the coolant. Figure 7.7 shows the various regimes as functions of surface heat flux and thermodynamic quality. Occurrence of the Film Boiling regime indicates a dry sheath surface. The physical appearance of these flow regimes is shown in Figure 7.8. This Figure represents a tube heated from the outside with flow on the inside. The heat flux may be assumed constant along the tube. Referring to Figure 7.7, the fluid in this tube can be thought of as traversing the long-dash horizontal line in that Figure. In subcooled nucleate boiling the bubbles collapse as they leave the tube wall and mix with the subcooled liquid.

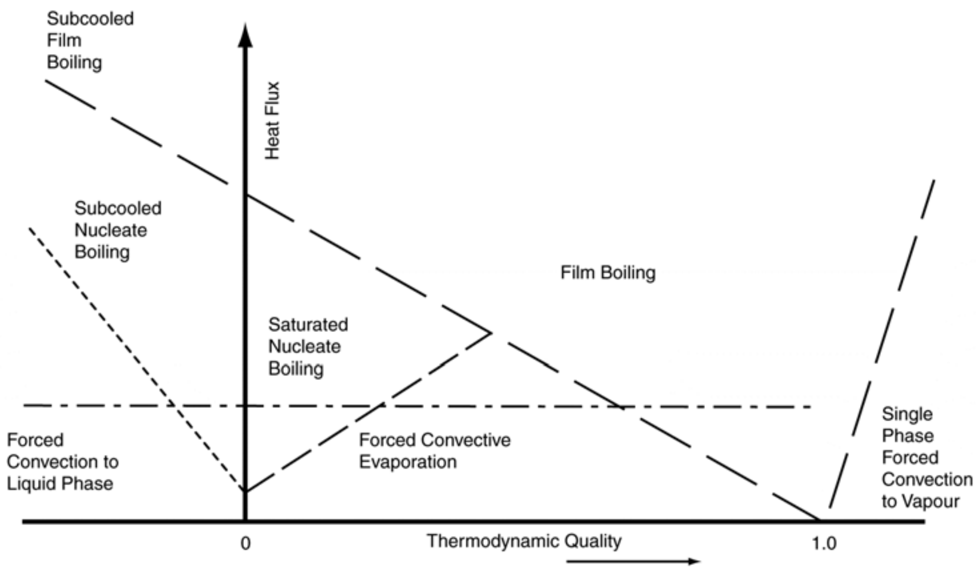


FIGURE 7.7
FLOW REGIMES IN A HEATED TUBE

PPI649 7-7

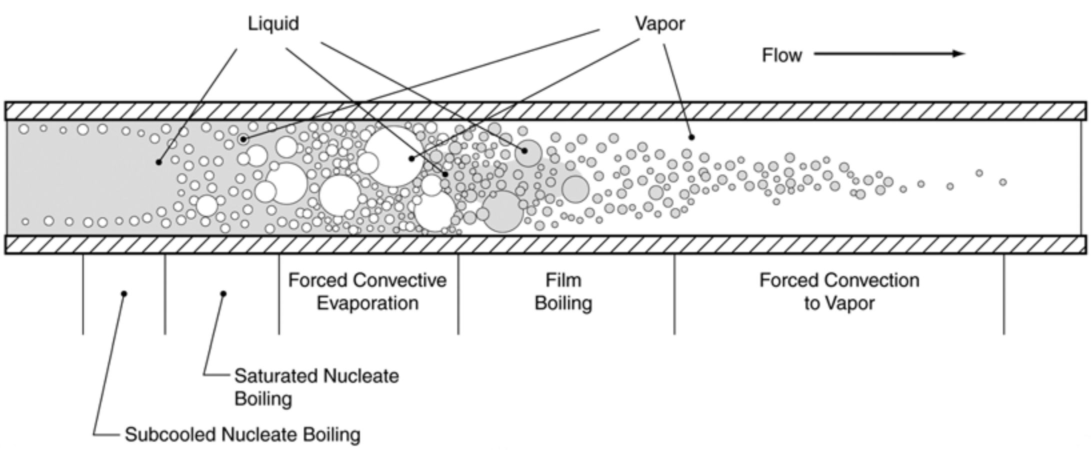
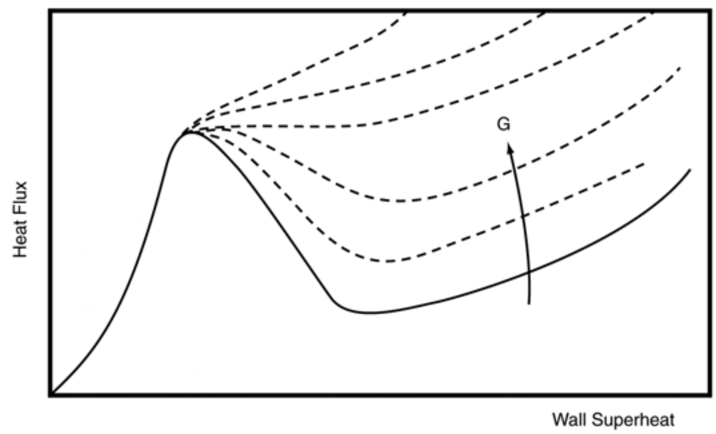


FIGURE 7.8
FLOW REGIMES IN A HEATED TUBE

PPI649 7-8

Vigorous saturated nucleate boiling occurs as the average quality increases, thereby producing high turbulence. As a result the flow regime switches to annular, with a liquid film on the surface from which evaporation takes place to the vapor core. On transition to film boiling the liquid is driven off the surface, leaving a layer of vapor. The liquid droplets then travel in the vapor flow until they eventually evaporate due to heat transfer directly from the vapor. When the quality reaches unity, heat transfer continues to superheated vapor in single-phase flow. From our point of view, the most interesting transition is the one to film boiling, because it is here that the fuel sheath temperature may increase abruptly.

Figure 7.9 shows the qualitative trends of the "wall superheat" (surface temperature - local saturation temperature) as functions of the mass flow rate and quality.



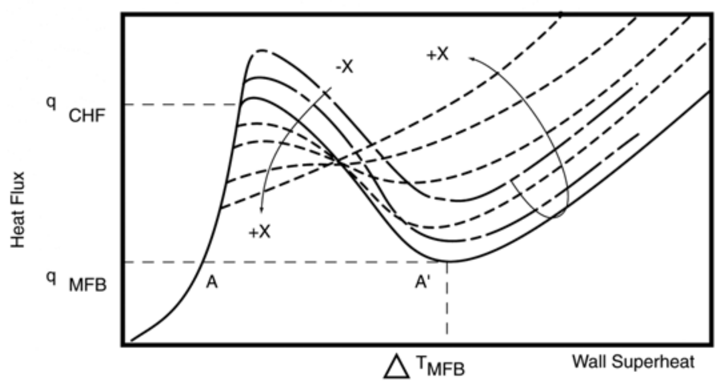


FIGURE 7.9
PARAMETRIC TRENDS OF
CONVECTIVE BOILING CURVE

The first graph in Figure 7.9 shows a decrease in the severity of the wall superheat increase on transition to film boiling, as flow rate increases. This is apparently due to the increasing turbulence near the wall and consequent decreasing vapor film thickness. A similar trend with increasing quality shows in the second graph. The heat flux at dryout decreases with increasing quality, but the transition temperature jump decreases. Physically, this effect might be due to the increasing physical velocity for a given mass flow rate, with its accompanying high turbulence and thin vapor film.

Dryout in a CANDU channel is a very complex process with many factors influencing the actual channel power that should not be exceeded in order to protect the fuel sheath from failure. As an example, consider the distribution of heat flux in the various rings of fuel pins inside a fuel bundle, as shown in Figure 7.10. This distribution results from the fact that most neutrons are thermalized in the moderator, and flow back to the bundle to produce fission. The outer rows of fuel pins absorb a number of thermal neutrons and therefore shield the inner pins. Within each pin there is a fine-structure power distribution caused by the same "self-shielding" effect. In addition, more plutonium is produced from uranium-238 capture near the outer surface of the pin. This plutonium is also fissile, so fissions to produce additional power.

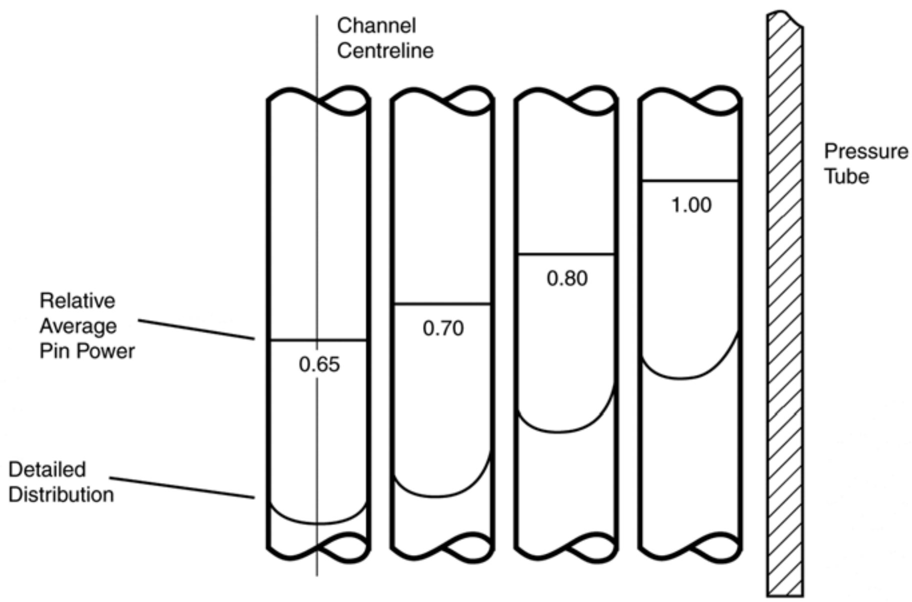


FIGURE 7.10
FUEL PIN POWER DISTRIBUTIONS

These variations in fuel element power are considered in bundle design by increasing the size of inter-pin flow subchannels toward the outer radius, so as to roughly equalize the water enthalpy rise over the length of the bundle.

Axial Flow Effects - As the HT cooling water flows through the fuel channel its enthalpy increases toward the boiling point. The flow itself results in a pressure drop of about 800 kPa; therefore the saturation enthalpy decreases toward the channel outlet. In the 600 MWe CANDU design, boiling starts at around 85 % of the channel length, as shown in Figure 7.11.

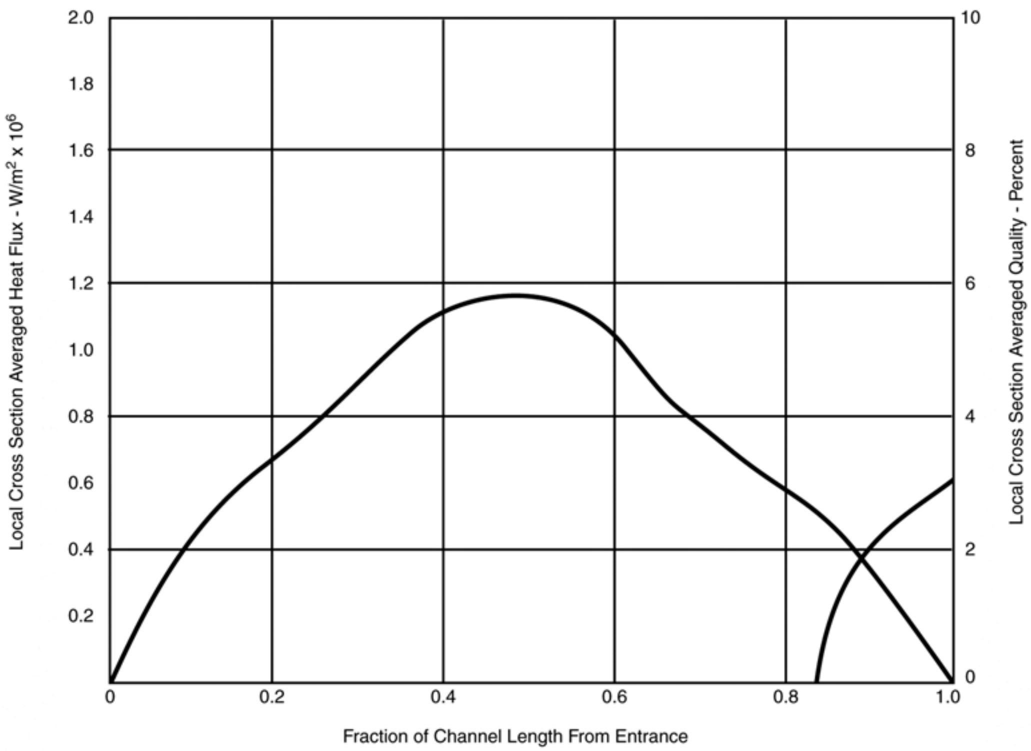


FIGURE 7.11
HEAT FLUX AND CHANNEL QUALITY AT FULL POWER

Two-phase flow exhibits higher pressure drop than single-phase flow for a given mass flow rate (this is plausible because the physical velocity must increase as the average density of the fluid decreases). In a CANDU channel, with approximately constant pressure drop between the inlet and outlet headers, the flow therefore decreases as the exit quality increases (flow is proportional to the square root of the pressure drop).

If we now consider a channel overpower condition, it is obvious that the channel flow will decrease with increasing power. The quantitative effect in the CANDU-600 is shown in Figure 7.12.

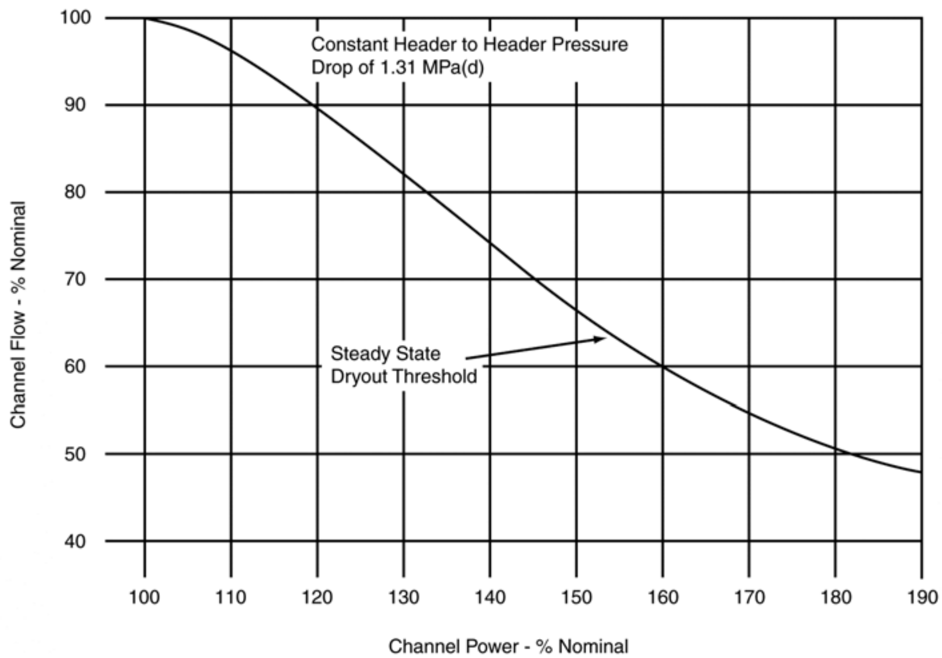


FIGURE 7.12
FLOW AS FUNCTION OF POWER IN A CANDU6 HIGH POWER CHANNEL

The quality and heat flux distributions in the channel at dryout power are shown in Figure 7.13. Dryout occurs at the tangent intersection of the heat flux curve with the critical heat flux curve; this point falls in a region of relatively high fluid quality. Therefore, one would expect that the post-dryout sheath temperature rise would be small; this has been shown experimentally to be the case. This fact is very important to the setting of safety system trip margins - it means that it is relatively <u>unimportant</u> if limited dryout occurs somewhere in the core. Trip setpoints therefore can be set somewhat higher.

The progressive effects of sheath dryout via overpower are listed below; the timing, endpoint, and eventual consequences of this sequence are strongly dependent on the details of the accident sequence being analyzed.

Overpower:

- Dryout
- Sheath temperature rise
- Zircaloy annealing
- Oxidation embrittlement of sheath
- Braze melting and attack on Zircaloy
- Zirconium-water reaction (exothermic)
- Bundle collapse
- Sheath melting
- Fuel melting (extremely unlikely)
- Pressure tube balloon or burst
    - Heat transfer to moderator

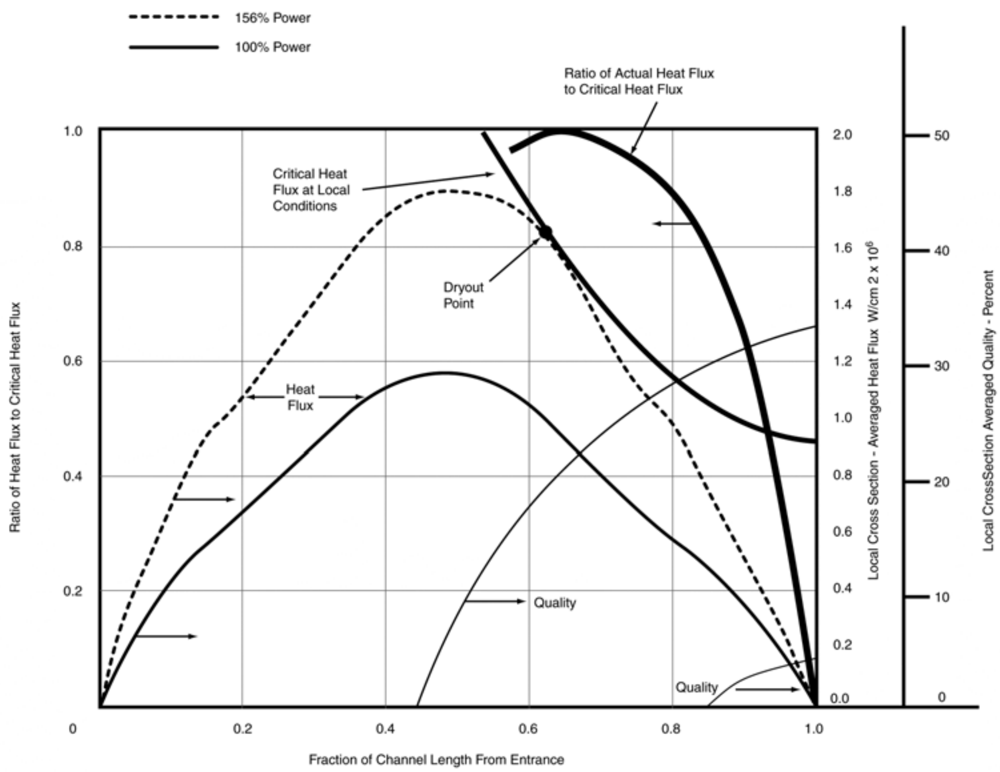


FIGURE 7.13
APPROACH TO FUEL SHEATH DRYOUT IN A CANDU CHANNEL

PPI649 7-13

(b) Low Coolant Flow

This type of accident might involve one coolant channel in a CANDU; flow blockage is the main cause. Loss of forced circulation (pump trip) is another possibility. As in the overpower case there is a flow/power mismatch; that is, more power is being produced in the fuel than can be removed by the coolant. The physical phenomena in the two cases are quite similar. Progressive effects of loss of flow are listed below.

Low Coolant Flow:

- Large increase in exit quality
- HT overpressure
- Dryout
- Sequence as for overpower

(c) Loss of Coolant

This case has characteristics similar to both overpower and low coolant flow. Depending on the location and size of the pipe break, low flow occurs in some subset of the coolant channels in the core. (Loss of pressure control low produces effects similar to a small loss of coolant, but generally less severe.) Loss of coolant results in a system low pressure, with resultant flashing of water in the channels. The increase in quality leads to dryout and fuel overheating.

Loss of Coolant:

- Rapid decrease in local saturation temperature
- Flashing (rapid boiling) proceeds upstream from channel exit
- Neutron overpower due to coolant voiding
- Same fuel sequence as for overpower
- HT pumps cavitate when pressure is about equal to saturation pressure at pump inlet - flow drops rapidly

In all of these situations, the pressure tube can be overheated via conductive, convective, and radiative heat transfer from the hot fuel. If the channel pressure is high the pressure tube might burst; in most cases it will either balloon or sag to contact the calandria tube and transfer heat to the moderator water. There will be more discussion on this topic later in the course.

It is quite obvious that these consequences should be avoided in the interests of protecting the public from potential effects of the radioactive material inside the fuel sheath. The design response to off-normal conditions is outlined in Figure 7.14.

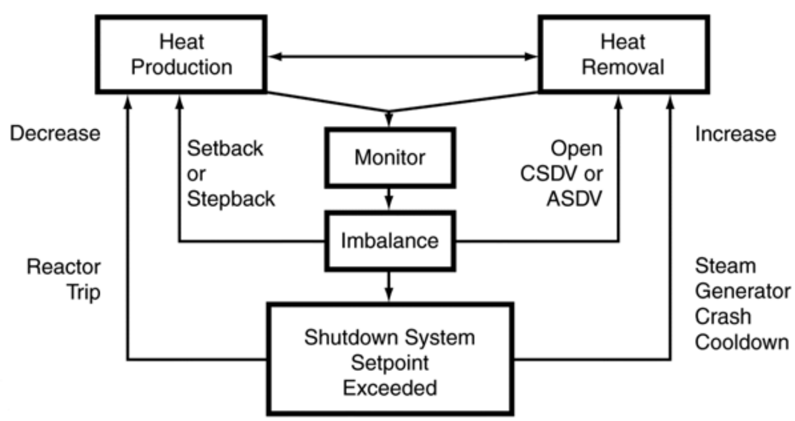


FIGURE 7.14
RESPONSES TO IMBALANCE BETWEEN HEAT PRODUCTION AND REMOVAL

The responses in the inner loop of the Figure are initiated by the reactor regulating system; only if the severity is too large do the independent safety system responses come into play. The particular response depends on the specific circumstances of the off-normal conditions.

In the CANDU 600, SDS1 and SDS2 can be activated by one or more of several monitored parameters:

1. Overpower (general or local)
2. Low coolant flow
3. Low HT pressure
4. High HT pressure
5. High log rate power increase
6. Low pressurizer level
7. Low steam generator level
8. Low steam generator feedline pressure
9. Manual trip

The first three parameters respond directly to conditions that might produce sheath failure. The next five are anticipatory; for example, high HT pressure can result in HT rupture and consequent LOCA, and a high rate of power increase indicates that overpower is imminent. Trips 6, 7, and 8 anticipate impending poor heat removal conditions in the HT system that could lead to HT rupture and consequent LOCA. Emergency coolant injection and steam generator crash cooldown are initiated in response to a LOCA signal. The initiation logic is shown in Figure 7.15.

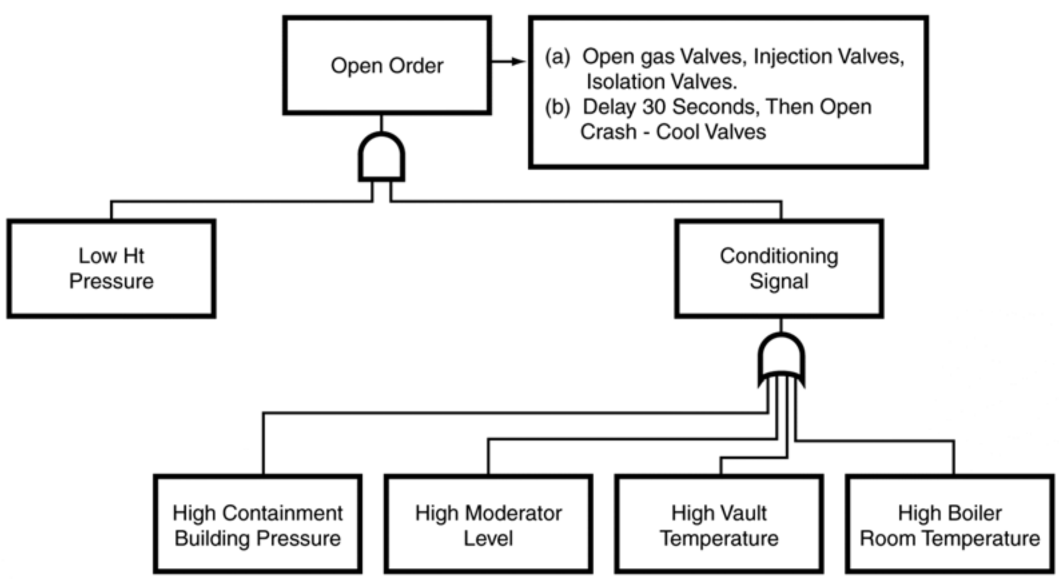


FIGURE 7.15
EMERGENCY COOLANT INITIATION LOGIC

One of the conditioning signals, in combination with the primary indication of low HT pressure (<5.5MPa) is taken to indicate a LOCA. Isolation of the HT loops, feed-bleed and purification is done independently at the same pressure.