22.06 Engineering of Nuclear Systems

MIT Department of Nuclear Science and Engineering

NOTES ON TWO-PHASE FLOW, BOILING HEAT TRANSFER, AND BOILING CRISES IN PWRs AND BWRs

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Definition of the basic two-phase flow parameters

In this document the subscripts "v" and "ℓ" indicate the vapor and liquid phase, respectively. The subscripts "g" and "f" indicate the vapor and liquid phase at saturation, respectively.

Void fraction:
$$\alpha \equiv \frac{A_v}{A_v + A_\ell}$$

Static quality:
$$x_{st} \equiv \frac{M_v}{M_v + M_\ell} = \frac{A_v \rho_v}{A_v \rho_v + A_\ell \rho_\ell}$$

Flow quality:
$$x \equiv \frac{\dot{m}_v}{\dot{m}_v + \dot{m}_\ell} = \frac{v_v A_v \rho_v}{v_v A_v \rho_v + v_\ell A_\ell \rho_\ell}$$

Slip ratio:
$$S \equiv \frac{v_v}{v_\ell}$$

Mixture density:
$$\rho_m \equiv \alpha \rho_v + (1-\alpha)\rho_\ell$$

Flow quality, void fraction and slip ratio are generally related as follows:

$$\alpha = \frac{1}{1 + \dfrac{\rho_v}{\rho_\ell} \cdot S \cdot \dfrac{1-x}{x}}$$

Figures 1 and 2 show the void fraction vs flow quality for various values of the slip ratio and pressure, respectively, for steam/water mixtures.
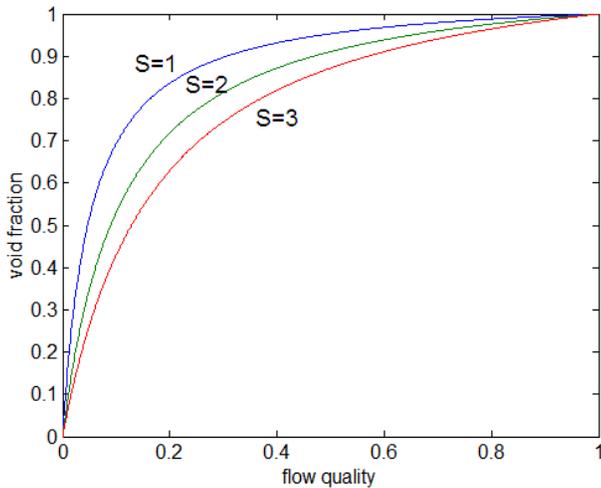
Figure 1. Effect of S on α vs x for water at 7 MPa.
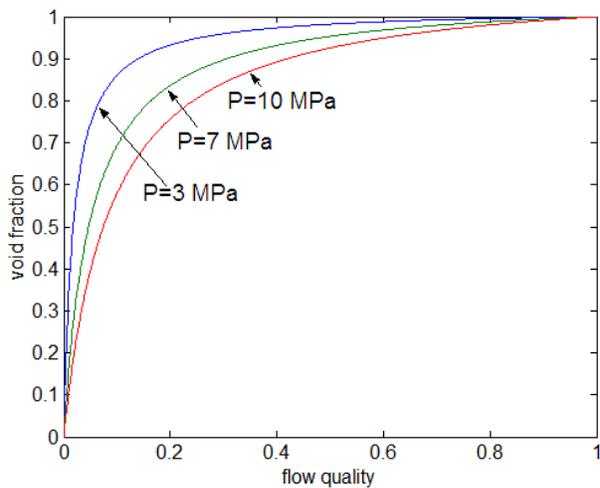
Figure 2. Effect of water pressure on α vs x for S=1.

In 22.06 we assume *homogeneous* flow, i.e. $v_v = v_\ell$ or S=1. This assumption makes it relatively simple to treat two-phase mixtures effectively as single-phase fluids with variable properties. If S=1, it follows immediately that $x_{st} = x$ and, after some trivial algebra, $\dfrac{1}{\rho_m} = \dfrac{x}{\rho_v} + \dfrac{1-x}{\rho_\ell}$.

Note that the assumption of homogeneous flow is very restrictive, and in fact accurate only under limited conditions (e.g. dispersed bubbly flow, mist flow). In general, a significant slip between the two phases is present, which requires the use of more realistic models in which $v_v \neq v_\ell$. Such models will be discussed in 22.312 and 22.313.

Two-phase flow regimes

With reference to upflow in vertical channel, one can (loosely) identify several flow regimes, or patterns, whose occurrence, for a given fluid, pressure and channel geometry, depends on the flow quality and flow rate. The main flow regimes are reported in Table 1 and shown in Figure 3. Note that what values of flow quality and flow rate are "low", "intermediate" or "high" depend on the fluid and pressure. In horizontal flow, in addition to the above flow regimes, there can also be stratified flow, typical of low flow rates at which the two phases separate under the effect of gravity.

Table 1. Qualitative classification of two-phase flow regimes.

Flow quality	Flow rate	Flow regime
Low	Low and intermediate	**Bubbly**
	High	**Dispersed bubbly**
Intermediate	Low and intermediate	**Plug/slug**
	High	**Churn**
High	High	**Annular**
	High (post-dryout)	**Mist**

Typical configuration of (a) bubbly flow, (b) dispersed bubbly (i.e. fine bubbles dispersed in the continuous liquid phase), (c) plug/slug flow, (d) churn flow, (e) annular flow, (f) mist flow (i.e. fine droplets dispersed in the continuous vapor phase) and (g) stratified flow. Note: mist flow is possible only in a heated channel; stratified flow is possible only in a horizontal channel.
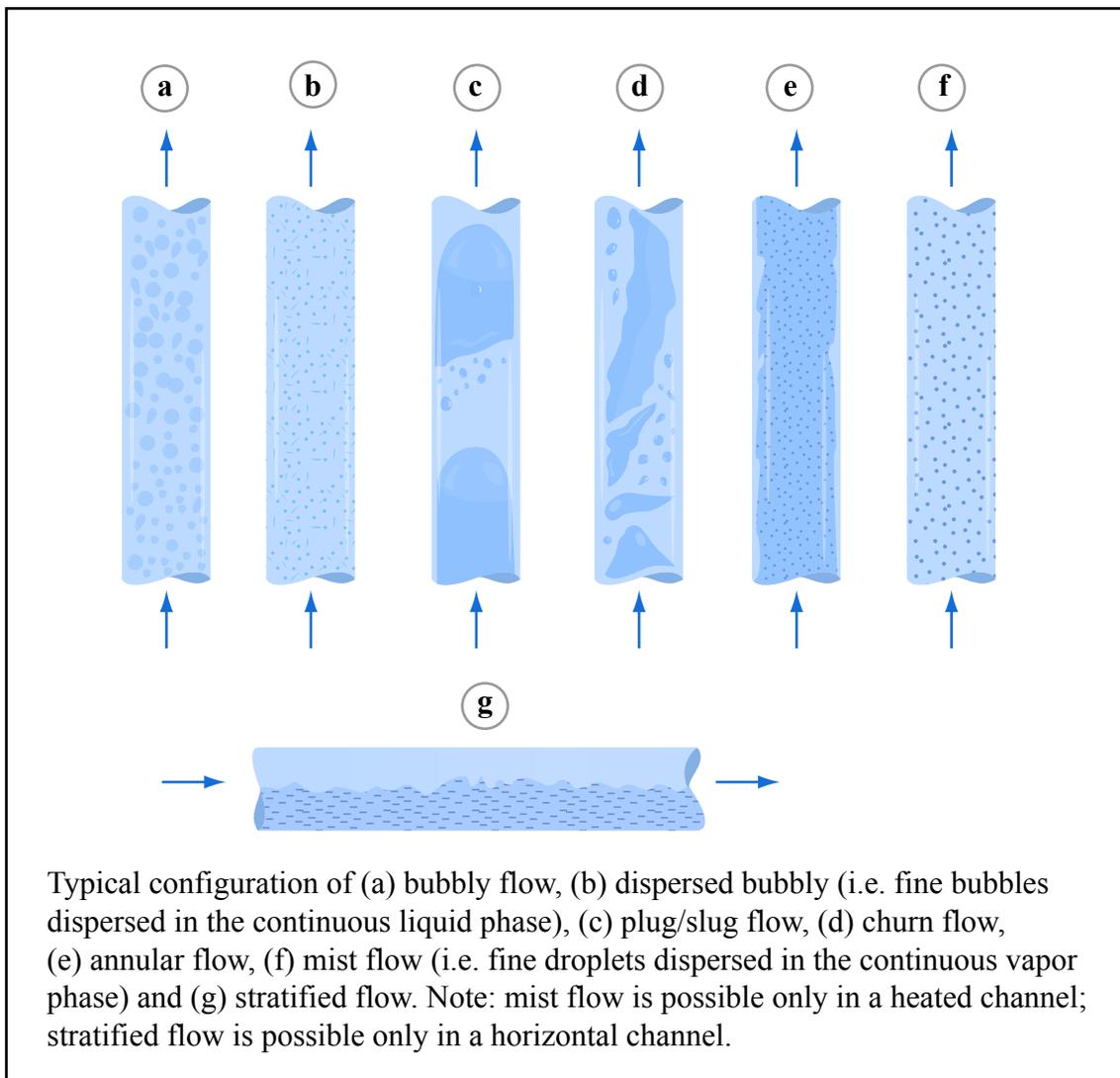
Image by MIT OpenCourseWare.

More quantitatively, one can determine which flow regime is present in a particular situation of interest resorting to an empirical flow map (see Figure 4). Note that such maps depend on the fluid, pressure and channel geometry, i.e., there is no "universal" map for two-phase flow regimes. However, there exist methods to generate a flow map for a particular fluid, pressure and geometry. These methods will be covered in 22.312 and 22.313.
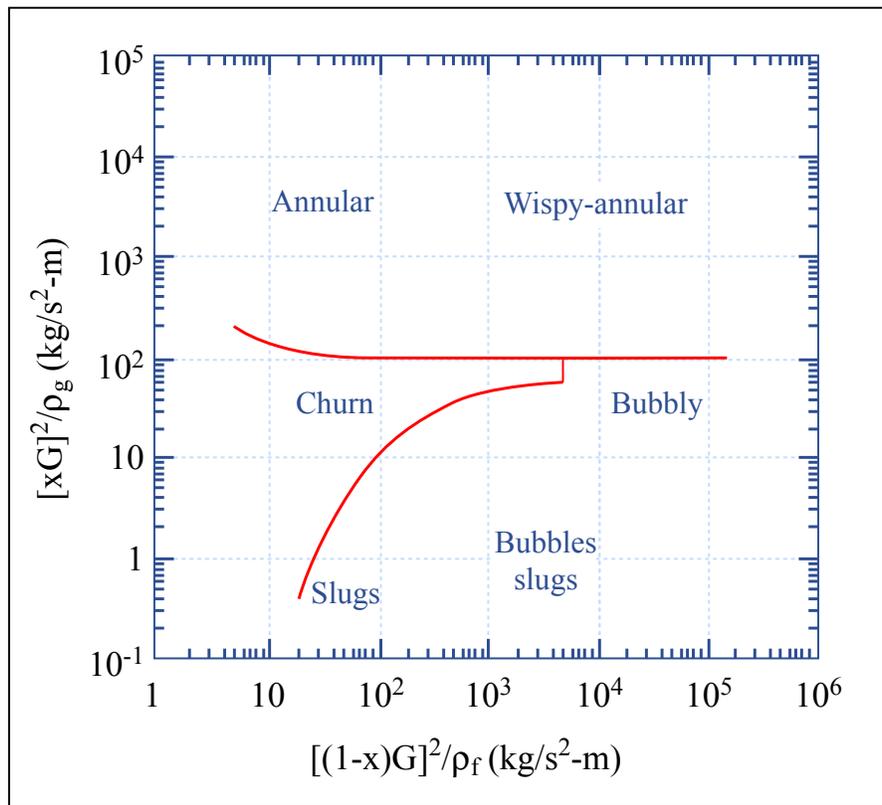
Image by MIT OpenCourseWare.

Figure 4. A typical flow map obtained from data for low-pressure air-water mixtures and high-pressure water-steam mixtures in small (1-3 cm ID) adiabatic tubes (adapted from Hewitt and Roberts 1969).

How to calculate pressure changes in two-phase flow channels

For a straight channel of flow area A, wetted perimeter P_w, equivalent (hydraulic) diameter D_e ($\equiv 4A/P_w$), inclination angle θ, and length L, connected to inlet and outlet plena, the total pressure drop (inlet plenum pressure minus outlet plenum pressure) can be obtained from the steady-state momentum equation:

$$P_0 - P_L = \Delta P_{tot} = \Delta P_{acc} + \Delta P_{fric} + \Delta P_{grav} + \sum_j \Delta P_{form} \qquad (1)$$

where the subscripts "0" and "L" refer to inlet and outlet plenum, respectively. The last term on the RHS of Eq. 1 represents the sum of all form losses in the channel, including those due to the inlet, the outlet and all other abrupt flow area changes in the channel (e.g. valves, orifices, spacer grids).

Under the assumption of homogenous flow (S=1 or $v_v = v_\ell$):

$$\Delta P_{acc} = G^2 \left[\frac{1}{\rho_{m,L}} - \frac{1}{\rho_{m,0}} \right] \qquad (2)$$

$$\Delta P_{fric} = \int_0^L f \frac{1}{D_e} \frac{G^2}{2\rho_m} dz \tag{3}$$

$$\Delta P_{grav} = \int_0^L \rho_m g \cos\theta \, dz \tag{4}$$

Note that the above equations are formally identical to the single-phase case with the mixture density, $\rho_m = \alpha\rho_v + (1-\alpha)\rho_\ell$, used instead of the single-phase density. The friction factor in Eq. 3 can be calculated from a single-phase correlation and using the "liquid-only" Reynolds number, $Re_{\ell o} = GD_e/\mu_\ell$.

For each abrupt flow area change in the channel (again including the inlet and outlet), the associated pressure loss is the sum of an acceleration term and an irreversible loss term:

$$\Delta P_{form} = \Delta P_{form,acc} + \Delta P_{form,irr} = \frac{1}{2\rho_m}(G_2^2 - G_1^2) + K\frac{G^2}{2\rho_m} \tag{5}$$

where the subscripts "1" and "2" refer to the location immediately upstream and downstream of the abrupt area change, respectively.

Boiling heat transfer

Boiling is the transition from liquid to vapor via formation (or nucleation) of bubbles. It typically requires heat addition. When the boiling process occurs at constant pressure (e.g. in the BWR fuel assemblies, PWR steam generators, and practically all other heat exchangers in industrial applications), the heat required to vaporize a unit mass of liquid is $h_{fg} \equiv h_g - h_f$, which can be found in the steam tables.

Writing the (Young-Laplace) equation for the mechanical equilibrium of a bubble surrounded by liquid, it can be shown that the vapor pressure within the bubble must be somewhat higher than the pressure of the surrounding liquid. It follows that the vapor (and liquid) temperature must be somewhat higher than the saturation temperature, T_{sat}, corresponding to the liquid (or system) pressure. Using the Clausius-Clapeyron equation to relate saturation temperatures and pressures, it can then be shown that the magnitude of the superheat ($T_{nucleation} - T_{sat}$) required to sustain the bubble is inversely proportional to the radius of the bubbles, r_{bubble}:

$$T_{nucleation} - T_{sat} \propto \frac{1}{r_{bubble}} \tag{6}$$

Equation 6 suggests that the smaller the bubble, the higher is the superheat required for nucleation. Therefore, if one wants to initiate bubble nucleation near the saturation temperature, one needs relatively large bubbles (order of microns) to begin with. It just so happens that the heat transfer surface of engineering systems (e.g. cladding of a BWR fuel rod, wall of a PWR steam generator tube,

etc.) is full of micro-cavities, naturally present due to roughness, corrosion, etc. Therefore, air or vapor trapped in those cavities serve as *nucleation sites* for the bubbles and ensure initiation of the boiling process. **The bottom line is that one needs to raise the temperature of the surface a little bit above T_{sat} if boiling is to be initiated and sustained.** For a rigorous derivation of Eq. 6 and more details about bubble nucleation, refer to any boiling heat transfer textbook (e.g. *Boiling, Condensation and Gas-Liquid Flow* by P. B. Whalley, Oxford Science Publications, 1987).

Pool Boiling

Consider a simple experiment in which water (or other fluid) boils off the upper surface of a flat plate. The plate is connected to an electric power supply, and thus is heated via Joule effect. This situation is referred to as pool boiling (vs flow boiling) because the pool of fluid above the heater is stagnant. In this experiment we control the heat flux q" (via the power supply) and we measure the wall temperature T_w (via a thermocouple). Then a q" vs T_w-T_{sat} curve can be constructed from the data. This curve is known as the boiling curve and is shown in Fig 5.
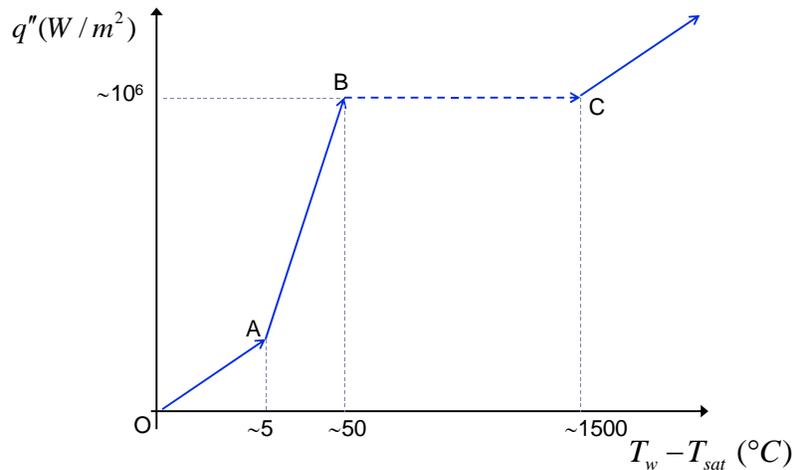
Figure 5. Qualitative boiling curve for water at atmospheric pressure (T_{sat}=100°C) on a flat plate.

As one progressively increases the heat flux, several regions (or heat transfer regimes) can be identified from this curve. A depiction of the physical situation associated with these heat transfer regimes is shown in Fig 6.

O→A
The wall temperature is not sufficiently beyond T_{sat} to initiate bubble nucleation. In this region the heat from the wall is transferred by single-phase free convection.

Point A
The first bubbles appear on the surface. This point is called the onset of nucleate boiling.

A→B

Many bubbles are formed and grow at the wall, and detach from the wall under the effect of buoyancy. This heat transfer regime is called nucleate boiling. The bubbles create a lot of agitation and mixing near the wall, which enhances heat transfer. As a result, nucleate boiling is a much more effective mechanism than free convection, as suggested by the higher slope of the boiling curve in this region.

B→C
At a critical (high) value of the heat flux, there are so many bubbles crowding the wall that they can merge and form a continuous stable vapor film. Thus, there is a sudden transition from nucleate boiling to film boiling. This sudden transition is called Departure from Nucleate Boiling (DNB), or Critical Heat Flux (CHF), or burnout, or boiling crisis, and is associated with a drastic reduction of the heat transfer coefficient because vapor is a poor conductor of heat. Consequently, the wall temperature increases abruptly and dramatically (to >1000°C), which may result in physical destruction of the heater.

C and beyond. This is the film boiling region. Heat transfer from the wall occurs mostly by convection within the vapor film and radiation across the vapor film.

Free convection (O → A)

Onset of nucleate boiling (A)

Nucleate boiling (A → B)

Film boiling (C and beyond)

The heat transfer regimes in pool boiling.
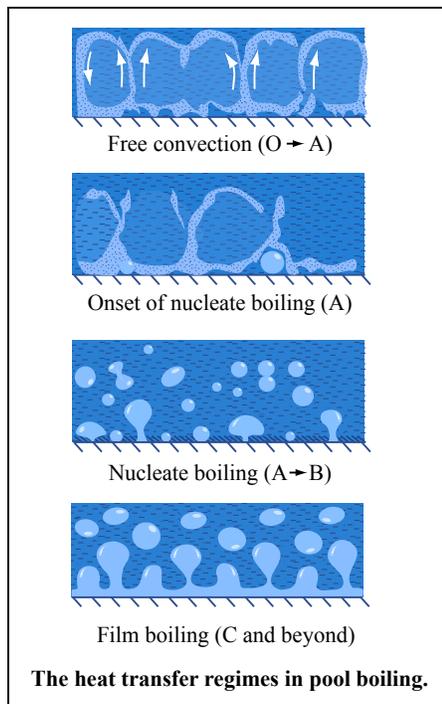
Image by MIT OpenCourseWare.

In the absence of an experiment, the various segments of the boiling curve can also be predicted by Newton's law of cooling, $q'' = h(T_w - T_{sat})$, with the heat transfer coefficient, h, given by the following empirical correlations:

O→A. Single-phase free convection from a flat plate can be predicted by the Fishenden and Saunders correlation:

$\dfrac{hL}{k_f} = 0.15 Ra_L^{1/3}$ where the Rayleigh number is $Ra_L \equiv \dfrac{g\beta(T_w - T_{sat})L^3}{\left(\dfrac{\mu_f k_f}{c_{p,f}\rho_f^2}\right)}$

In this correlation (valid for $10^7 < Ra_L < 10^{10}$), L is the representative dimension of the plate and is calculated as A/P, where A is the plate area and P is the plate perimeter, β is the thermal expansion coefficient for the liquid (a property).

Point A. The onset of nucleate boiling can be calculated as the intersection of the correlations for free convection and nucleate boiling.

A→B. The Rosenhow correlation is a popular tool to predict nucleate boiling heat transfer:

$$h = \mu_f h_{fg} \left[\frac{g(\rho_f - \rho_g)}{\sigma} \right]^{0.5} \left[\frac{c_{p,f}}{C_{s,f} h_{fg} Pr_f} \right]^3 (T_w - T_{sat})^2$$

In this correlation Pr_f is the Prandtl number for the liquid and $C_{s,f}$ is an empirical coefficient which depends on the fluid/surface combination. For example, for water on polished stainless steel the recommended value for $C_{s,f}$ is 0.013. $C_{s,f}$ is meant to capture the effect of surface (micro-cavities) on nucleate boiling.

B→C. The heat flux corresponding to DNB can be estimated by the correlation of Zuber:

$$q''_{DNB} = 0.13 \rho_g h_{fg} \left[\frac{g\sigma(\rho_f - \rho_g)}{\rho_g^2} \right]^{1/4}$$

C and beyond. In film boiling, the correlation of Berenson expresses the heat transfer coefficient as the sum of a convection term and a radiation term:

$$h = h_c + 0.75 h_{rad}$$

With:

$$h_c = 0.425 \left[\frac{g(\rho_f - \rho_g)\rho_g k_g^3 h'_{fg}}{\mu_g (T_w - T_{sat})\lambda_T} \right]^{0.25}$$

$$h_{rad} = \varepsilon \sigma_{SB} \frac{T_w^4 - T_{sat}^4}{T_w - T_{sat}}$$

$$h'_{fg} \equiv h_{fg} + 0.5 c_{p,g} (T_w - T_{sat})$$

$$\lambda_T \equiv \sqrt{\frac{\sigma}{g(\rho_f - \rho_g)}}$$

Flow boiling

The situation of interest in nuclear reactors is flow boiling. Consider a vertical channel of arbitrary cross-sectional shape (i.e. not necessarily round), flow area A, equivalent diameter D_e, uniformly heated (axially as well as circumferentially) by a heat flux q". Let T_{in} be the inlet temperature ($T_{in}<T_{sat}$), $\dot{m}$ the mass flow rate and P the operating pressure of the fluid (e.g. water) flowing in the channel. We wish to describe the various flow and heat transfer regimes present in the channel as well as the axial variation of the bulk and wall temperatures. The physical situation is shown in Fig 7.
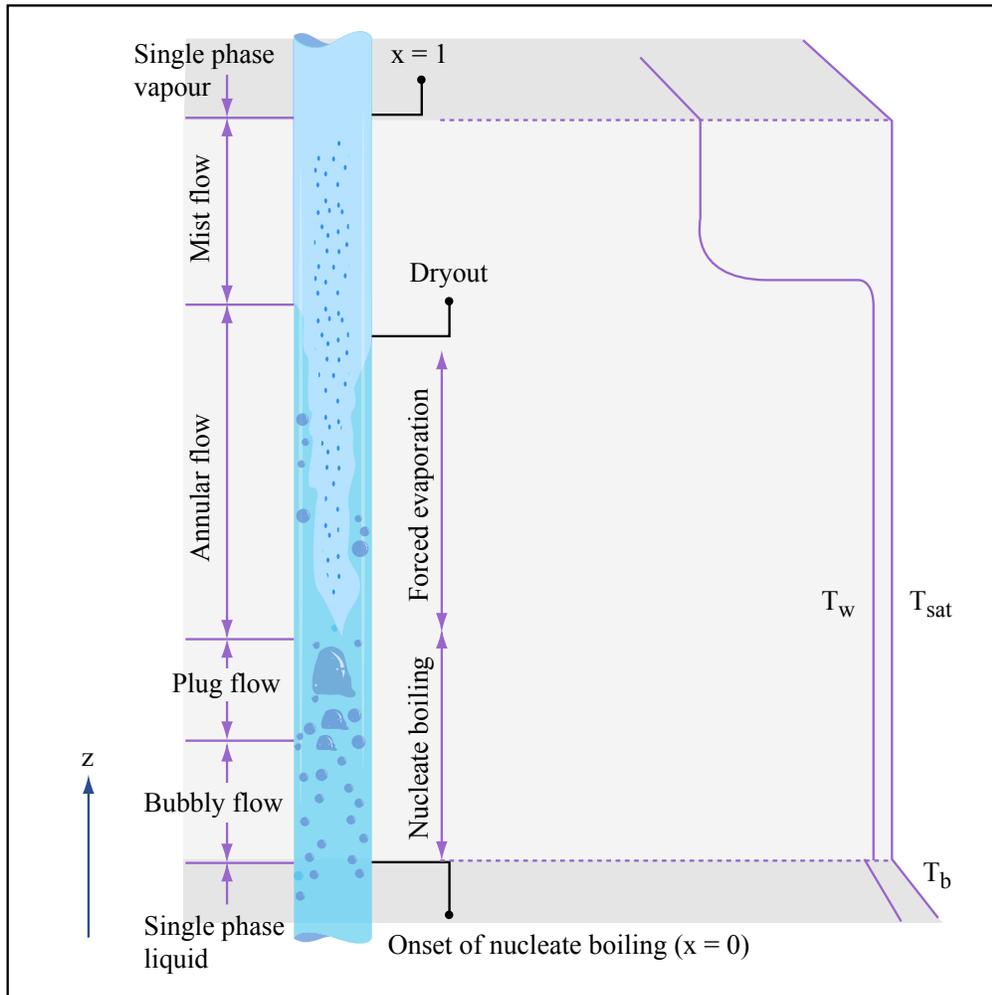
Figure 7. Heat transfer and flow regimes in a vertical heated channel. (Thermal non-equilibrium effects have been neglected in sketching the bulk temperature)

At axial locations below the onset of nucleate boiling, the flow regime is single-phase liquid. As the fluid marches up the channel, more and more steam is generated because of the heat addition. As a result, the flow regime goes from bubbly flow (for relatively low values of the flow quality) to plug (intermediate quality) and annular (high quality). Eventually, the liquid film in contact with the wall dries out. In the region beyond the point of dryout, the flow regime is mist flow and finally, when all droplets have evaporated, single-phase vapor flow.

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To calculate the bulk temperature in the channel, start from the steady-state energy equation:

$$G\frac{dh}{dz} = \frac{q''P_h}{A}$$

where P_h is the channel heated perimeter and $G=m/A$ is the mass flux. Since q'' is assumed axially constant, this equation is readily integrated:

$$h(z) = h_{in} + \frac{q''P_h}{GA}z \tag{7}$$

where h_{in} is the enthalpy corresponding to T_{in}. The liquid becomes saturated ($h=h_f$) at $z_{sat} = \frac{GA(h_f - h_{in})}{q''P_h}$. In the subcooled liquid region ($z<z_{sat}$), we also have $dh=c_{p,\ell}dT_b$ where $c_{p,\ell}$ is the liquid specific heat. Therefore, Eq 7 can be re-written as $T_b(z) = T_{in} + \frac{q''P_h}{GAc_{p,\ell}}z$. Thus, the bulk temperature distribution in the subcooled liquid region is linear (under the assumption of axially uniform heat flux).

For $z>z_{sat}$, T_b remains equal to T_{sat} until all liquid has evaporated. This location in the channel can be calculated by setting $h=h_g$ in Eq. 7 and solving for z. For z beyond this location, i.e. the single-phase vapor region, we have again a linearly increasing bulk temperature but with a different slope equal to $\frac{q''P_h}{GAc_{p,v}}$ where $c_{p,v}$ is the vapor specific heat.

The flow quality is zero in the liquid phase region. In the two-phase region, the flow quality is related to enthalpy: $h=(1-x)h_f+xh_g$. Then Eq. 7 suggests that in this region the flow quality increases (from 0 to 1) linearly with z. For locations at which $h>h_g$ (single-phase vapor), x stays equal to 1.

To calculate the wall temperature distribution in the channel, one can use Newton's law of cooling:

$$q'' = h(T_w - T_b) \qquad \Rightarrow \qquad T_w(z) = T_b(z) + q''/h$$

where the heat transfer coefficient h is given by the following empirical correlations:

Single-phase liquid region: use a single-phase heat transfer correlation appropriate for the geometry, fluid and flow regime (turbulent vs laminar) present in the channel. For example, the Dittus-Boelter correlation could be used, if one has fully-developed turbulent flow of a non-metallic fluid.

Two-phase region: for this region (from the onset of nucleate boiling to the point of dryout), we can use Klimenko's correlation. The Klimenko's correlation distinguishes between two subregions: a nucleate boiling dominated subregion (roughly corresponding to bubbly and plug flow) and a forced evaporation dominated subregion (roughly corresponding to annular flow). In the nucleate boiling subregion, heat

transfer mostly occurs through nucleation and detachment of bubbles at the wall, while in the forced evaporation subregion there are typically no bubbles, and heat transfer occurs mainly thru convection within the liquid film and evaporation at the liquid film/vapor core interface. The heat transfer coefficient is high in both subregions. The physical situation is shown in Fig 8.

(a)   (b)
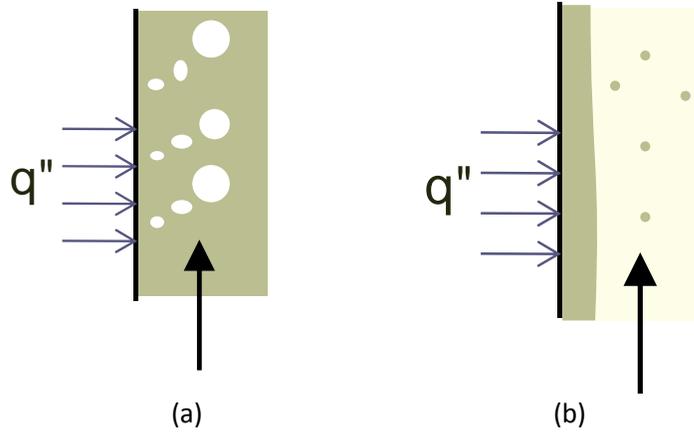
Figure 8. (a) Nucleate boiling (typical of bubbly and plug flow), (b) Forced evaporation (typical of annular flow).

Klimenko's correlation comprises the following equations:

Nucleate boiling dominated subregion:

$$\frac{hL_c}{k_f} = 4.9 \times 10^{-3} \, Pe_m^{0.6} \, \mathrm{Pr}_f^{-0.33} \left(\frac{PL_c}{\sigma} \right)^{0.54} \tag{8}$$

Forced evaporation subregion:

$$\frac{hL_c}{k_f} = 0.087 \, \mathrm{Re}_m^{0.6} \, \mathrm{Pr}_f^{1/6} \left(\frac{\rho_g}{\rho_f} \right)^{0.2} \tag{9}$$

Where in Eq. 8, P is the operating pressure, Pr_f is the liquid Prandtl number and the following definitions are used for Pe_m, Re_m and L_c:

$$Pe_m \equiv \frac{q'' L_c \rho_f c_{p,f}}{h_{fg} \rho_g k_f}$$

$$\mathrm{Re}_m \equiv G \left[1 + x \left(\frac{\rho_f}{\rho_g} - 1 \right) \right] \frac{L_c}{\mu_f}$$

$$L_c \equiv \sqrt{\frac{\sigma}{g(\rho_f - \rho_g)}}$$

Klimenko recommends use of the following criterion to determine if nucleate boiling or forced evaporation dominates at any given location in the channel:

If $N_{CB}<1.6\times10^4$, heat transfer is nucleate boiling dominated $\Rightarrow$ use Eq. 8

If $N_{CB}>1.6\times10^4$, heat transfer is forced evaporation dominated $\Rightarrow$ use Eq. 9

where $N_{CB} \equiv G\left(\dfrac{h_{fg}}{q''}\right)\left[1+x\left(\dfrac{\rho_f}{\rho_g}-1\right)\right]\left(\dfrac{\rho_g}{\rho_f}\right)^{1/3}$

Note that Klimenko's correlation is valid up to the point of dryout, but not beyond it. Dryout is an undesirable condition (a so-called boiling crisis) because the heat transfer coefficient drops drastically and thus the wall temperature rises abruptly, which can result in damage of the wall. Since PWRs and BWRs are designed not to reach the point of dryout, Klimenko's correlation is a good predictive tool to estimate the heat transfer coefficient in our applications.

For the sake of simplicity, the above discussion has ignored thermal non-equilibrium effects. In reality, boiling starts at the wall for $z<z_{sat}$, as the wall temperature reaches and exceeds T_{sat} before z_{sat}. The region of the channel between the onset of nucleate boiling and z_{sat} is called *subcooled boiling*, because the bulk liquid is still subcooled ($T_b<T_{sat}$), while the liquid and vapor in contact with the wall are slightly above saturation. Therefore in this region the two phases are not in thermal equilibrium with each other. Subcooled boiling is important in PWRs (in fact it is the only type of boiling allowed in PWR fuel assemblies) and will be covered quantitatively in recitation.

The other non-equilibrium region is the post-dryout region, where the vapor phase may become highly superheated ($T_b>>T_{sat}$) while the liquid droplets are at saturation. This region is important in BWR fuel assemblies if the point of dryout is exceeded (e.g. during an accident). Post-dryout will be covered in 22.312 and 22.313.
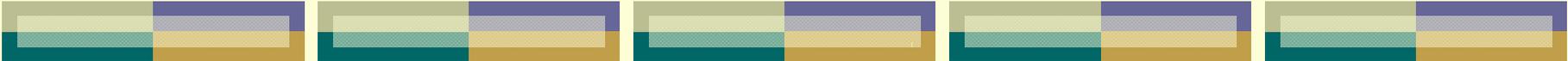
Boiling crises in LWRs

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22.06: Engineering of Nuclear Systems

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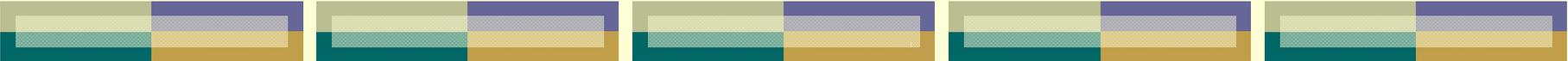
Objectives

- Identify and explain the physical bases of the thermal limits associated with the reactor coolant in PWRs and BWRs

- Explain how the thermal limits are verified in core design:
    - DNBR (for PWR)
    - CPR (for BWR)
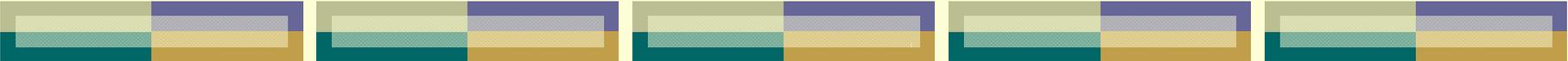
Boiling Crisis   the general idea

- There is only so much heat the coolant can remove before it experiences a thermal crisis, i.e., <u>a sudden and drastic deterioration of its ability to remove heat</u> from the fuel.

- If the thermal crisis occurs, the cladding overheats and fails, thus releasing fission products.

- To understand the thermal crisis we need to understand <u>boiling</u>.

Thermal-hydraulics of PWR Core

Channel geometry

Fuel pin

Coolant

9.5 mm

12.6 mm
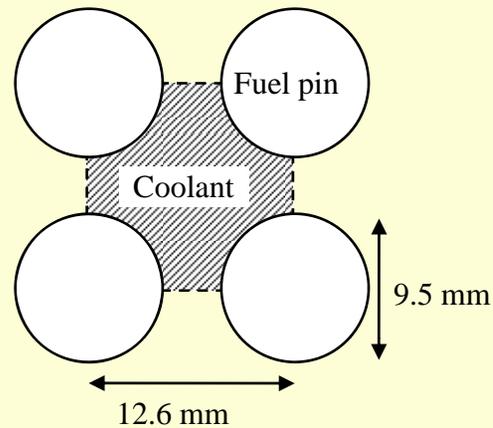
Assumed axially uniform for simplicity

q''

Average channel
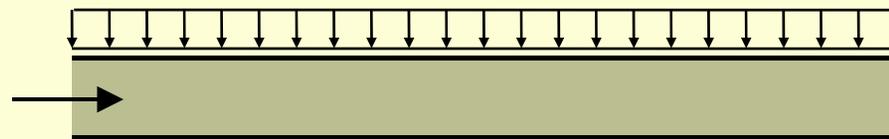
No boiling in the average channel

Temperature

T_{sat}

T_{co}

T_b

z

4
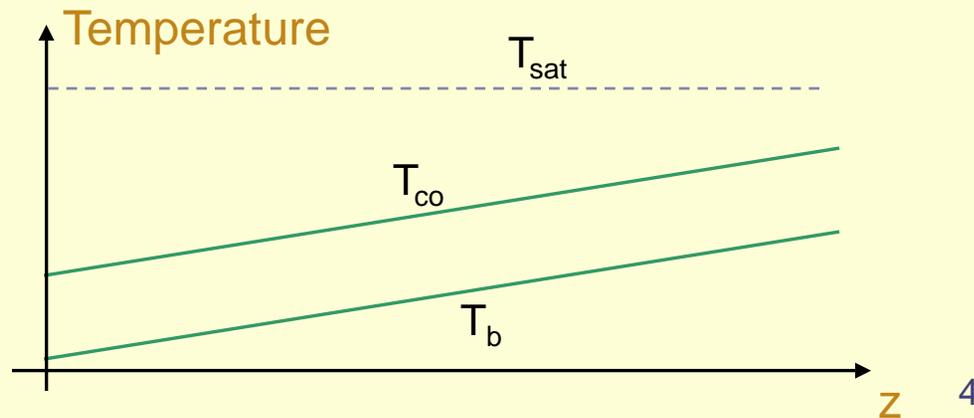
Thermal-hydraulics of PWR Core (2)

$$q''$$
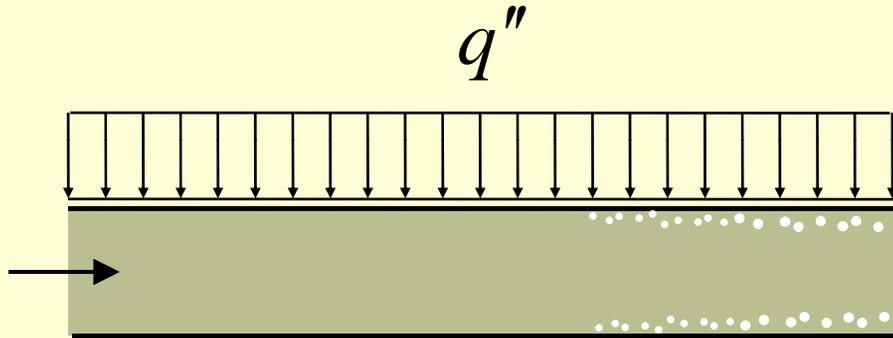
Hot channel

Subcooled boiling in the hot channel

Temperature

T_{sat}

T_{co}

T_b

z
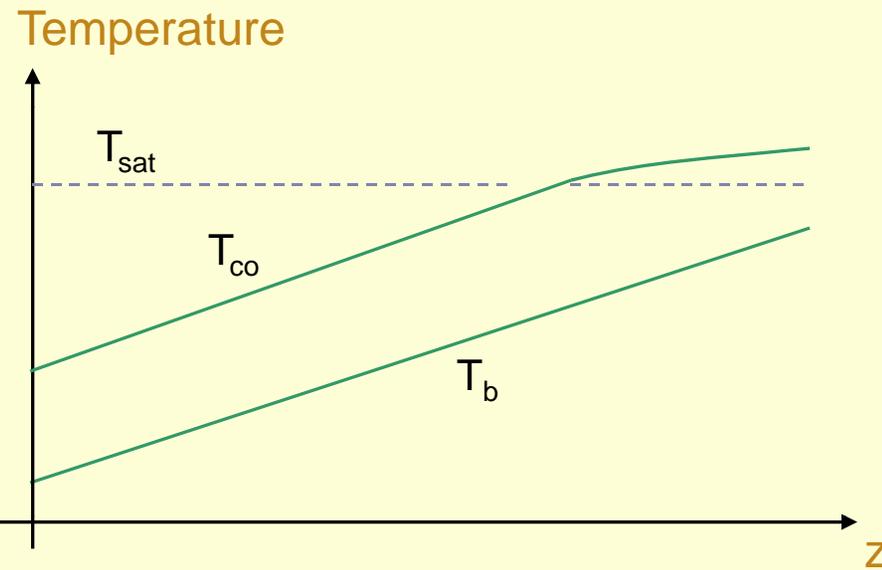
Thermal-hydraulics of PWR Core (3)

$$q''$$

DNB (thermal crisis)
in hot channel

DNB

Temperature

T_{sat}
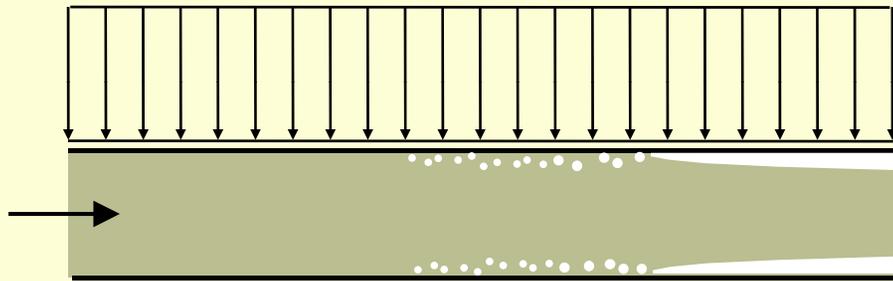
Clad temperature
skyrockets when
DNB occurs

T_{co}

T_b

z
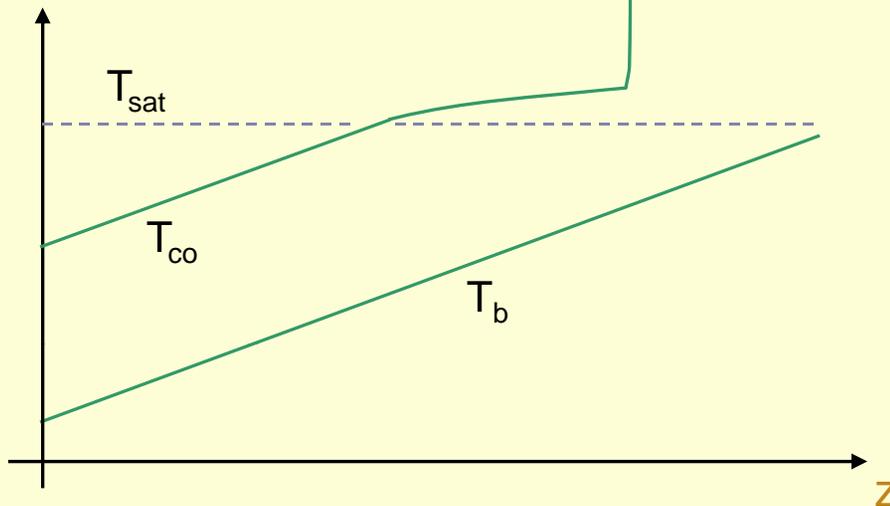
Thermal-hydraulics of PWR Core (4)

Heat flux to cause DNB depends on T_b, G and P $\Rightarrow$ $q''_{DNB}(T_b, G, P)$

Heat flux

q''_{DNB}

q''

z

Heat flux

q''_{DNB}

q''

z
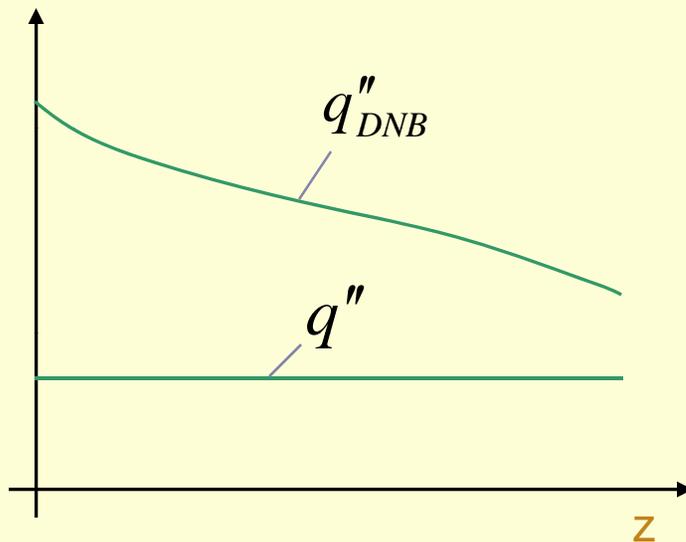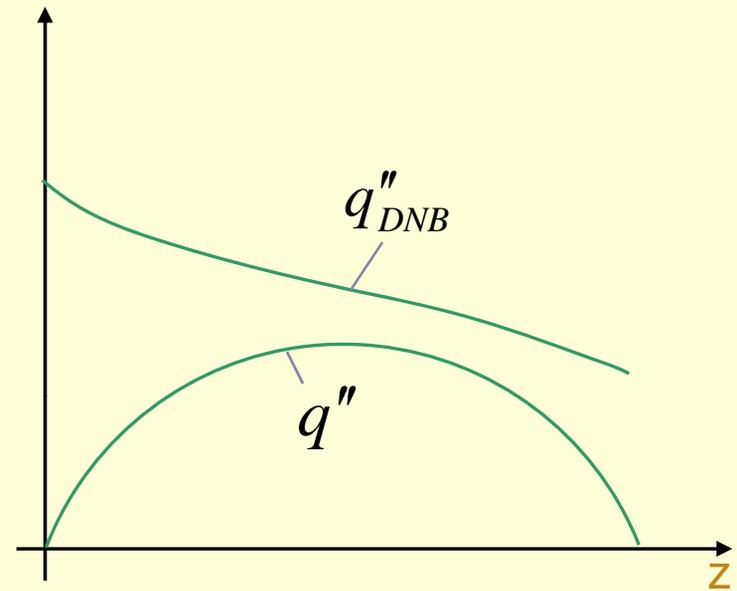
$$DNBR(z) \equiv \frac{q''_{DNB}}{q''}$$

$$MDNBR \equiv \text{min. DNBR} > 1.3 \text{ in the US}$$

Thermal-hydraulics of PWR Core (5)

Correlation (Tong 68) to calculate q''_{DNB}

$$q''_{DNB} = K_{Tong} \frac{G^{0.4} \mu_f^{0.6} h_{fg}}{D_e^{0.6}}$$

Where:

$$K_{Tong} = [1.76 - 7.433 x_e + 12.222 x_e^2]$$

$$x_e = -\frac{c_{p,\ell}(T_{sat} - T_b)}{h_{fg}} \quad \text{< 0 in a PWR}$$

Thermal-hydraulics of BWR Core

Channel geometry

Fuel pin

Coolant

12.3 mm

16.2 mm

q''
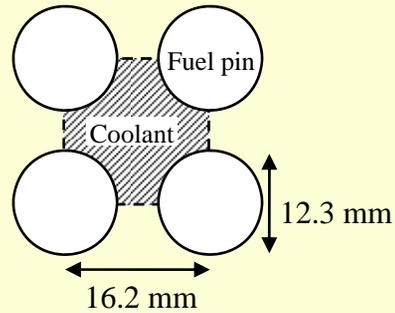
Average or hot channel
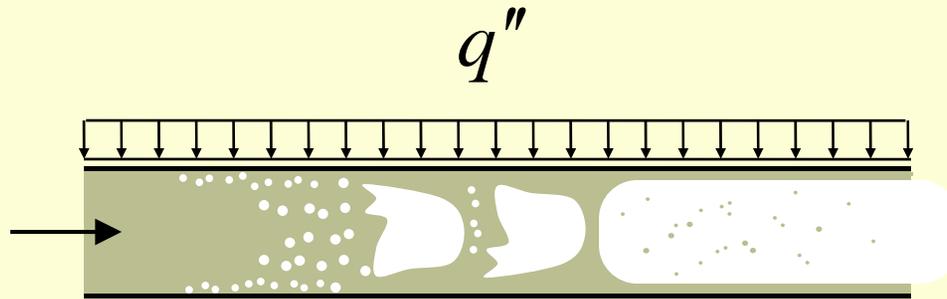
Saturated boiling in most of the channel

Temperature

T_{co}

T_{sat}

T_b

z

9
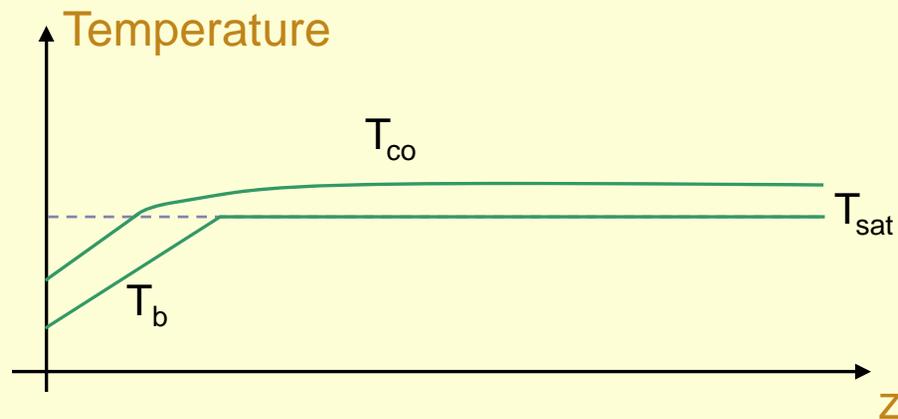
Thermal-hydraulics of BWR Core (2)

q''

Dryout (thermal crisis) in hot channel

Dryout of the liquid film in annular flow
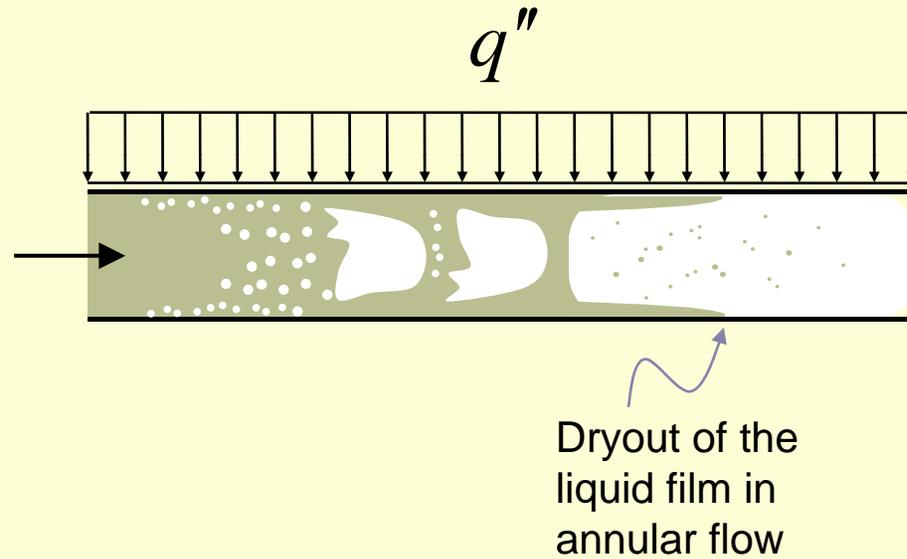
Clad temperature increases sharply when dryout occurs

Temperature

T_{co}

T_{sat}

T_b

z
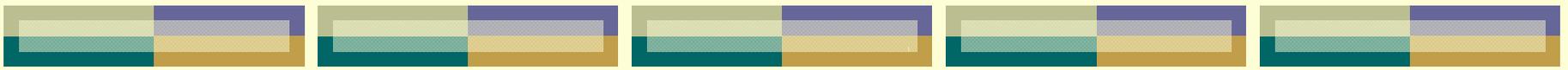
Thermal-hydraulics of BWR Core (4)

Correlation (CISE-4) to calculate x_{cr}

$$x_{cr} = \frac{P_h}{P_w} a \frac{z}{z+b}$$

Where:

$$a = (1 - P/P_{cr})/(G/1000)^{1/3}$$

$$b = 0.199(P_{cr}/P - 1)^{0.4} G D_e^{1.4}$$

Note:

- P_{cr} is the thermodynamic critical pressure of water (22.1 MPa)
- The units for G and D_e must be kg/m²-s and m, respectively

Boiling Crisis - Summary

Reactor	PWR	BWR
Thermal crisis type	Departure from Nucleate Boiling (DNB)	Dryout
Phenomenon		
Design limit	MDNBR > 1.3	CPR > 1.2
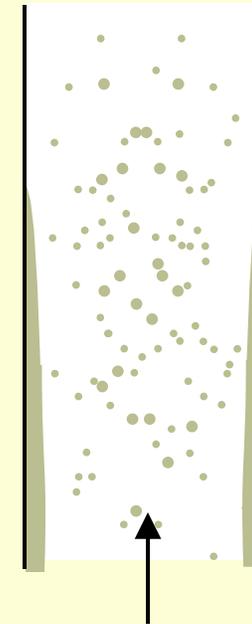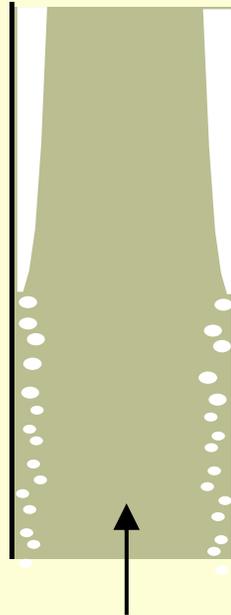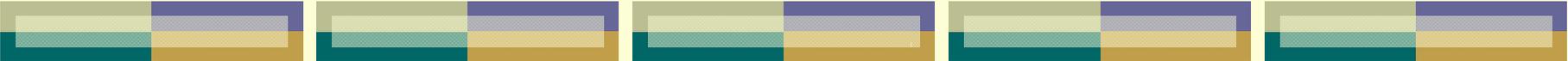
Boiling demos

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22.06: Engineering of Nuclear Systems

Boiling – a simple experiment

Pool boiling of water at 1 atm. The current delivered to the wire is controlled by the power supply.

Power supply

Water

Wire

L

I

Electric resistance of the wire

$$\dot{q} = I^2 R$$

$$q''_w = \frac{\dot{q}}{2\pi r_w L}$$

Heat flux at the wire surface

Wire radius
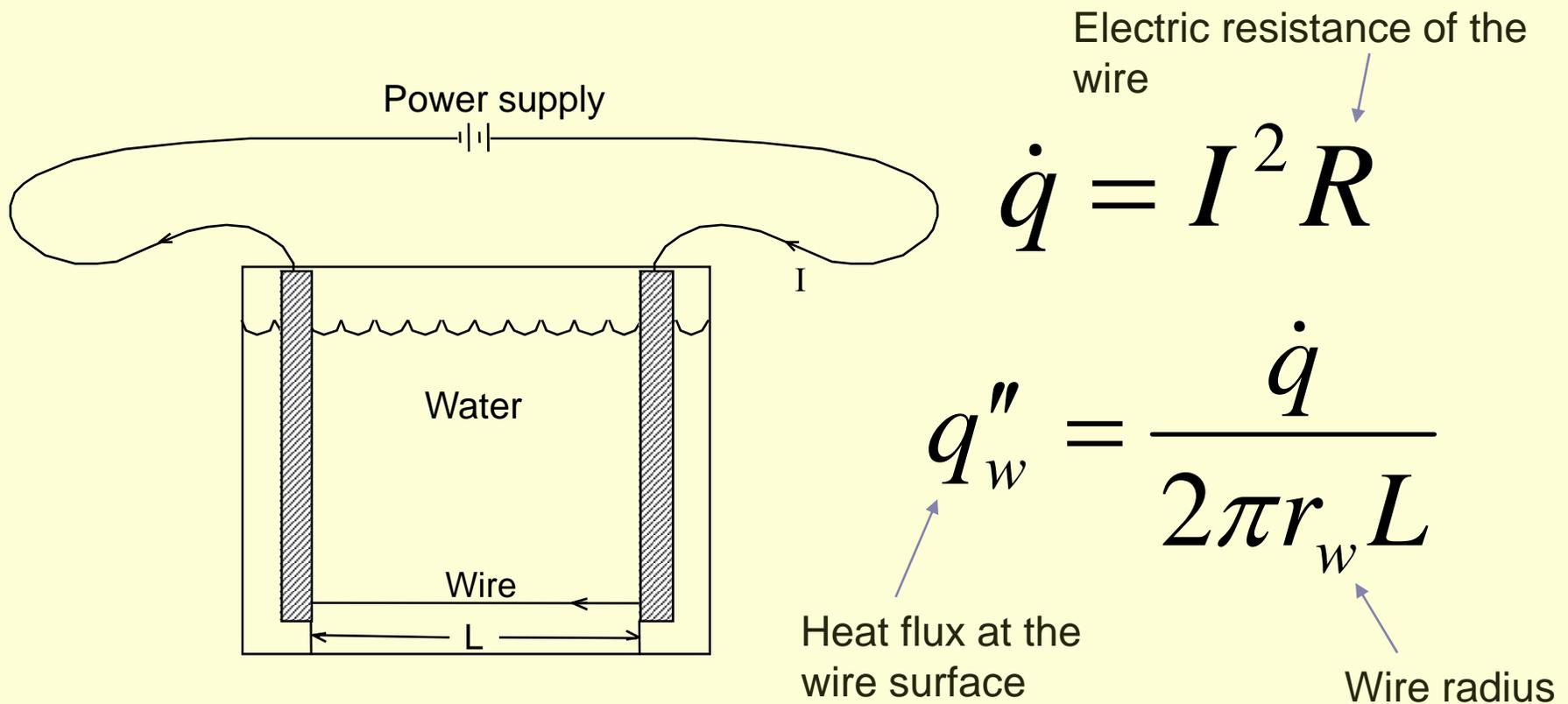
Boiling – a simple experiment (2)

Plot of wire temperature vs. heat flux (the *boiling curve*)
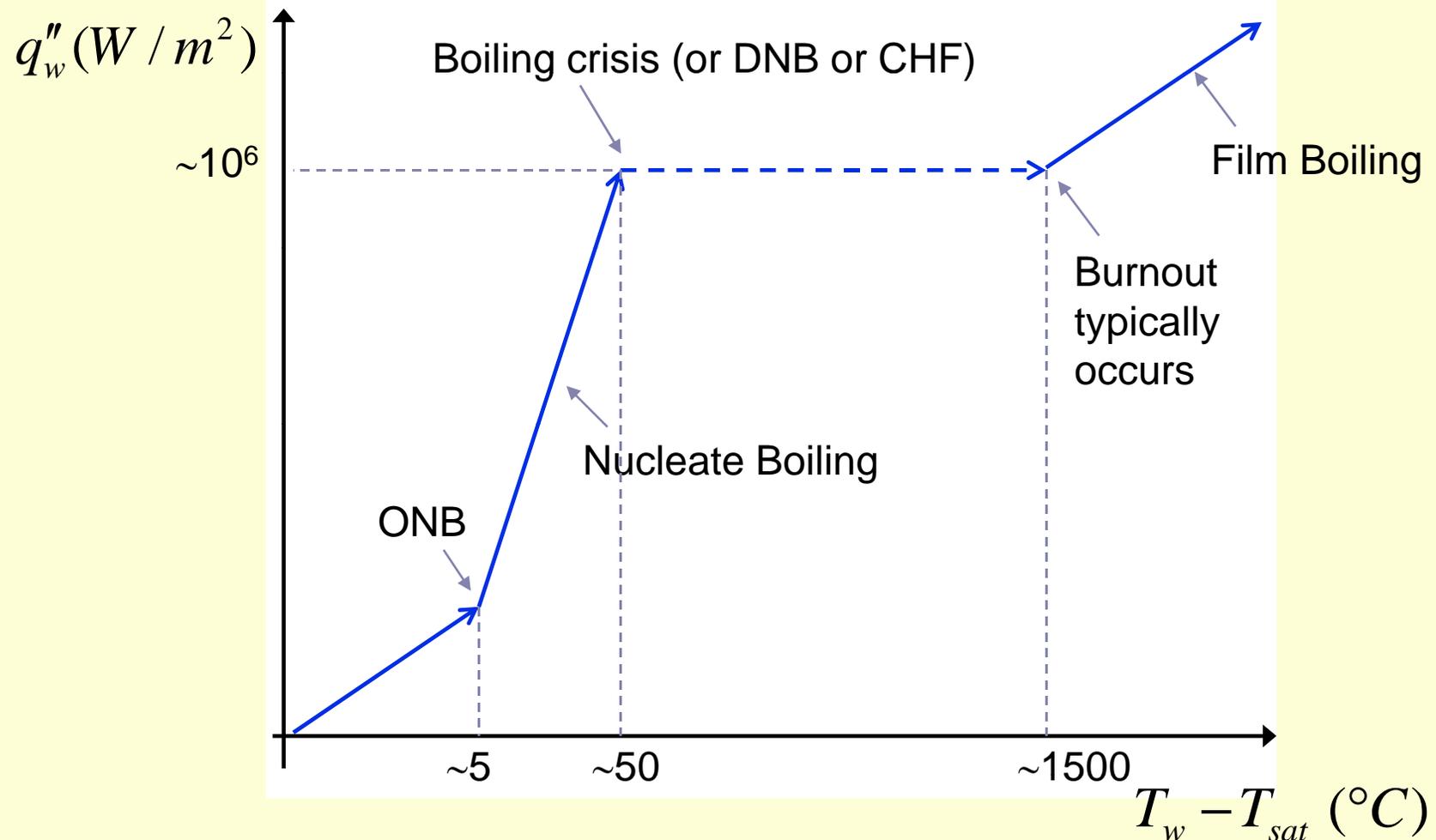
Main message: Nucleate boiling = good, DNB = bad

Boiling – a simple experiment (3)

Let's run the demo!

Flow Regimes

The situation of interest in nuclear reactors is flow boiling (vs pool boiling). There are three basic flow patterns:
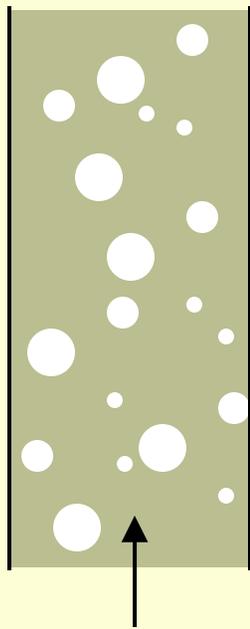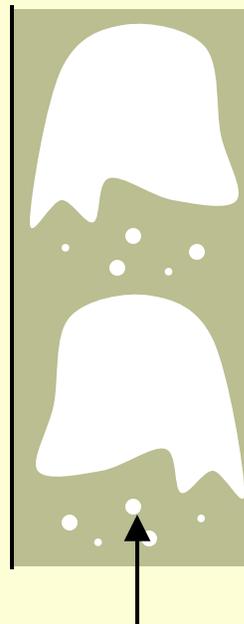
Bubbly flow   **Plug flow**   **Annular flow**
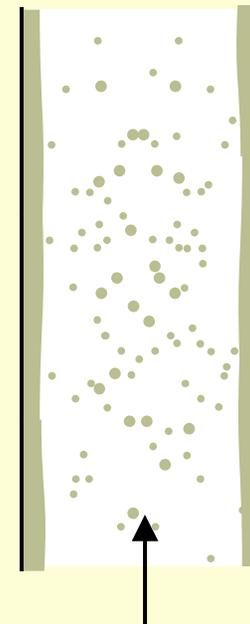
Increasing amount of steam (or steam quality, or void fraction)
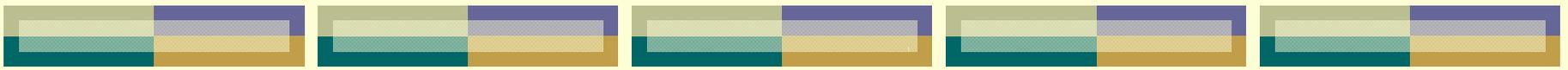
Flow Regimes (2)

Two important definitions that measure the amount of steam in a channel:

$$\text{Steam Quality } (x) \equiv \frac{\text{mass flow rate of steam}}{\text{total mass flow rate}}$$

$$\text{Void Fraction } (\alpha) \equiv \frac{\text{volume of steam}}{\text{total volume}}$$

Both x and α are by definition between 0 (all liquid) and 1 (all steam)

Flow Regimes (3) - Demo

Injection of air into column of water.

Water column

Injection valve

Air compressor
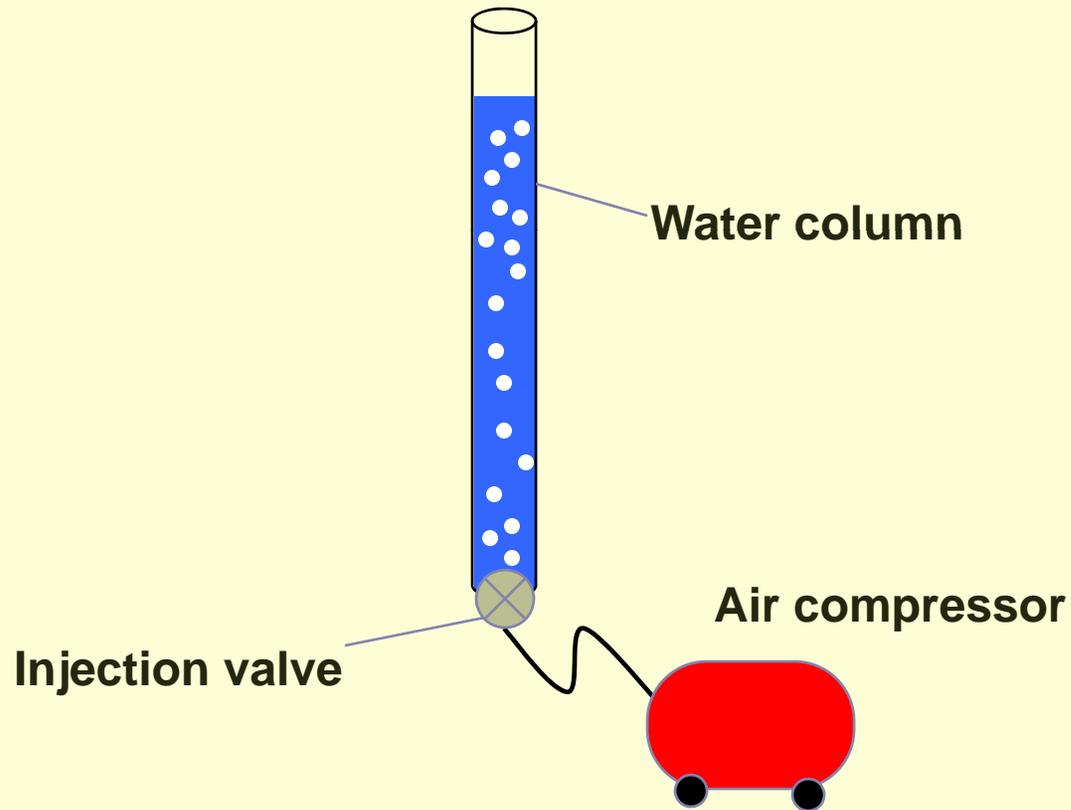
Watch how the flow pattern changes as air flow is increased

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