# Fundamentals of Nuclear Engineering

Module 12: Two Phase Heat Transfer and Fluid Flow

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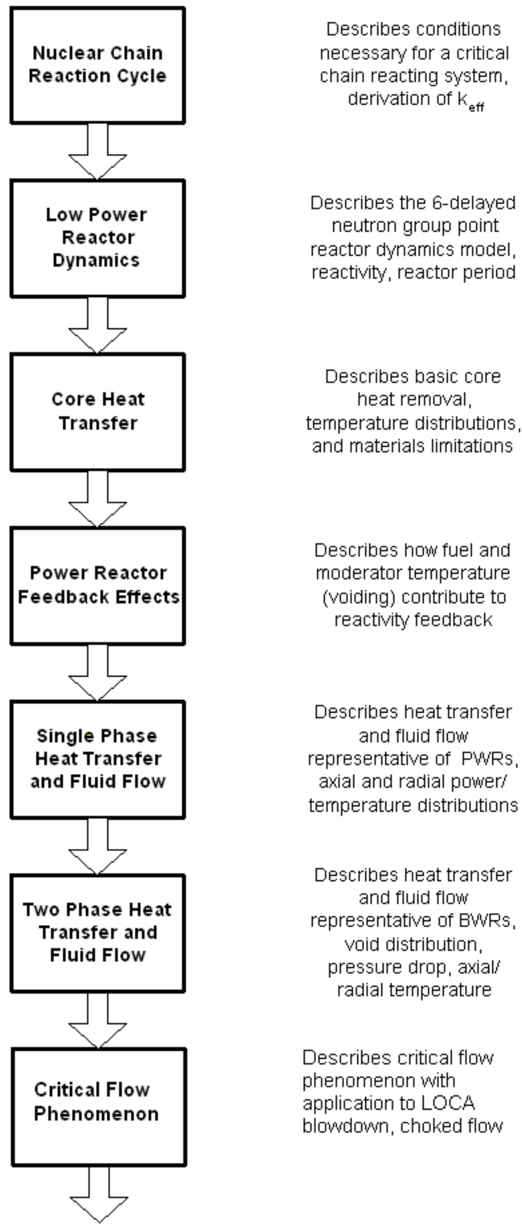

Nuclear Chain Reaction Cycle
Describes conditions necessary for a critical chain reacting system, derivation of $k_{eff}$
Low Power Reactor Dynamics
Describes the 6-delayed neutron group point reactor dynamics model, reactivity, reactor period
Core Heat Transfer
Describes basic core heat removal, temperature distributions, and materials limitations
Power Reactor Feedback Effects
Describes how fuel and moderator temperature (voiding) contribute to reactivity feedback
Single Phase Heat Transfer and Fluid Flow
Describes heat transfer and fluid flow representative of PWRs, axial and radial power/ temperature distributions
Two Phase Heat Transfer and Fluid Flow
Describes heat transfer and fluid flow representative of BWRs, void distribution, pressure drop, axial/ radial temperature
Critical Flow Phenomenon
Describes critical flow phenomenon with application to LOCA blowdown, choked flow

# *Objectives:*

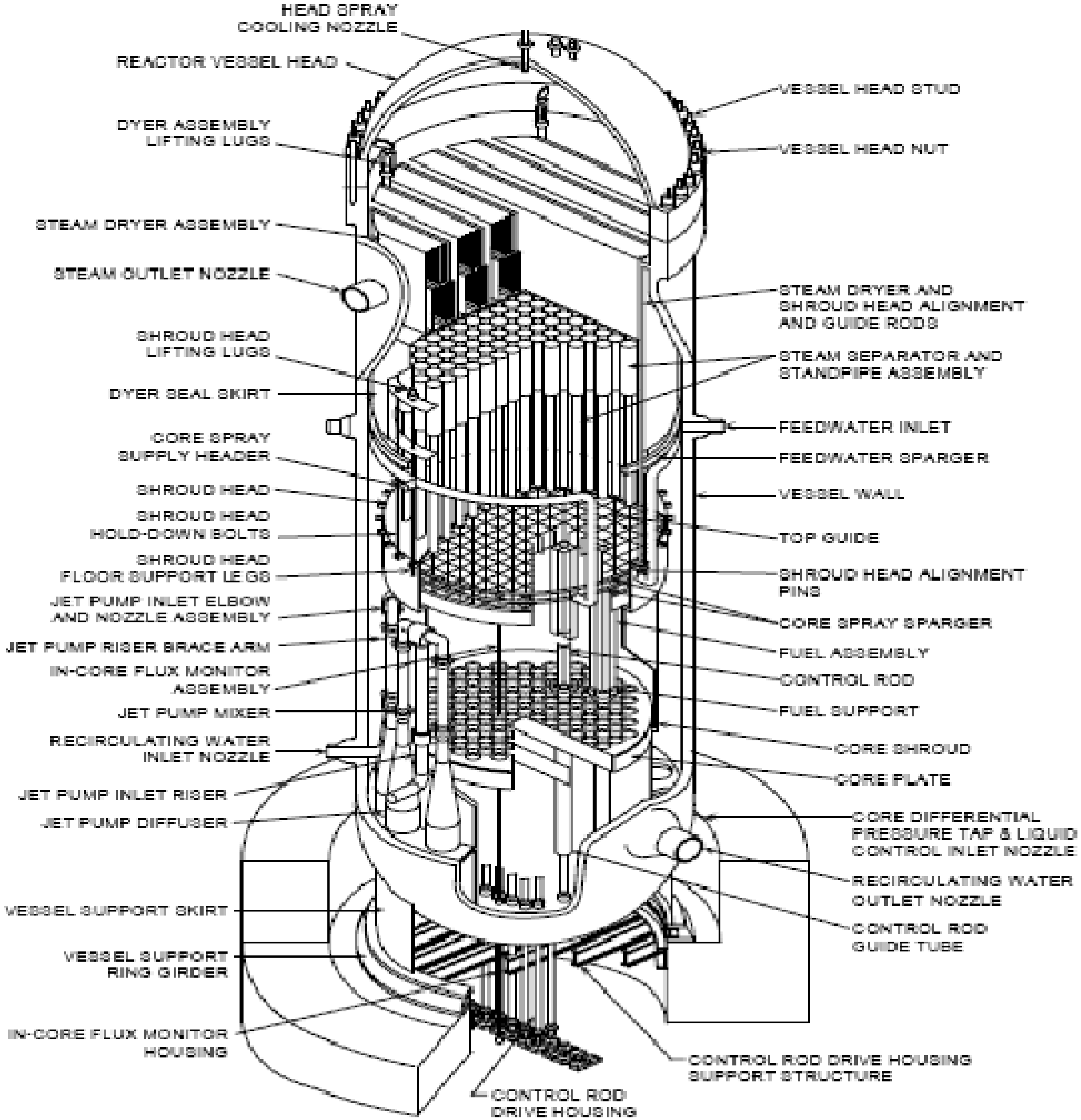
Previous Lectures described heat transfer and reactivity in single phase systems. This lecture:

1. Describe two-phase Systems
2. Describe important thermal-hydraulic concepts important to a BWR
3. Describe two-phase flow equations
4. Describe two phase heat transfer rates from fuel to coolant and Boiling Transition
5. Describe steady state core temperature profiles
6. Describe fluid flow, and pressure drops in two phase systems
7. Describe behavior of system during accident

# *1. Two Phase Fluid Systems*

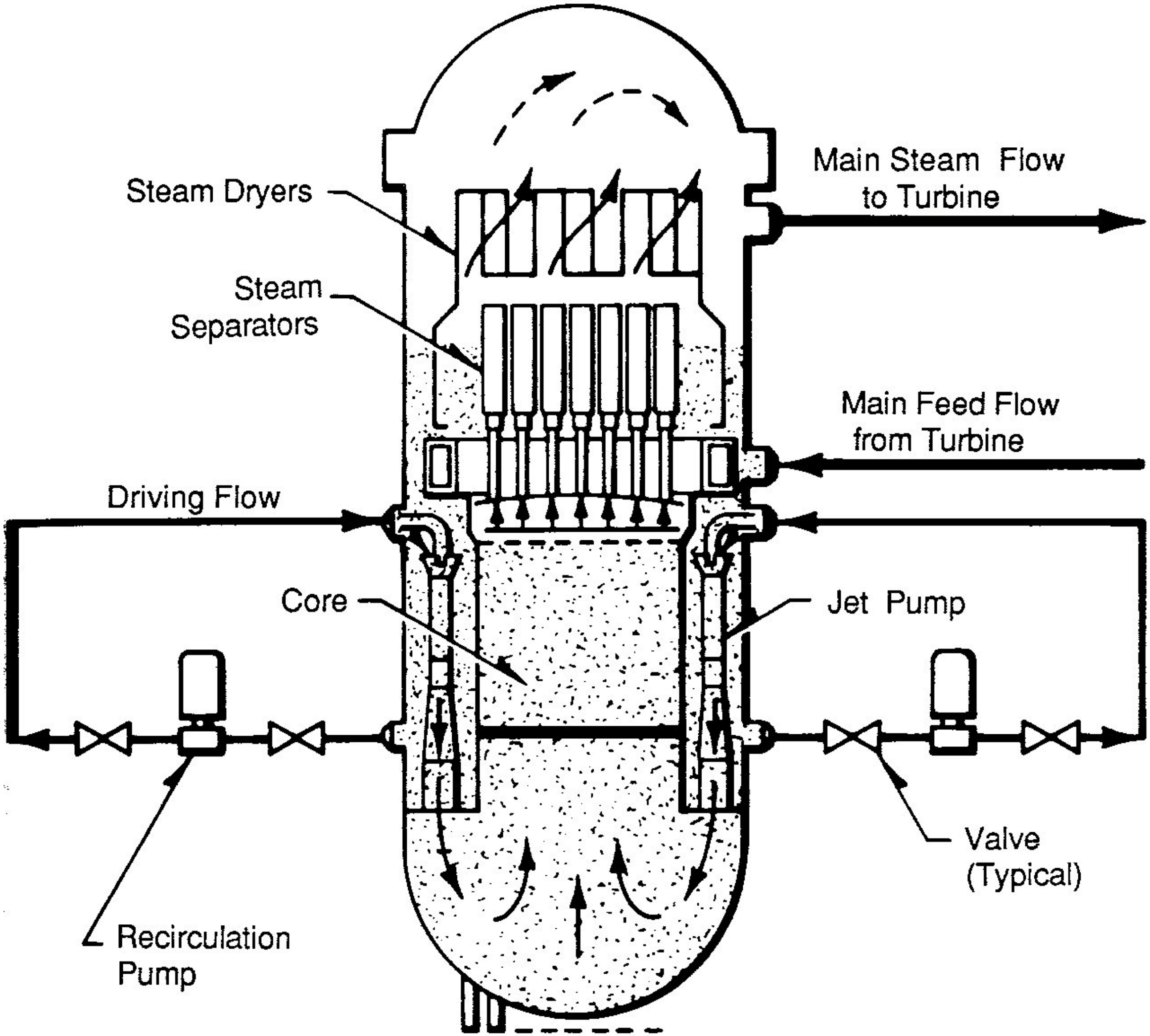
# Two-Phase Flow Systems in Nuclear Engineering

- Heat Exchangers
- Piping Systems in Balance of Plant and Reheat of Feedwater
- Steam Generator
- BWR Reactor

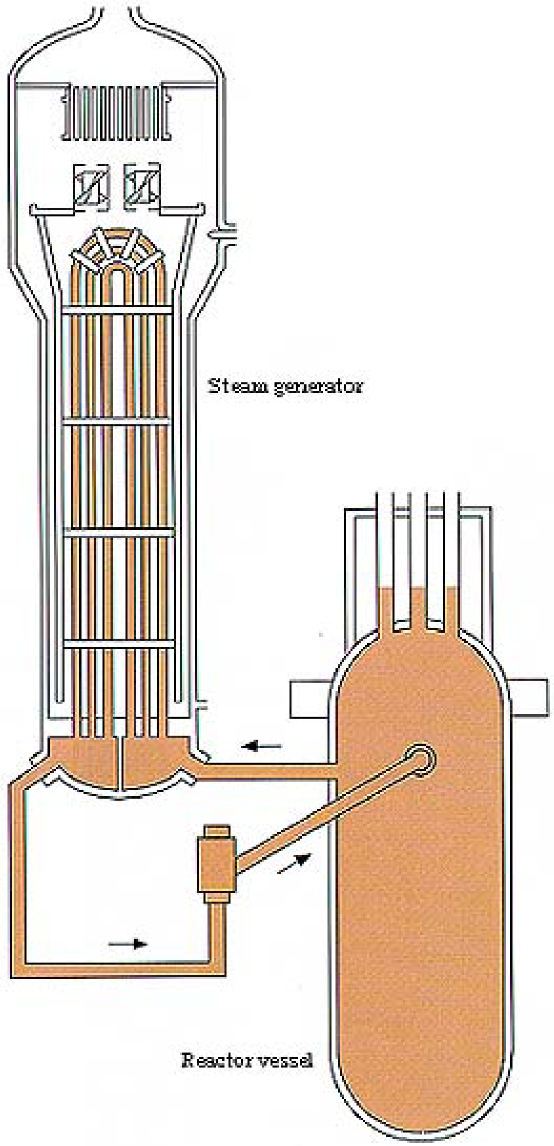


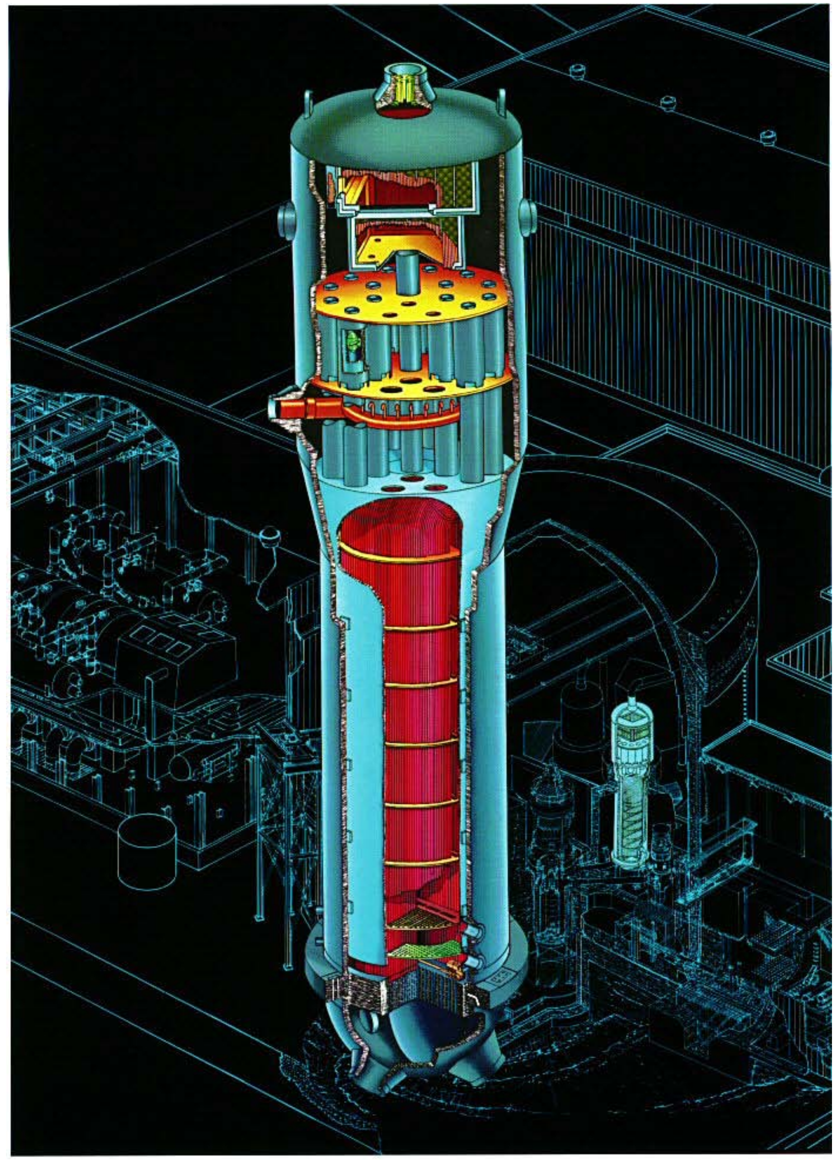

Reactor Vessel Cutaway

# BWR Flow Paths



# PWR Steam Generator



Westinghouse STEAM GENERATOR

# 2. Important Thermal Hydraulic Stages for BWR

- Start-up and Steady State Operation
- Operational Transients
- Loss of Coolant Accidents

Each of these stages require many different analytical techniques for predicting heat transfer and fluid flow in a BWR.

# Start-up and Steady State Operation

- Core Thermal Power
- Power-Flow Map
- Control Rod Positioning
- Feedwater Temperature Control – Amount of Subcooling – more power
- Core Response to Recirculation Flow Changes - BWR

# Core Thermal Power

- Core thermal power by energy balance and use of instrumentation.
- Inlet Subcooling
- Quality in channel
- Void Fraction in the Fuel Channel
- Fluid flow and pressure drop
- Core orificing
- Core bypass flow
- Enrichment distribution

# Power-Flow Map

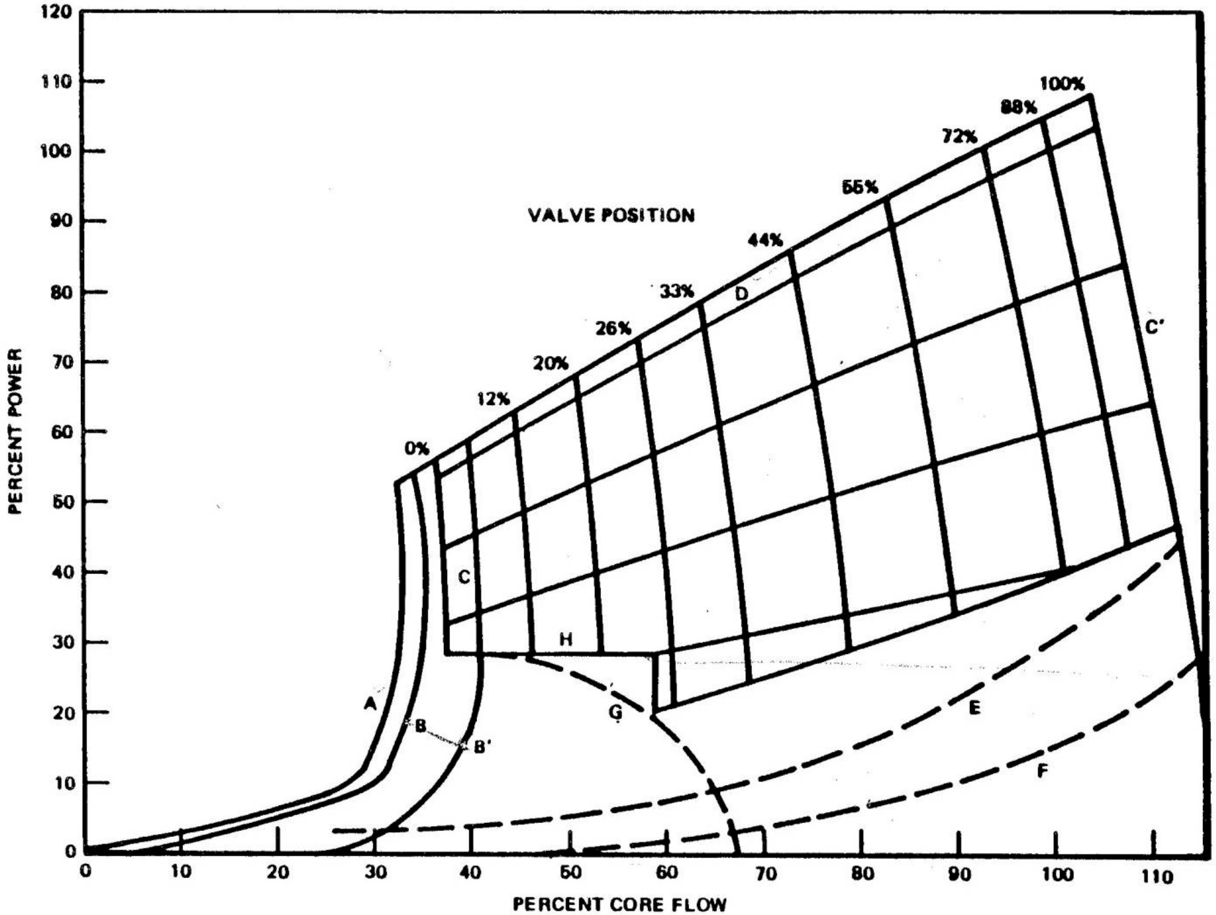



*Figure 4-9. Typical Power-Flow Map for BWR/5 and BWR/6*

# Control Rod Positioning

- Shallow Control Rods
- Deep Control Rods
- Intermediate Control Rod
- Shallow-deep Combination

# Operational Transients

- Based on FSAR Chapter 15 requirements for initiating events
- Nuclear reactor system pressure increases by reactor trip, MSIV isolation, etc.
- Positive reactivity insertion by moderator temperature increase as in loss of feedwater heating.

# Types of FSAR Chapter 15 Transients

- increase in heat removal from the primary system
- decrease in heat removal by the secondary system
- decrease in reactor coolant system (RCS) flow rate
- reactivity and power distribution anomalies
- increase in reactor coolant inventory
- decrease in reactor coolant inventory
- anticipated transients without scram (ATWSs)

# Loss of Coolant Accidents

- Large breaks
- Small breaks
- Intermediate breaks

# *3. Two Phase Flow Equations*

# General Terminologies

- Two-phase flow
  - Simultaneous flow of any two phases (liquid-gas/vapor, solid-gas, liquid-solid) of a single substance
  - Examples: reactor fuel channels, steam generators, kettle on a hot stove
  - Also referred to as “Single-component two-phase flow”

- Two-component flow
  - Simultaneous flow of liquid and gas of two substances
  - Examples: oil-gas pipelines, beer, soft drink, steam-water-air flow at discharge of safety valve.
  - Also referred to as “Two-component, two-phase flow”

# *Terminology Unique to Two Phase*

- In static (non-flowing system) steam quality: $\chi$ is defined:

$$\chi = (\text{mass of steam}) / (\text{total mass of steam + liquid})$$

- In static (non-flowing system) void fraction: $\alpha$ is defined:

$$\alpha = (\text{volume of steam in mixture}) / (\text{total volume of steam + liquid})$$

- Void fraction can be expressed in terms of steam quality and specific volumes (from *Steam Tables*) as follows:

$$\alpha = \chi v_g / (v_g + \chi v_{fg}) = 1 / \{1 + [(1 - \chi)/ \chi]\, v_f / v_g\}$$

- Where: $v_g$ is specific volume of steam in $ft^3/lb\text{-}m$

  $v_f$ is specific volume of liquid in $ft^3/lb\text{-}m$

  $v_{fg} = v_g - v_f$ is difference in specific volumes

Good source for fluid properties: http://webbook.nist.gov/chemistry/fluid/

# Void Fraction

- Ratio of Vapor flow area to total flow area
- Depends strongly on pressure, mass flux, and quality
- Applied to calculate the acceleration pressure drop in steady-state homogeneous code
- Large number of correlations proposed
- Solved from conservation equations in two-fluid reactor safety codes

# *Steam Rises Faster in Channel Than Liquid*

- Because of lower density (buoyancy) steam will rise up vertical channel faster than surrounding liquid

- Slip ratio: $S$, is the ratio of steam velocity to liquid velocity

$$S = V_g / V_f$$

where: $V_g$ is steam velocity in ft. / sec.
$V_f$ is liquid velocity in ft. / sec.

- Slip ratio modifies static definitions of $\alpha$ (void fraction) and $\chi$ (steam quality) in flowing two phase system

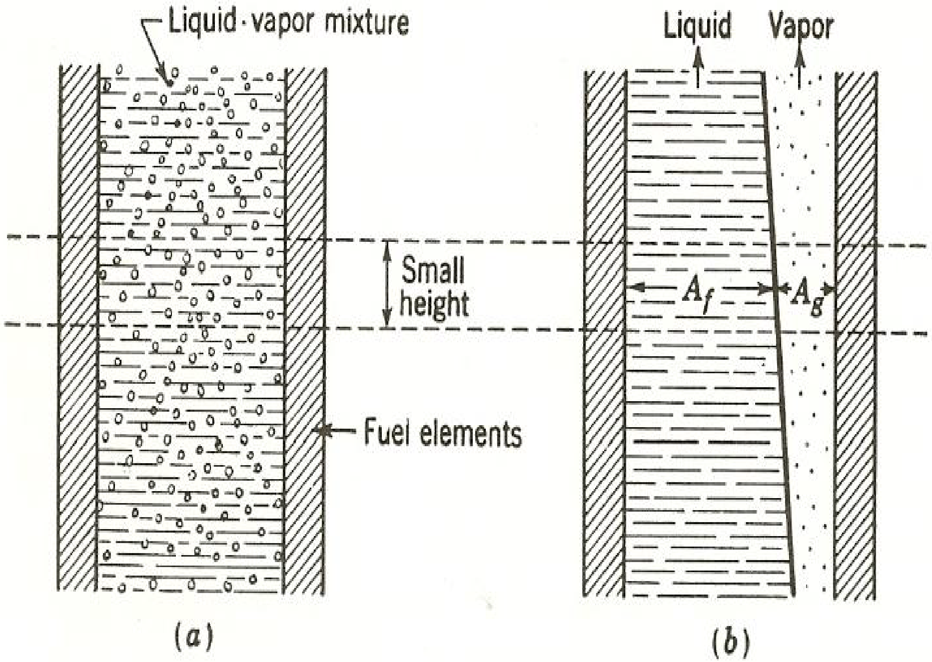
# *Relative Velocities Are Different*

- If total mass flow rate is:
  $W$ *(lb-m/sec)*
- Steam flow rate is: $\chi W$
- Liquid flow rate is: $(1-\chi)W$
- Phase volumetric velocity of steam is:

  $$V_g = v_g W \chi / A_g$$

  -where: $A_g$ is relative cross sectional area of steam in two phase column
- Phase volumetric velocity of liquid is:

  $$V_f = v_f W(1-\chi) / A_f$$




Two-phase flow in a heated channel.

# Definition of Slip in Terms of Steam Quality and Void Fraction

- Slip: $S = V_g / V_f$
- Slip defined in terms of steam quality by combining:

$$V_g = v_g W \chi / A_g$$

$$V_f = v_f W(1 - \chi) / A_f$$

- This yields:

$$S = (\chi / 1 - \chi)(A_f / A_g)(v_g / v_f)$$

- Noting that in small slice of column, ratio of steam to total mixture is: $\alpha = A_g / A_g + A_f$ , rearranging this:

$$(A_f / A_g) = (1 - \alpha) / \alpha$$

- Slip can then be expressed:

$$S = (\chi / 1 - \chi)[(1 - \alpha) / \alpha](v_g / v_f)$$

# *Definition of Void Fraction in Terms of Steam Quality and Slip*

- Slip equation can be rearranged to define void fraction: $\alpha$ in terms of steam quality and Slip:

$$\alpha = \frac{1}{1+[\frac{(1-\chi)}{\chi}]\left(\frac{v_f}{v_g}\right)S}$$

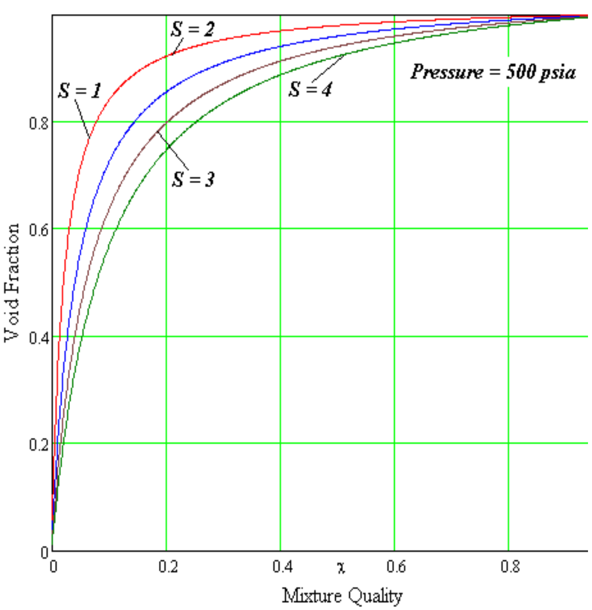
- When $S = 1$: steam and liquid move at exact same speed

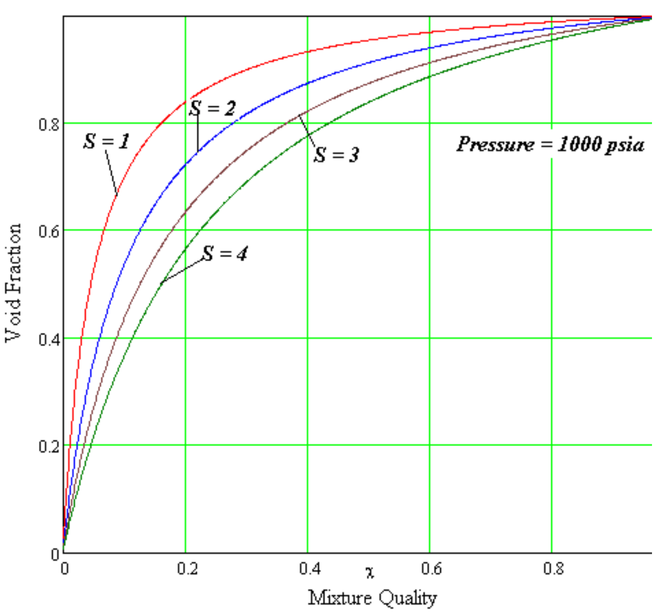
*Effect of slip:*

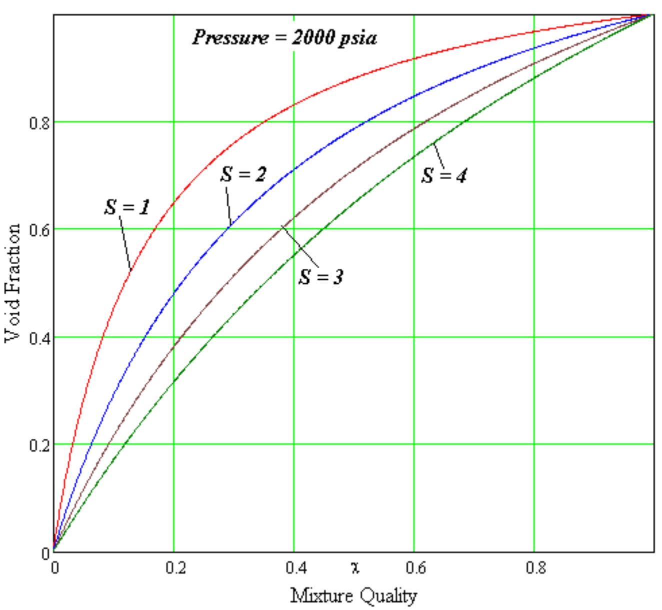
- *Slip decreases void fraction* $\alpha(\chi)$ below that which exists in situation of no slip between steam and liquid

# *Void Fraction Sensitivities*:

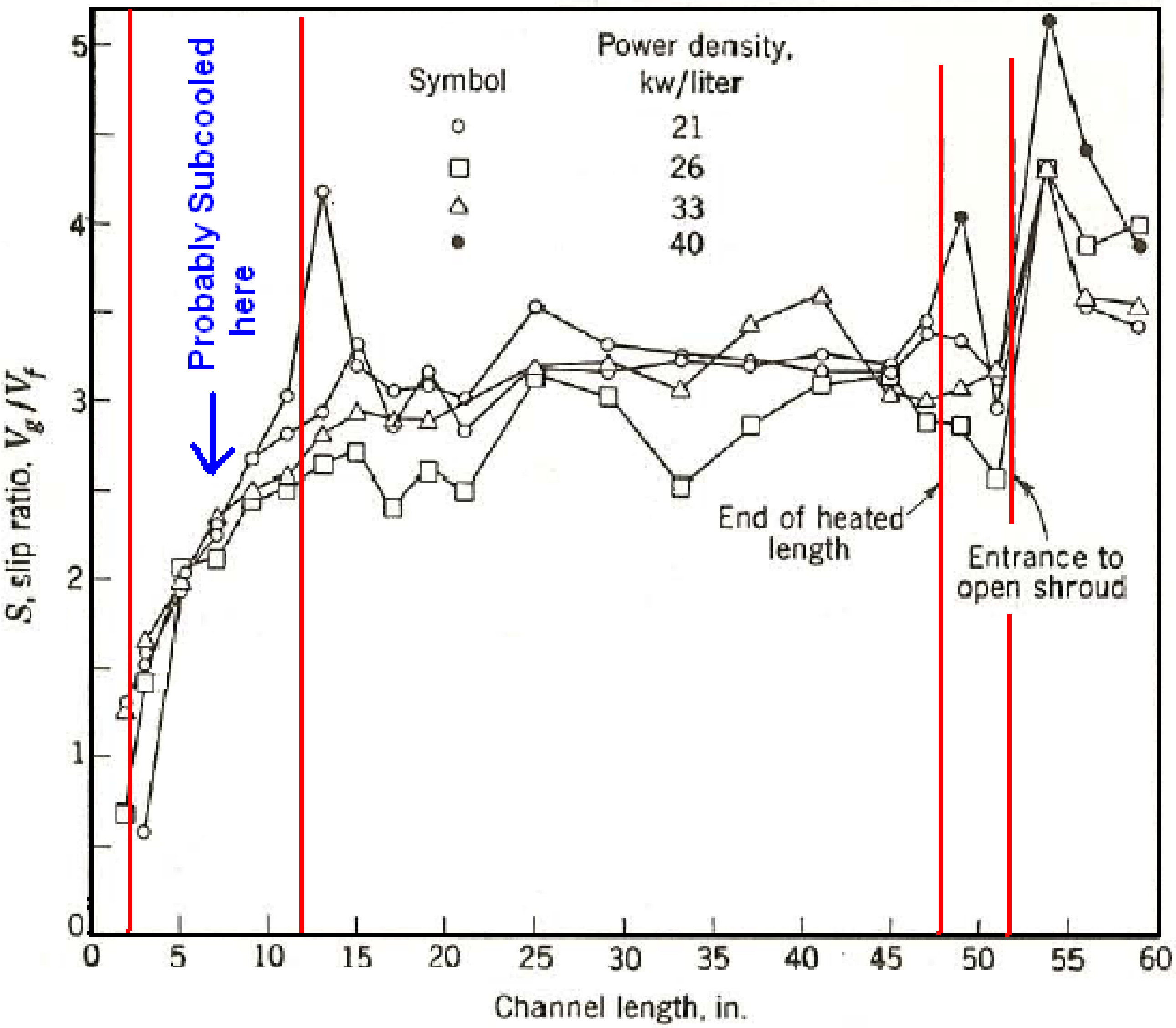
- Sensitivity of Void fraction $\alpha$ to Mixture Quality and Slip can be seen by computing $\alpha(\chi)$ for a spectrum of pressure and assumed Slip values
- $S = 1$ implies homogeneous steam/water flow (moving together)



# Example Slip Ratio for BWR Fuel Channel




Typical plots of slip ratio versus channel length at 114.9 psia.

# *What Does Slip Mean to the Core Fluid Flow?*

*Low S value ($S = 1$) implies:*

- Voids and water travel about same speed

*Higher S value ($S = 2,3$) implies:*

- Steam carries out higher enthalpy thus heat is removed faster
- Voids swept out of channel faster which is benefit for neutron economy
- Predicting Slip from first principles is not easy
- Designers rely upon tests and scaling-up from previous

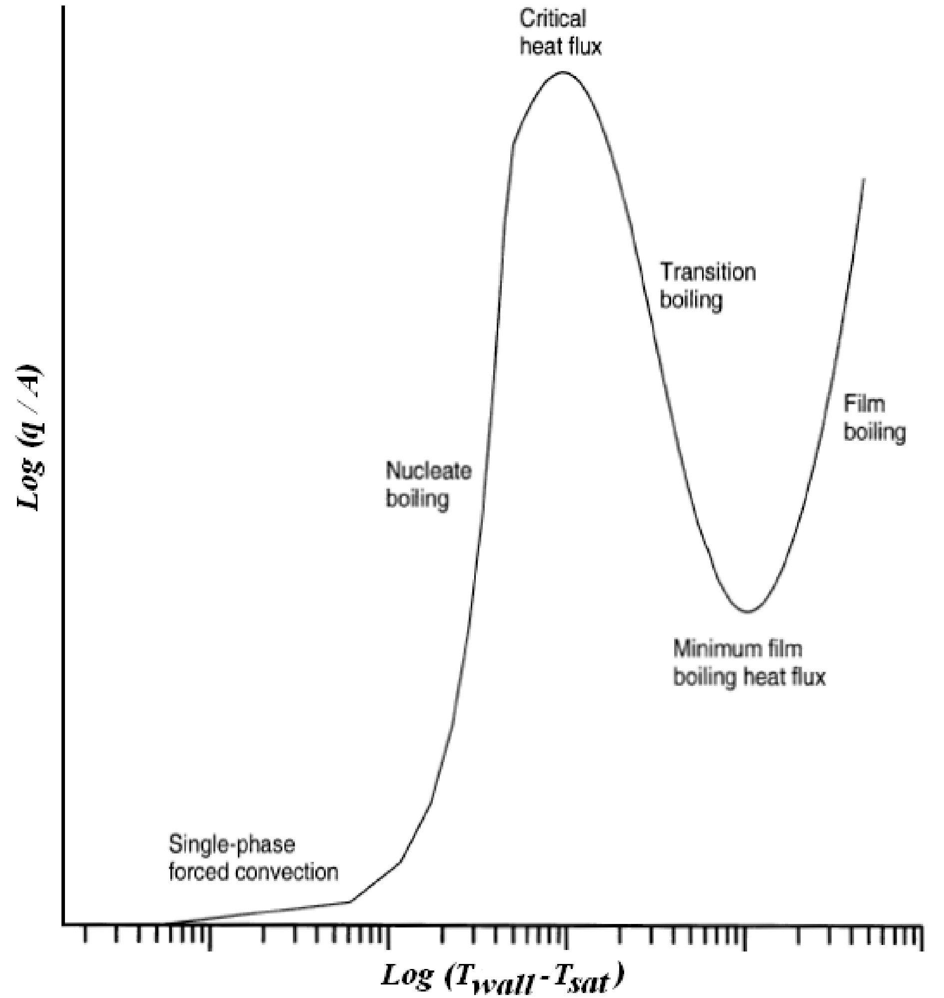
# *4. Two Phase Heat Transfer Rates from Fuel to Coolant and Boiling Transition*

# *Two Phase Heat Transfer Regimes*

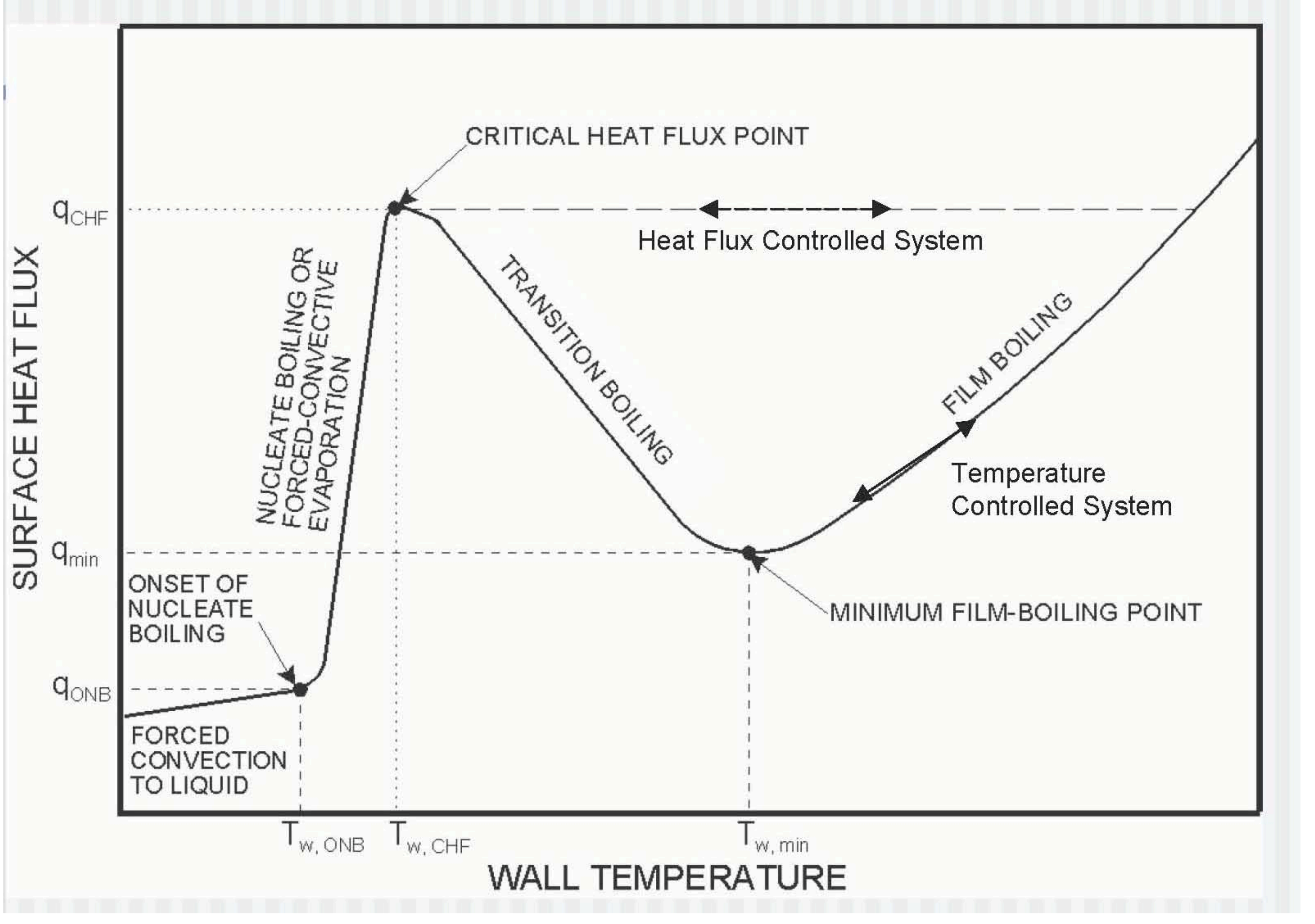
# Six Distinctive Boiling Regions:

- Single phase, forced convection
- Nucleate boiling
- Critical heat flux
- Transition boiling
- Minimum film boiling
- Film boiling

Methods and experimental correlations exist to describe each region



# Boiling Curve



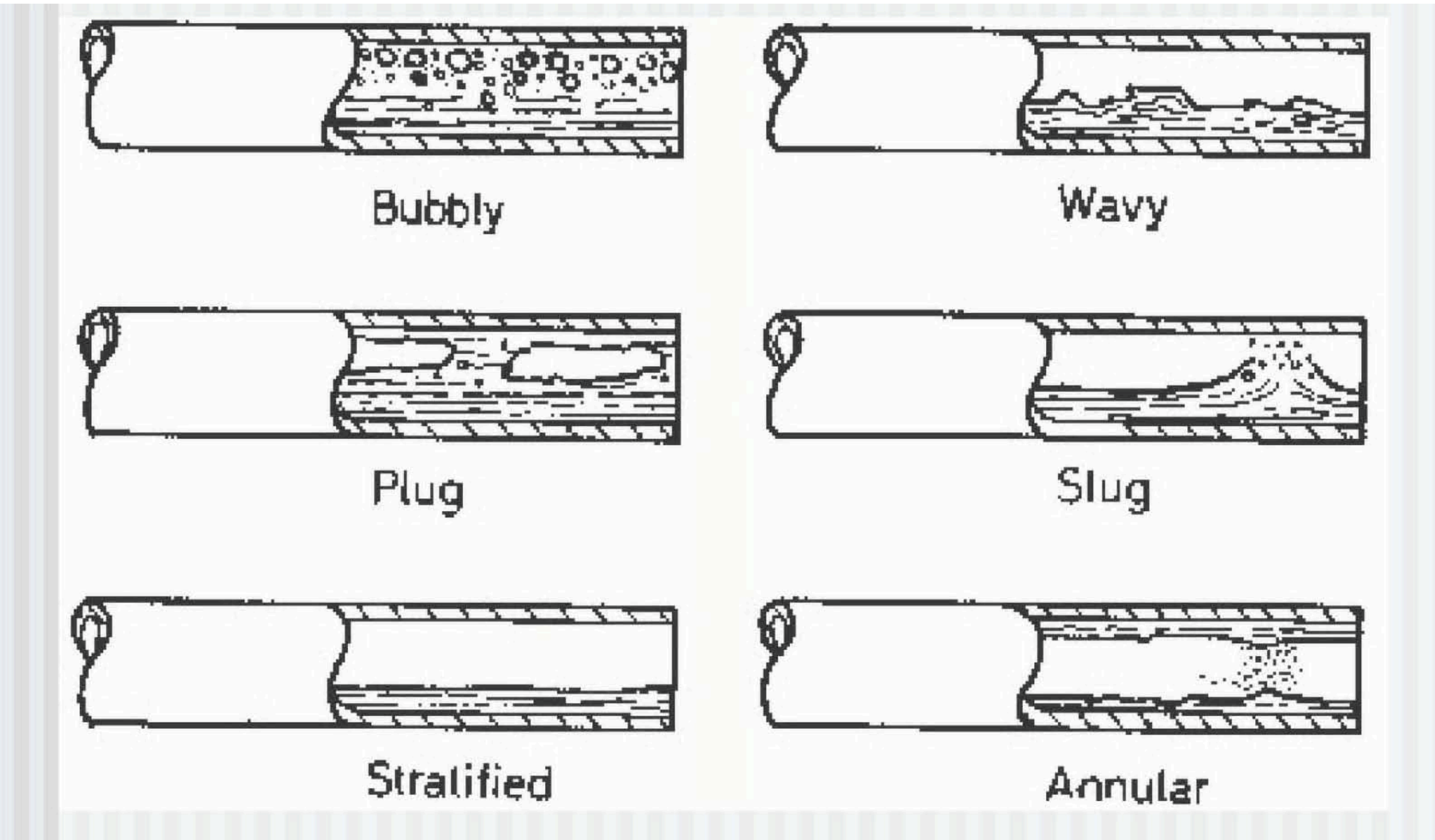
# Definitions for Transition points

- Onset of nucleate boiling. Transition point between single-phase and boiling heat transfer
- Onset of net vapor generation. Transition point between single-phase and two-phase flow (mainly for pressure-drop calculations)
- Saturation point.  Boiling initiation point in an equilibrium system.
- Critical heat flux point.  Transition point between nucleate boiling and transition/film boiling.
- Minimum film-boiling point.  Transition point between transition boiling and film boiling.

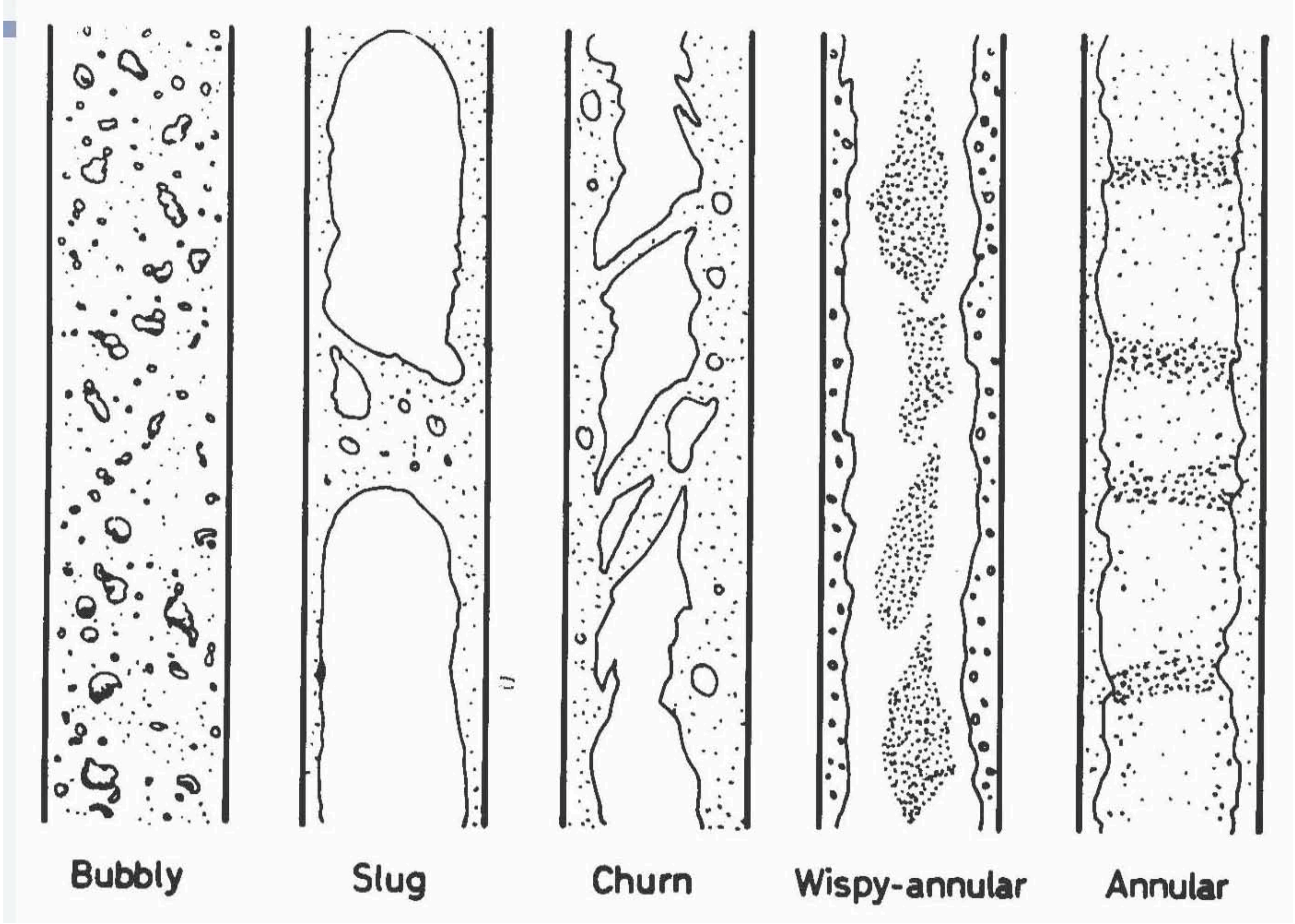
# Flow Patterns

- Distribution of phases inside a confined area
- Depend strongly on liquid and vapor velocities
- Channel geometry
- Surface Heating

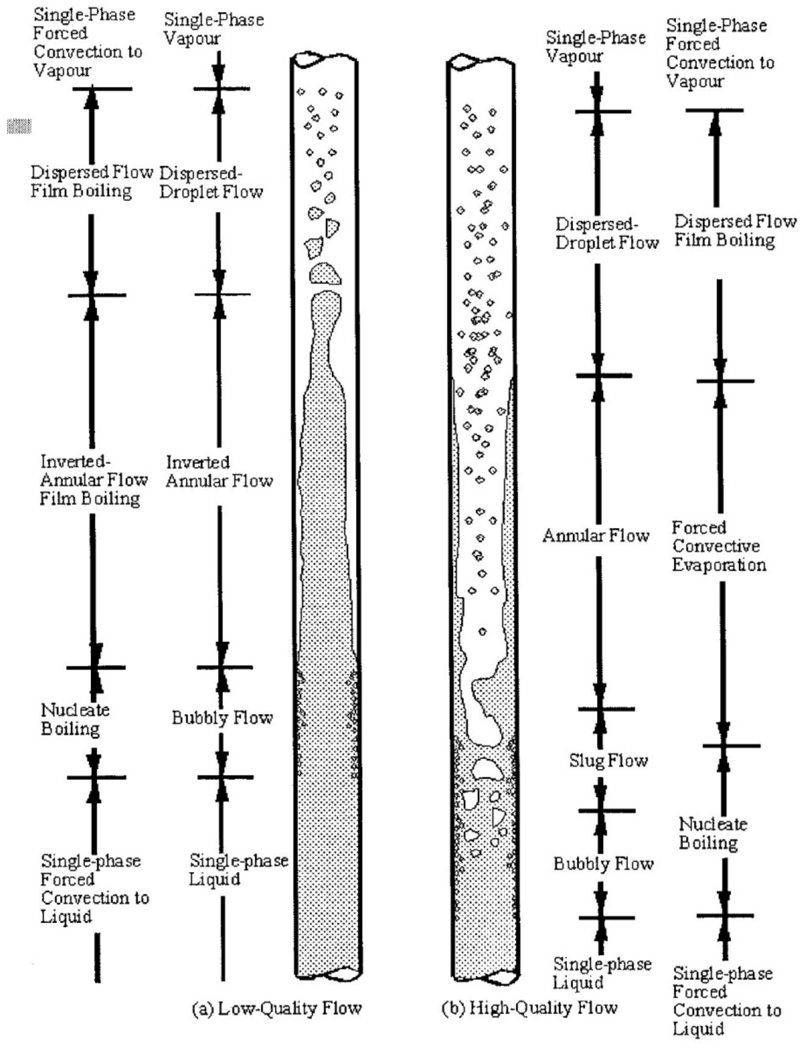
# Flow Patterns in Horizontal Flow



# Flow Patterns in Vertical Flow



# Vertical Heated Channels



# *Single Phase Forced Convection Heating*

- Dittus-Boelter correlation was previously described for PWR steady state heat removal
- Dittus-Boelter correlation is appropriate for subcooled region of BWR fuel channel (*before boiling starts*)

$$h_{film} = (k / D_h)\ 0.023\ Pr^{0.4}\ Re^{0.8}$$

- Above subcooled region – different $h_{film}$ model would apply
- *ALSO:* When modeling *cooling on tube*:

$$h_{film} = (k / D_h)\ 0.023\ Pr^{0.3}\ Re^{0.8}$$

- Example problem: heat transfer in subcooled region
- Assume: $1000$ *psia*, $T_{in} = 515°F$, $V_f = 6.8$ *ft./sec.*
- Use standard dimensions of GE 8x8 Fuel Bundle
- Calculate $h_{film}$

# Example $h_{film}$ Calculation for 8x8 BWR Fuel

**Calculation of $h_{film}$ Conductance Based on Dittus-Boelter Correlation:**

| | | |
|---|---|---|
| Fuel Assembly Pitch in cm.: | $p := 1.62$ | |
| Clad Outer Radius in cm: | $Rc := \frac{1.25}{2}$ | |
| Hydraulic diameter in cm: | $De := 2Rc\cdot\left[\frac{4}{\pi}\cdot\left(\frac{p}{2Rc}\right)^{2} - 1\right]$ | $De = 1.423$ |
| Hydraulic diameter in in: | $Dh := De\cdot 0.394$ | $Dh = 0.561$ |
| Conductivity of Water in BTU/ft.hr.F: | $kw := 0.3435$ | For 515 F, 1000 psi kw= 0.305 BTU/hr ft F |
| Prandtl Number: 515 F, 1000 psi | $Pr := 0.87$ | |
| Kinematic Viscosity in $ft^2$./sec. 515F, 1000 psi | $\upsilon := 1.38\cdot 10^{-6}$ | |
| Flow velocity in ft/sec: | $vf := 6.82$ | Taken from Table 4.4-1 River Bend FSAR |
| Reynolds Number (lengths computed in ft): | $Re := \left(\frac{Dh}{12}\right)\cdot\frac{vf}{\upsilon}$ | $Re = 2.309 \times 10^{5}$ |
| Nusselt Number: | $Nu := 0.023\cdot Pr^{0.4}\cdot Re^{0.8}$ | |
| | $Nu = 424.935$ | |
| Calculated film conductance in BTU/hr.ft.$^2$ F: | $hfilm := \frac{kw}{Dh}\cdot Nu$ | $hfilm = 260.309$ |

# *Nucleate Boiling*

- Thom correlation is one commonly used for evaluating nucleate boiling:

- $q_{THOM} = 0.05358 \times exp(P/630) \times (T_c - T_{sat}(P))^2$

- $q_{THOM}$ is the heat transfer rate in BTU/sec.ft.$^2$
- $P$ is pressure in psia
- $T_c$ is clad surface temperature in °F
- $T_{sat}$ is saturation temperature for pressure: $P$

# *Critical Heat Flux*

The EPRI correlation [Columbia University 1982] can be written as:

$$q''_{CHF} = \frac{1}{0.0036}\frac{AF_A - x_{in}}{CF_C F_g F_{nu} + \dfrac{h - h_{in}}{0.0036 \cdot q'' \cdot h_{fg}}}$$

with:

$$A = 0.5328 \cdot P_r^{0.1212} \cdot (0.0036 \cdot G)^{(-0.3040 - 0.3285 \cdot P_r)}$$

$$C = 1.6151 \cdot P_r^{1.4066} \cdot (0.0036 \cdot G)^{(0.4843 - 2.0749 P_r)}$$

and:

| | | |
|---|---|---|
| $q''_{CHF}$ | = | critical heat flux (Btu/s/ft$^2$), |
| $q''$ | = | local heat flux (Btu/s/ft$^2$), |
| $G$ | = | coolant mass flux (lbm/s/ft$^2$), |
| $P_r$ | = | critical pressure ratio (= system reference pressure/critical pressure), |
| $h$ | = | local enthalpy (Btu/lbm), |
| $h_{in}$ | = | inlet enthalpy (Btu/lbm), |
| $h_{fg}$ | = | vaporization enthalpy (Btu/lbm). |

$F_A$, $F_C$, $F_g$ and $F_{nu}$ are optional factors which correct the critical heat flux for various effects; otherwise they are assigned to the value of 1.0.

The correction for cold wall that can be applied to subchannels adjacent to BWR canister walls, is represented as a function of the coolant mass flux in the following way:

$$F_A = (0.0036 \cdot G)^{0.1}$$
$$F_C = 1.183\ (0.0036 \cdot G)^{0.1}$$

The correction for grid spacers is related to the grid pressure loss coefficient $C_g$ which is supplied in input as follows:

$$F_g = 1.3 - 0.3C_g$$

Finally, the correction for nonuniform axial heat flux at axial level X is written as::

$$F_{nu} = 1.0 + \frac{Y - 1}{1 + 0.0036 \cdot G}$$

$$Y = \frac{\int_0^X q''(X)dX}{q''(X)X}$$

with Y=1 for an axially uniform heat flux.

This is an example BWR Fuel Bundle CHF correlation developed by EPRI

# *Transition Boiling*

## *Something Like This Would Be For LOCA*

The modified Condie-Bengtson for high flowrate transition boiling is as follows:

$$q''_{TB} = C_1 e^{-\frac{1}{2}\sqrt{T_w - T_{sat}}} \left(T_w - T_{sat}\right)$$

where:

$$C_1 = \frac{q''_{CHF} - q''_{FB}}{T_{CHF} - T_{sat}} e^{\frac{1}{2}\sqrt{T_{CHF} - T_{sat}}}$$

$q''_{CHF}$ = critical heat flux (Btu/s/ft$^2$),
$q''_{FB}$ = $h_{FB}(T_{CHF}-T_{sat})$=film boiling heat flux at Critical Heat Flux temperature (Btu/s/ft$^2$),
$q''_{TB}$ = transition boiling heat flux (Btu/s/ft$^2$).

Therefore, for $T_w$=$T_{CHF}$:

$$q''_{TB}=q''_{CHF}-q''_{TB}$$

Since the film boiling flux will be added to the transition boiling component, the boiling curve turns out to be continuous at the CHF temperature.

# *Film Boiling*

## *Something Like This Would Be For LOCA*

$$q''_{FB} = H_{FB}\left(T_w - T_{sat}\right)$$

$$H_{FB} = 0.052\frac{k_g}{D_h}\mathrm{Re}_{hom}^{0.688}\mathrm{Pr}_f^{1.26} / \gamma^{1.06}$$

$$\gamma = 1.0 - 0.1\left(1-x\right)\left(\frac{\rho_f}{\rho_g} - 1\right)^{0.4}$$

$$\mathrm{Pr}_f = \left(\frac{C_{pv}\mu_v}{k_v}\right)_f$$

$$\mathrm{Re}_{hom} = \frac{GD_h x}{\mu_g \alpha} = \frac{GD_h}{\mu_g}\left[x + \frac{\rho_g}{\rho_f}\left(1-x\right)\right]$$

Vapor properties are evaluated at the film temperature $T_f = 1/2(T_w + T_{sat})$ and the homogeneous void correlation (3.20) is used for $x/\alpha$.

| | | |
|---|---|---|
| $\alpha$ | = | void fraction, |
| $x$ | = | flowing vapor quality, |
| $k_g$ | = | thermal conductivity of saturated vapor (Btu/s/ft/F), |
| $\rho_f$ | = | saturated liquid density ($lbm/ft^3$), |
| $\rho_g$ | = | saturated vapor density ($lbm/ft^3$), |
| $\mu_g$ | = | dynamic viscosity of saturated vapor (lbm/s/ft), |
| $G$ | = | coolant mass flux ($lbm/s/ft^2$), |
| $C_{pv}$ | = | specific heat of superheated vapor (Btu/lbm/F), |
| $\mu_v$ | = | dynamic viscosity of superheated vapor (lbm/s/ft), |
| $k_v$ | = | thermal conductivity of superheated vapor (Btu/s/ft/F). |

# Thermal Limits of Operation

- MAPLHGR – Maximum linear heat generation rate is based on burn-up and not exceeding maximum fuel temperature limits of 2200 °F during LOCA.
- LHGR-Linear heat generation rate limit is 13.4 kw/ft as a conservative limit to ensure that 1% plastic strain on the clad is not exceeded.
- MCPR- Minimum Critical Power Ratio is thermal hydraulic limits of the fluid in the core and is calculated by GEXL correlation, which has been developed based on experiments to avoid Boiling Transition.

# *5. Steady State Core Temperature Profiles*

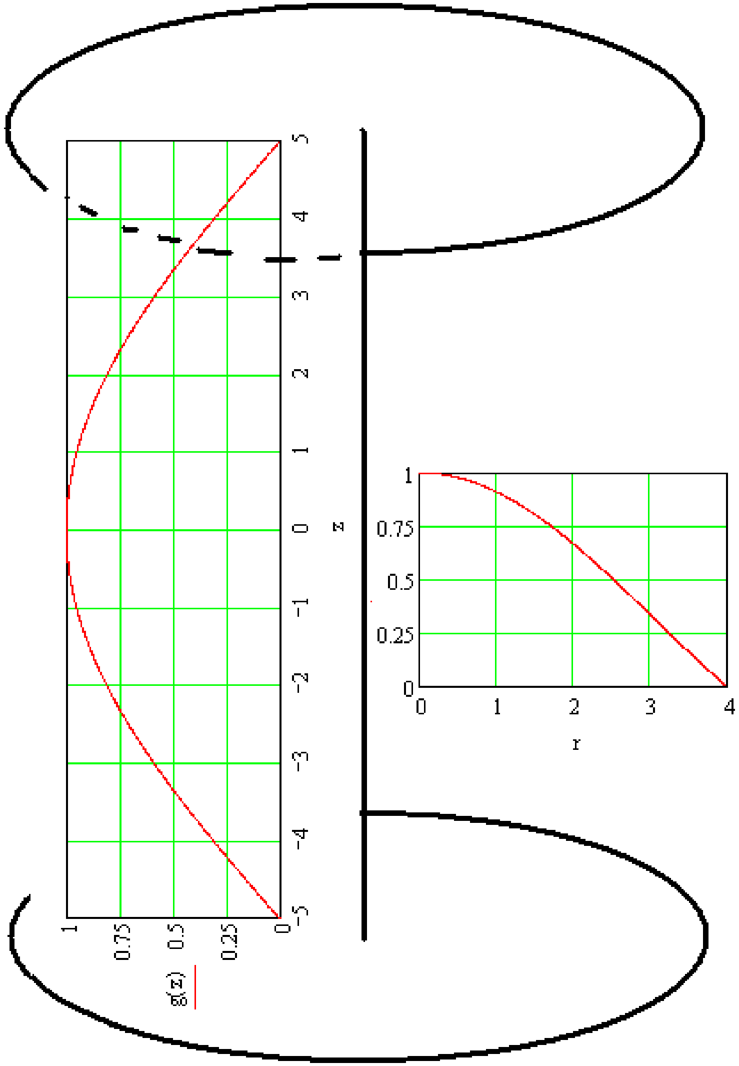
# *BWR Axial Heat Transfer*

# BWR Axial Heat Transfer

- Recall: Axial, radial distribution derived earlier (same as in PWR)
- $\Phi(r,z) = \Phi_o J_o(2.405r/R)Cos(\pi z/H)$
- Again assume linear power density in individual rod given by:

  $q(z) = q_o Cos(\pi z/H)$

  $q_o = (\pi R_c^2)\, E_f \sum_f \Phi_o\, J_o(2.405r/R)$
- Energy balance along single rod in BWR must now reflect heating subcooled water up to saturation point below: $H_{BOIL}$
- Above $H_{BOIL}$: ***boiling heat transfer***



# BWR Axial Heat Transfer

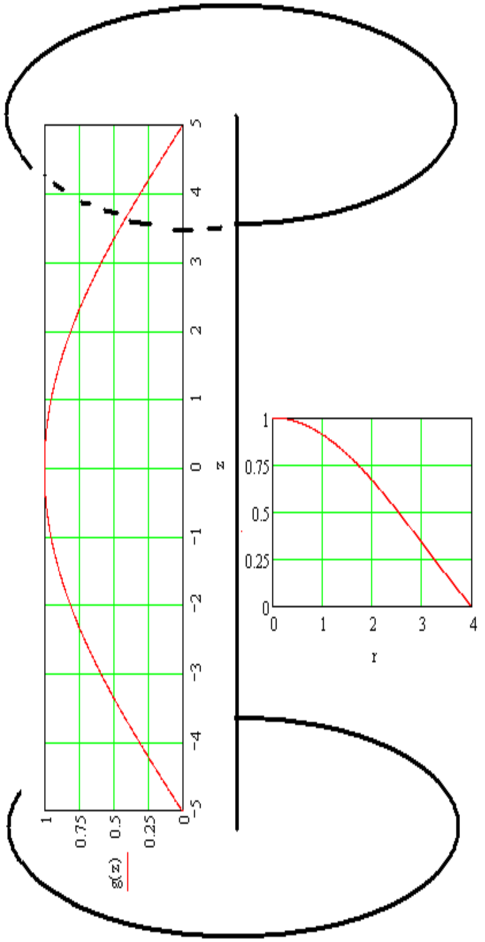
- As subcooled water enters heated channel
- Temperature rises until boiling point $T_{sat}(P)$ at $H_{BOIL}$:

$$T(z,P) = T_{in}(P) + \int_{\frac{-H_{eff}}{2}}^{z} \frac{q(z)}{WC_p(P)}dz$$

$$h(z,P) = h_{in}(P) + \int_{\frac{-H_{eff}}{2}}^{z} \frac{q(z)}{W}dz$$

- Above $H_{BOIL}$ further heat addition only increases steam content, not temperature
- Enthalpy rise: $h(z,P) = h_{sat}(P) + \chi(z)h_{fg}(P)$

$$where: \chi(z) = \int_{H_{BOIL}}^{z} \frac{q(z)}{Wh_{fg}(P)}dz$$



# *Simulation of Uniform Linear Power Density*

**Channel inlet flow in lb-m/sec.:** $W := 0.6$

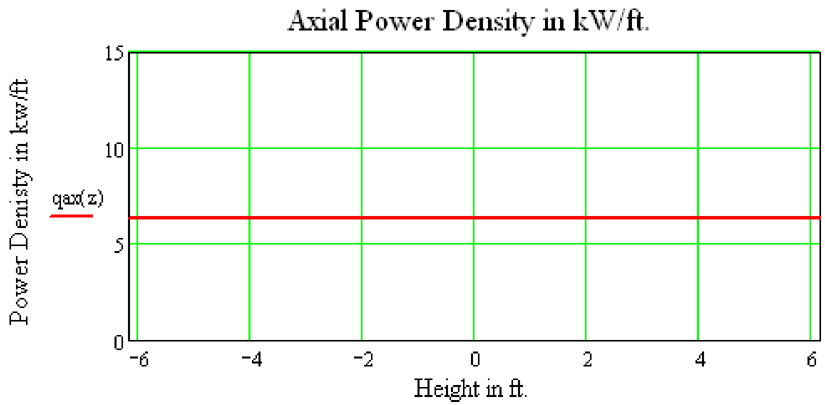
**Reactor Inlet enthalpy:** assuming 1000 psia, T=515 F — $hin := 505.75$ BTU/lb-m

**Saturated liquid enthalpy:** assuming 1000 psia, Tsat=544.58 F — $hf := 542.6$ BTU/lb-m

**Vaporization enthalpy:** assuming 1000 psia, Tsat=544.58 F — $hfg := 650.4$ BTU/lb-m

**Peak axial power in BTU/sec.ft** — $qo := 12.71$

**Axial Power Distribution in BTU/sec ft:** — $qax(z) := qo \cdot \frac{1}{2}$




**Routine to generate Subcooled Axial enthalpy profile:**

$$f(z) := hin + \frac{1}{W} \cdot \int_{-\frac{Heff}{2}}^{z} qax(z)\, dz \qquad h(z) := \begin{cases} f(z) & \text{if } f(z) \le hf \\ hf & \text{otherwise} \end{cases}$$

**Routine to find the point where boiling starts:**

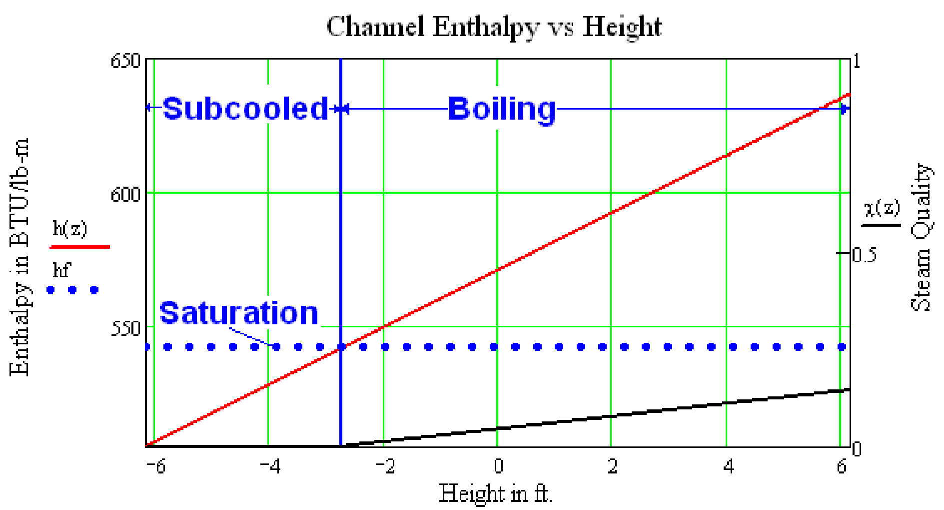
$$F(z) := f(z) - hf \qquad Hboil := root\left(F(z), z, \frac{-Heff}{2}, \frac{Heff}{2}\right) \qquad Hboil = -2.689$$

Height above bottom of core: $Hbot := \frac{Heff}{2} + Hboil$ $\quad Hbot = 3.479$

**Routine to generate the Axial Quality Profile and Axial Enthalpy:**

$$\chi(z) := \begin{cases} 0 & \text{if } f(z) \le hf \\ \int_{Hboil}^{z} \frac{qax(z)}{W \cdot hfg}\, dz & \text{if } f(z) > hf \end{cases}$$

$$h(z) := \begin{cases} f(z) & \text{if } f(z) \le hf \\ hf + \chi(z) \cdot hfg & \text{otherwise} \end{cases}$$



# Simulation of Cosine Linear Power Density

**Channel inlet flow in lb-m/sec.:** $W := 0.6$

**Reactor Inlet enthalpy:** assuming 1000 psia, T=515 F — $hin := 505.75$ BTU/lb-m

**Saturated liquid enthalpy:** assuming 1000 psia, Tsat=544.58 F — $hf := 542.6$ BTU/lb-m

**Vaporization enthalpy:** assuming 1000 psia, Tsat=544.58 F — $hfg := 650.4$ BTU/lb-m

**Peak axial power in BTU/sec.ft** — $qo := 12.71$

**Axial Power Distribution in BTU/sec ft:** $qax(z) := qo \cdot \cos\left(\frac{\pi \cdot z}{Heff}\right)$

**Routine to generate Subcooled Axial enthalpy profile:**

$$f(z) := hin + \frac{1}{W} \cdot \int_{\frac{-Heff}{2}}^{z} qax(z)\, dz \qquad h(z) := \begin{cases} f(z) & \text{if } f(z) \le hf \\ hf & \text{otherwise} \end{cases}$$

**Routine to find the point where boiling starts:**

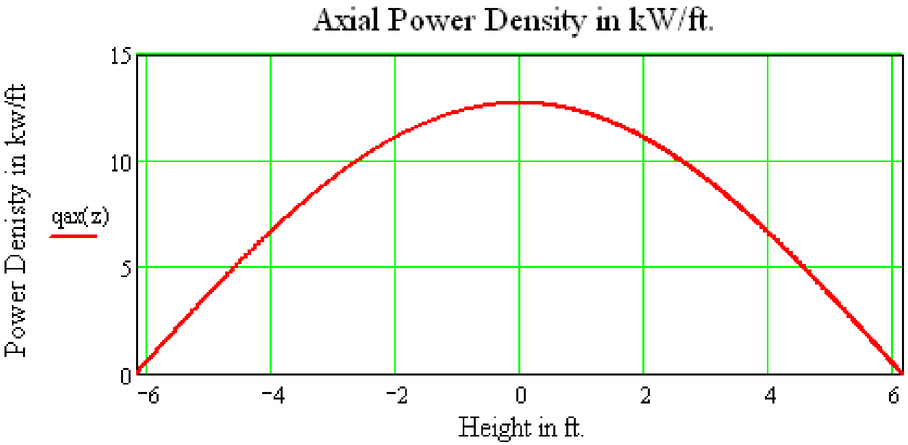
$$F(z) := f(z) - hf \qquad Hboil := root\left(F(z), z, \frac{-Heff}{2}, \frac{Heff}{2}\right) \qquad Hboil = -2.32$$

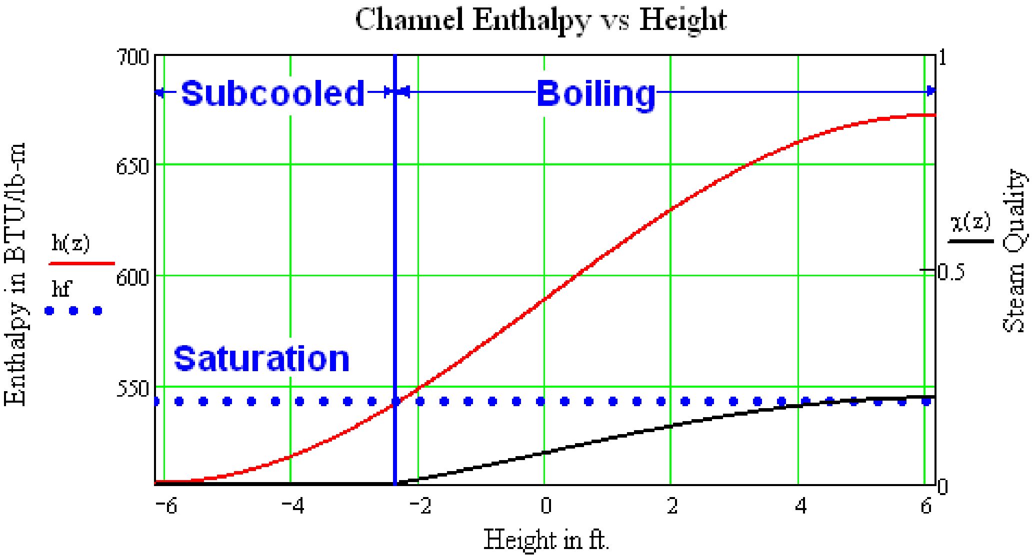
Height above bottom of core: $Hbot := \frac{Heff}{2} + Hboil$ $\quad Hbot = 3.848$

**Routine to generate the Axial Quality Profile and Axial Enthalpy:**

$$\chi(z) := \begin{cases} 0 & \text{if } f(z) \le hf \\ \int_{Hboil}^{z} \frac{qax(z)}{W \cdot hfg}\, dz & \text{if } f(z) > hf \end{cases}$$

$$h(z) := \begin{cases} f(z) & \text{if } f(z) \le hf \\ hf + \chi(z) \cdot hfg & \text{otherwise} \end{cases}$$



# Simulation of Cosine Linear Power Density

**BWR Fuel Data:**

Fuel Rod Length in ft.: $Heff := 12.336$

Fuel Pellet Radius in in.: $Ro := \frac{0.41578}{2}$ $\quad Ro = 0.208$

Uranium Dioxide
thermal conductivity in BTU/hr.ft.F: $kfuel := 2.02$

Gap Conductance in BTU/hr.ft²F: $hgap := 1200$

Zircaloy Cladding outer radius in in.: $Rc := \frac{0.4921}{2}$ $\quad Rc = 0.246$

Zircaloy thermal conductivity
in BTU/hr.ft.F $kc := 7.37$

$Tin := 515 \qquad Cp := 1.17$

**Coolant Temperature vs. height distribution:**

$$T(z) := \begin{cases} \left( Tin + \displaystyle\int_{\frac{-Heff}{2}}^{z} \frac{qax(z)}{W \cdot Cp}\, dz \right) & \text{if } z < Hboil \\ Tsat & \text{otherwise} \end{cases}$$

**Clad Surface Temperature vs. height distribution:**

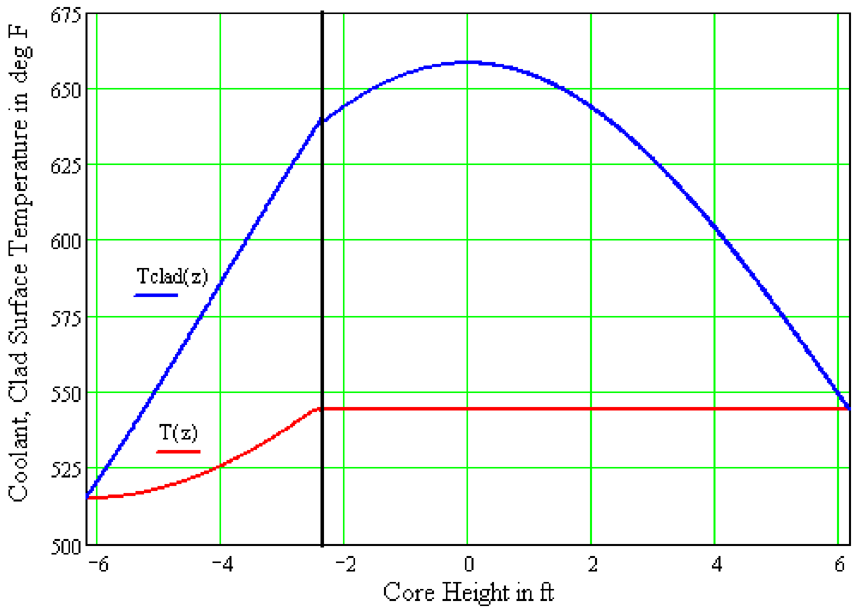
$$Tclad(z) := T(z) + \frac{qax(z) \cdot 3600}{2 \cdot \pi} \cdot \left( \frac{\ln\left(\frac{Rc}{Rc}\right)}{2 \cdot kc} + \frac{1}{Rc \cdot hfilm} \right)$$

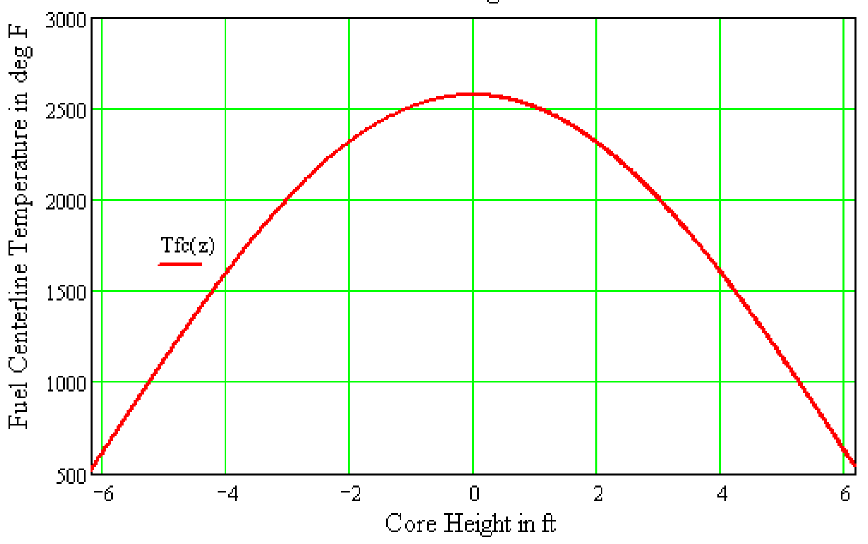
**Heat Flux calculated based upon temperature difference:**

$$qfc(z) := \frac{2 \cdot \pi \cdot Rc\, hfilm \cdot (Tclad(z) - T(z))}{3600}$$

**Fuel Centerline Temperature vs. height distribution:**

$$Tfc(z) := T(z) + \frac{qax(z) \cdot 3600}{2 \cdot \pi} \cdot \left( \frac{1}{2 \cdot kfuel} + \frac{1}{Ro \cdot hgap} + \frac{\ln\left(\frac{Rc}{Ro}\right)}{2 \cdot kc} + \frac{1}{Rc \cdot hfilm} \right)$$

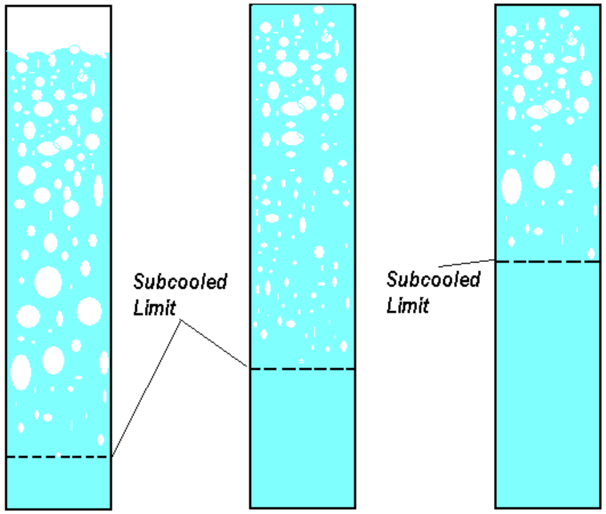


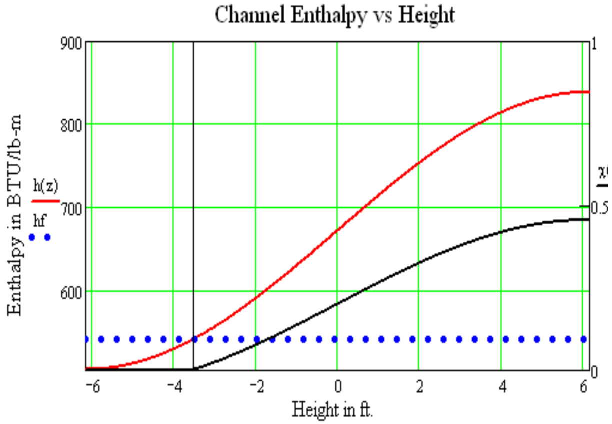
# *6. Fluid Flow and Pressure Drops in Two-Phase Systems*

# Simulation of Variable Recirculation Flow

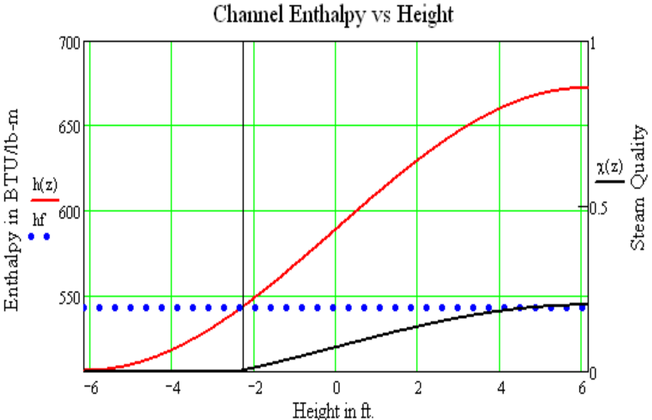
•Previous lecture noted BWR capability to vary recirculation flow to raise/lower power



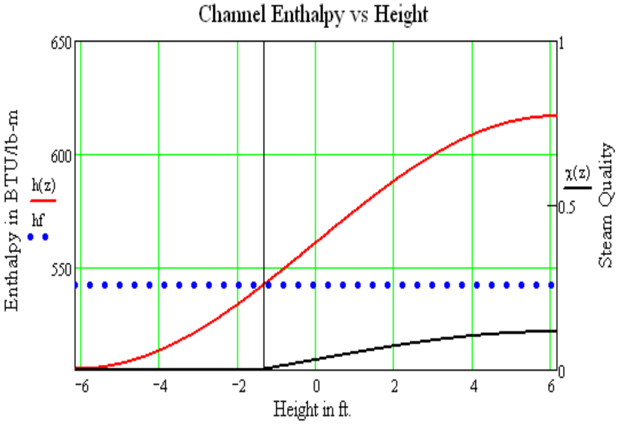

*Effect of Increasing Flow on Subcooled Height and Void Fraction Assuming Same Channel Power*




**W = 0.3 lb-m/sec.** **Hbot = 2.66 ft.**




**W = 0.6 lb-m/sec.** **Hbot = 3.848 ft.**




**W = 0.9 lb-m/sec.** **Hbot = 4.825 ft.**

# *Two Phase Flow Pressure Drop*

# *Pressure Drop in Two Phase System*

- Recall: For single phase flow system in channel, pressure drop in psia can be calculated:

$$\Delta P_{friction} = \left(\frac{fL}{D_h}\right)\frac{\rho\upsilon^2}{2} + \sum_i \left(K_i \frac{\rho\upsilon_i^{\,2}}{2}\right)$$

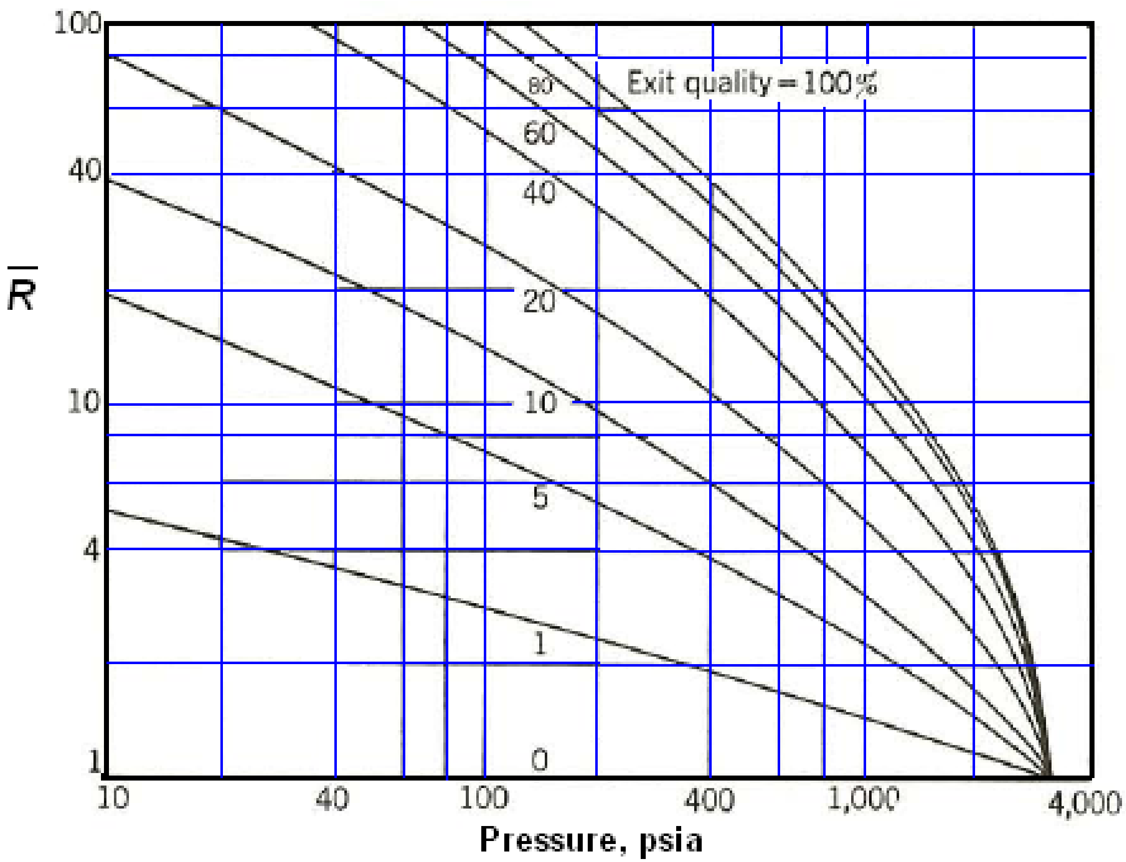
- In two phase system: *pressure drop is larger*
- Experimental tests have lead to a simple working relationship between single phase and two phase pressure drops.
- Following homogeneous two phase pressure drop has been developed for steady state flow conditions:

$$\overline{R} = \frac{\Delta P_{2\phi}}{\Delta P_{1\phi}}$$

- -where: $\Delta P_{2\Phi}$ is calculated assuming all liquid flow at total mass flow rate

# *Martinelli-Nelson Friction Multiplier*

- This is classical approach. Advanced approaches exist
- If equivalent single phase pressure drop is known
- Homogeneous two phase pressure drop is: $\Delta P_{2\phi} = \overline{R} \times \Delta P_{1\phi}$
- Where:




Martinelli-Nelson friction-multiplier correlation

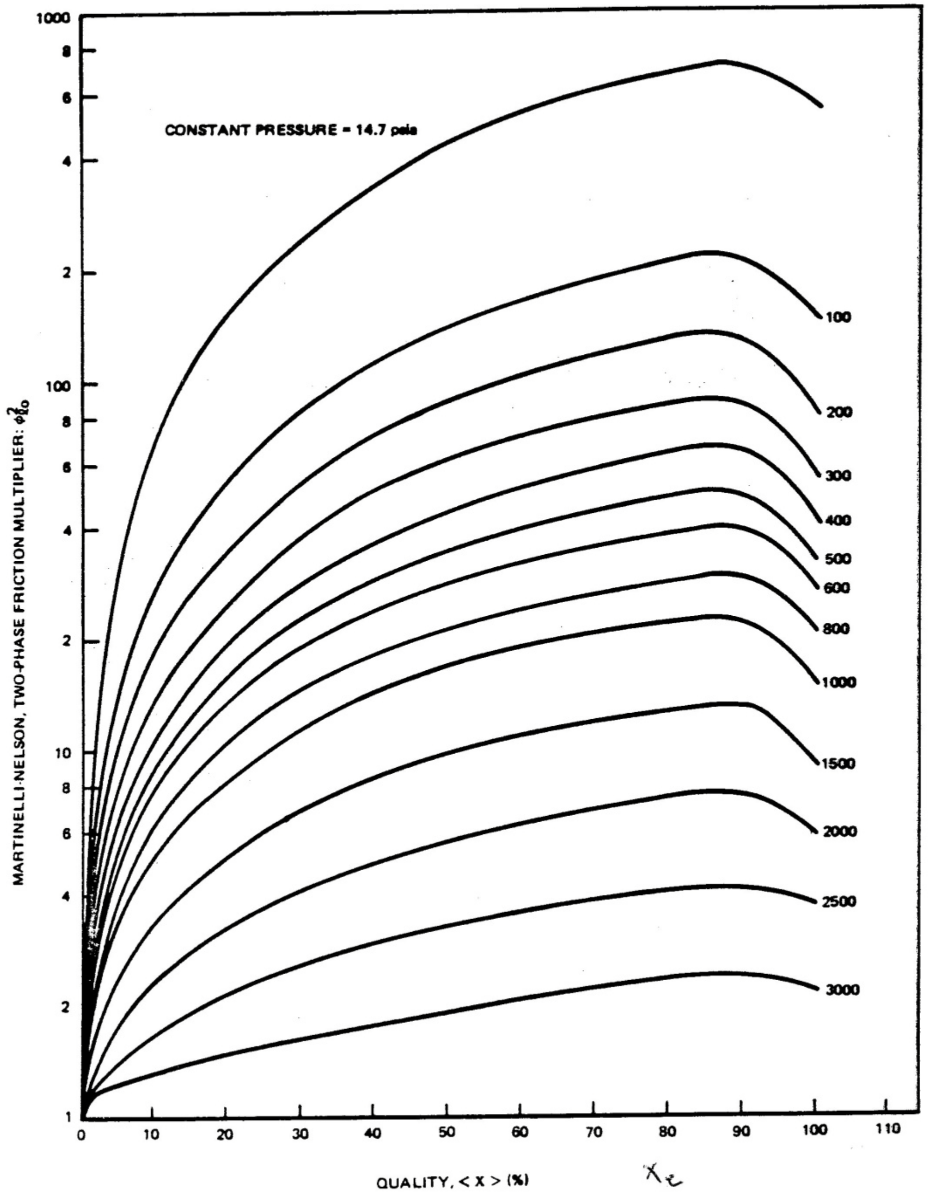
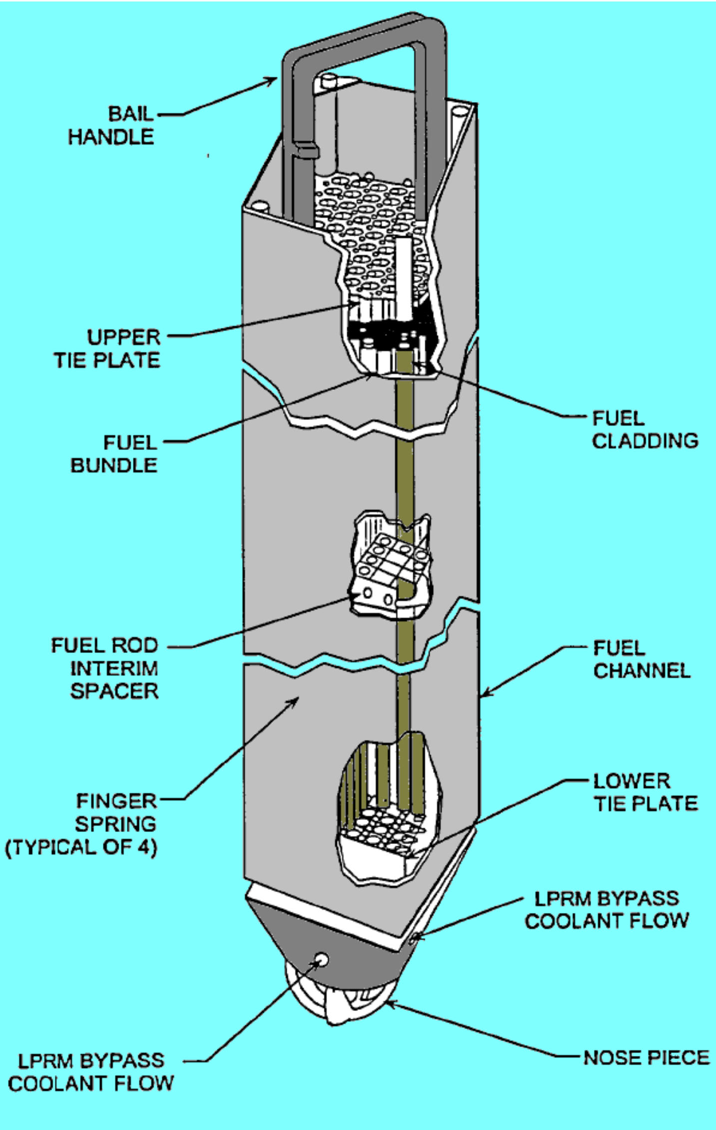
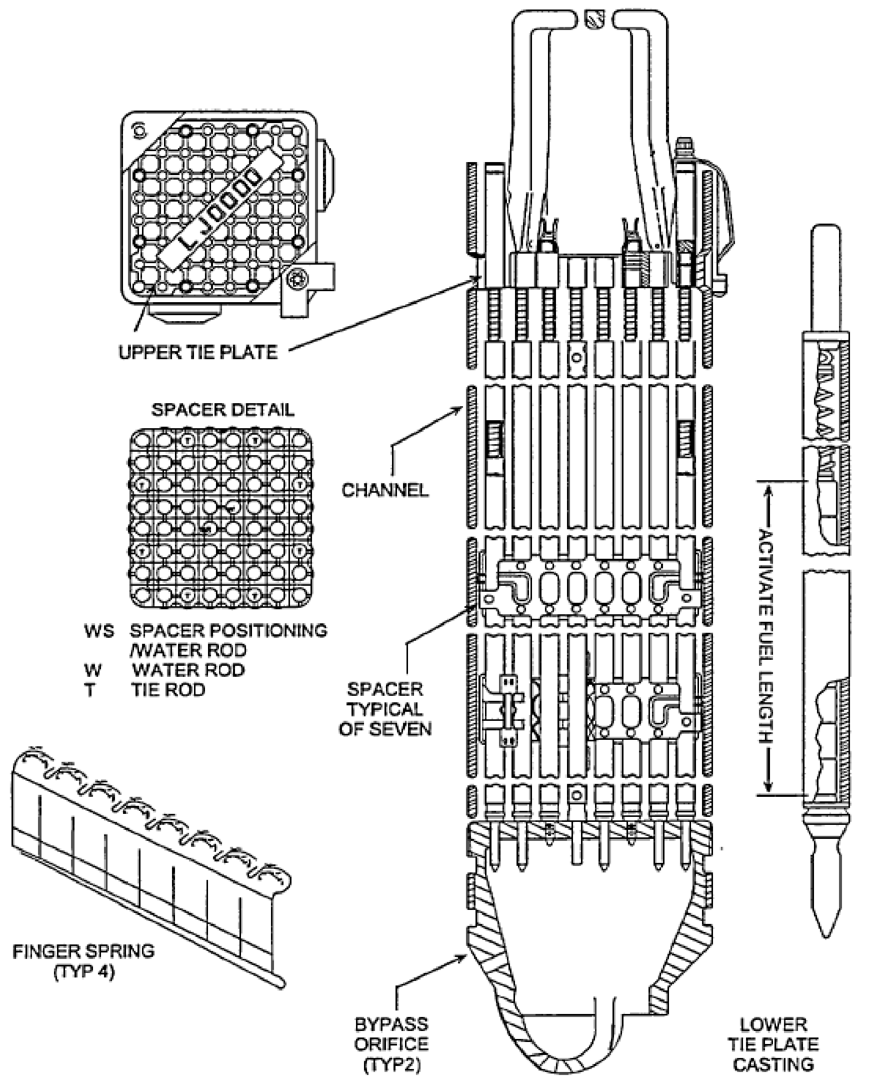



*Figure 3-4. Ratio of Two-Phase to Single-Phase Flow Resistance Versus Quality and Pressure*

# BWR Fuel Bundle Geometry

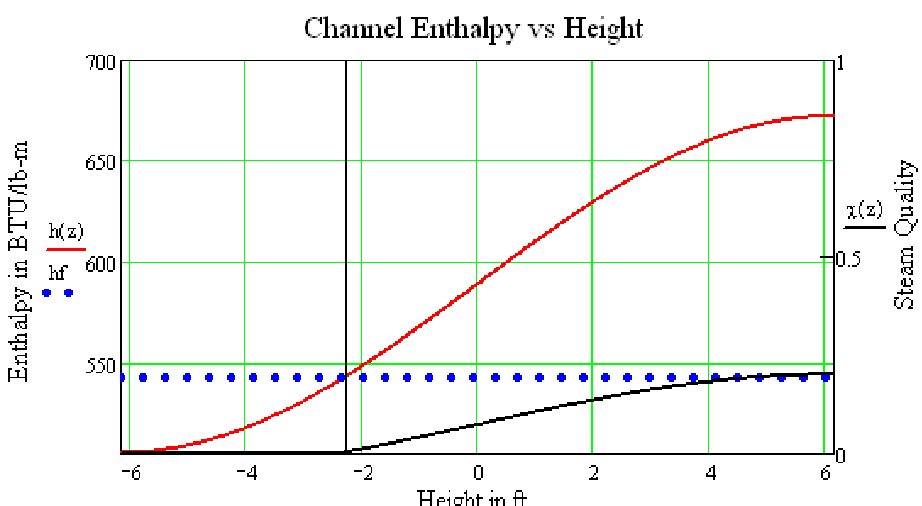


# *BWR Fuel Channel Pressure Drop*

*(Pressure Drops Due to <u>Grid Spacers, Inlet/Outlet Geometry</u> Would Need to be Added!)*

This calculation uses steam quality and bundle geometry from previous example:




**W = 0.6 lb-m/sec.**

**Hbot = 3.848 ft.**

**Calculation of Pressure Drop Across Fuel Bundle:**

| | | |
|---|---|---|
| Moody friction factor: | $f := \frac{0.184}{Re^{0.2}}$ | $f = 0.016$ |
| Hydraulic Diameter in in.: | $Dh = 0.56$ | NOTE: Need to 1/12 to convert to feet. |
| Dynamic viscosity in ft²/sec: at $T_{sat}$, P = 1000 psi | $\upsilon = 1.38 \times 10^{-6}$ | |
| Flow velocity in ft/sec.: | $vf = 6.82$ | |
| Density in lb/ft³ at $T_{sat}$, P = 1000 psi | $\rho := 46.6418$ | |
| Height of subcooled length in ft.: | $Hbot = 3.848$ | |
| Conversion constant from lbm to lbf: | $gc := 4.17 \cdot 10^{8}$ | |
| Pressure drop in subcooled portion of fuel channel in psi: | $\Delta Psc := \left(\frac{f \cdot Hbot}{\frac{Dh}{12}}\right) \cdot \frac{\rho \cdot (vf \cdot 3600)^2}{2 \cdot gc \cdot 144}$ | $\Delta Psc = 0.3$ |
| Length of fuel channel in ft. above boiling point: | $L := Heff - Hbot$ | $L = 8.488$ |
| Exit Quality: | $\chi exit := \chi\left(\frac{Heff}{2}\right)$ | $\chi exit = 0.199$ |
| Martinelli-Nelson friction factor for $\chi_{exit}$ | $R := 7$ | |
| Pressure drop in saturated portion of fuel channel in psi: | $\Delta Psat := R \cdot \left[\left(\frac{f \cdot L}{\frac{Dh}{12}}\right) \cdot \frac{\rho \cdot (vf \cdot 3600)^2}{2gc \cdot 144}\right]$ | $\Delta Psat = 4.636$ |

# *Acceleration Pressure Drop*

- Coolant change of phase causes increase in volume
- Increased volume causes acceleration in fuel channel
- Assuming cross-sectional area within fuel channel is $A_c$
- Force due to change in fluid momentum is:

$$F = \Delta p_a A_c = [ (m_f V_f + m_g V_g ) - m_{tot} V_{in} ]$$

- where: $\Delta p_a$ is pressure drop due to acceleration in psi

- $A_c$ is the channel area in: in.$^2$

- $m_{tot}$ , $m_f$ ,$m_g$ are respectively incoming, and exit fluid/gas mass flow rates in lb-m./hr.

- $V_{in}$, $V_f$ , $V_g$ are respectively corresponding fluid velocities

# Acceleration Pressure Drop

- Solving for $\Delta p_a$ by inserting relationships for $m_f$ ,$m_g$ and $\chi_{exit}$

$$\Delta p_a = [\ (1- \chi_{exit})m_{tot}V_f + \chi_{exit}m_{tot}V_g - m_{tot}V_{in}]$$

$$= (m_{tot}/A_c)\ [\ (1- \chi_{exit})V_f + \chi_{exit}V_g - V_{in}]$$

- Defining mass-flow rate per unit cross-sectional area: $G$ in units of lb-m./hr. in.$^2$ - this equates to:

$$\Delta p_a = G\ [(1- \chi_{exit})V_f + \chi_{exit}V_g - V_{in}]$$

- Considering continuity, fluid exit velocity ($V_f$ ) can be expressed in terms of specific volume: $v_f$

$$V_f = m_f\, v_f\, /A_f = (1- \chi_{exit})m_{tot}v_f\, /A_f = (1- \chi_{exit})m_{tot}v_f\, /(1- \alpha_{\ exit})A_f$$

$$= (1- \chi_{exit})Gv_f\, /(1- \alpha_{\ exit})$$

# Acceleration Pressure Drop

- Similarly – gas flow velocity and inlet fluid velocity are:

$$V_g = \chi_{exit} G v_g / \alpha_{exit} \qquad V_{in} = G v_{in} \approx G\, v_f$$

- Substituting these – acceleration pressure drop becomes:

$$\Delta p_a = G^2 [(1 - \chi_{exit})^2 v_f /(1 - \alpha_{exit}) + \chi_{exit}{}^2 v_g / \alpha_{exit} - v_f]$$
$$= G^2\, v_f\, [(1 - \chi_{exit})^2/(1 - \alpha_{exit}) + \chi_{exit}{}^2 v_g / v_f \alpha_{exit} - 1]$$

- An overall acceleration Multiplier $R_a$ can now be defined:

$$R_a = v_f\, [(1 - \chi_{exit})^2/(1 - \alpha_{exit}) + \chi_{exit}{}^2 v_g / v_f \alpha_{exit} - 1]$$

- Acceleration pressure drops would be calculated:

$$\Delta p_a = G^2 R_a$$

# Example Acceleration Pressure Drop Calculation

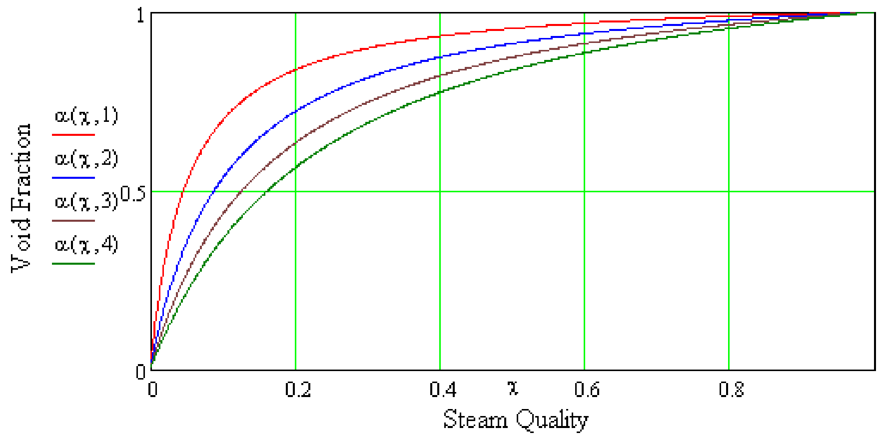
**Assume 1000 psi, $T_{in}$ = 515 F, $\chi_{exit}$ = 0.199, S = 2.5**

Specific volume of liquid in $ft^3/lb$: $vf := 0.02159$

Specific volume of steam in $ft^3/lb$: $vg := 0.44596$

Definition of Void Fraction in terms of Steam Quality and Slip Ratio:

$$\alpha(\chi, S) := \frac{1}{1 + \left[\frac{1-\chi}{\chi} \cdot \left(\frac{vf}{vg}\right) \cdot S\right]} \qquad \alpha(0.199, 2.5) = 0.672$$




**Acceleration Pressure Drop Multiplier**

Acceleration Multiplier definition:

$$Ra(\alpha, \chi) := vf \cdot \left[\frac{(1-\chi)^2}{(1-\alpha)} + \frac{\chi^2 vg}{\alpha \cdot vf} - 1\right]$$

$$Ra(0.672, 0.199) = 0.047$$

Inlet fluid velocity in ft/hr.: $Vf := 6.82 \cdot 3600$

Inlet fluid density in lb-m/$ft^3$ at 1000 psi, 515 F: $\rho in := 48.2393$

Mass flow rate per unit cross-sectional area in lb-m/hr $ft^2$.: $G := \rho in \cdot Vf$ $\qquad G = 1.184 \times 10^6$

Conversion constant: $gc := 4.17 \cdot 10^8$

Definition of acceleration pressure drop in psi

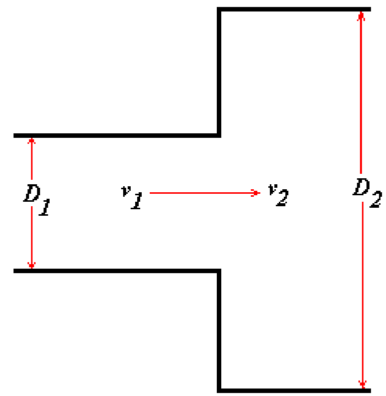
$$\Delta Pa := \frac{Ra(0.672, 0.199) \cdot G^2}{144 \cdot gc} \qquad \Delta Pa = 1.096$$

# 2-Phase Expansion and Contraction Losses

- Recall in treatment of single phase pressure drops:

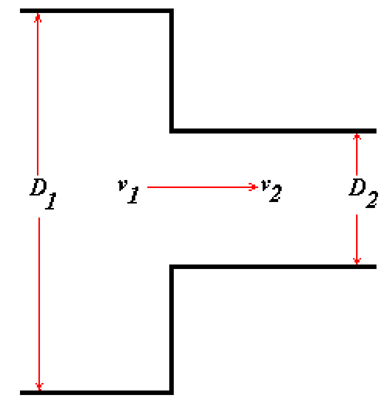
  $$\Delta P = K \rho Vin^2 / 2$$

- Situation for two-phase flow is more complicated
- Corresponding pressure drops for two-phase flow are larger.
- Higher void fractions result in larger pressure drops



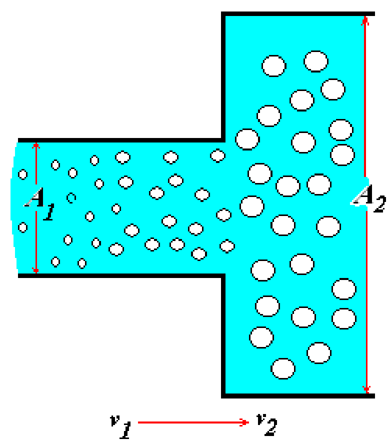

*Abrupt Expansion in Fluid Flow*

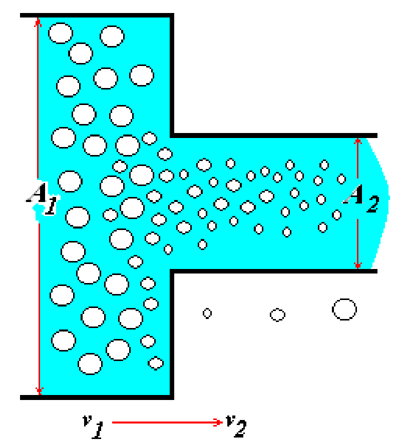
$$K_e = (1 - D_1^2/D_2^2)^2$$




*Abrupt Contraction in Fluid Flow*

| $D_1/D_2$ | 0.8 | 0.6 | 0.4 | 0.2 | 0.0 |
|---|---|---|---|---|---|
| Kc | 0.13 | 0.28 | 0.38 | 0.45 | 0.50 |



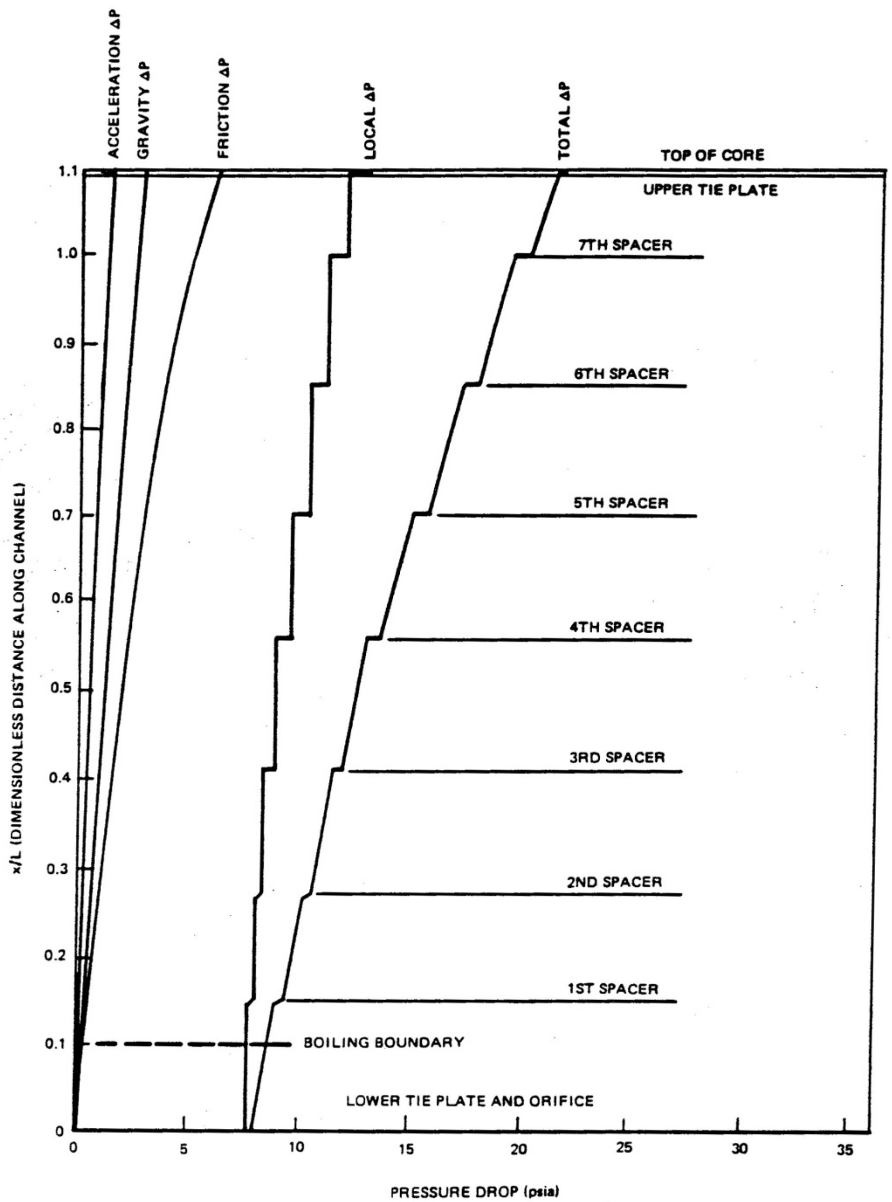
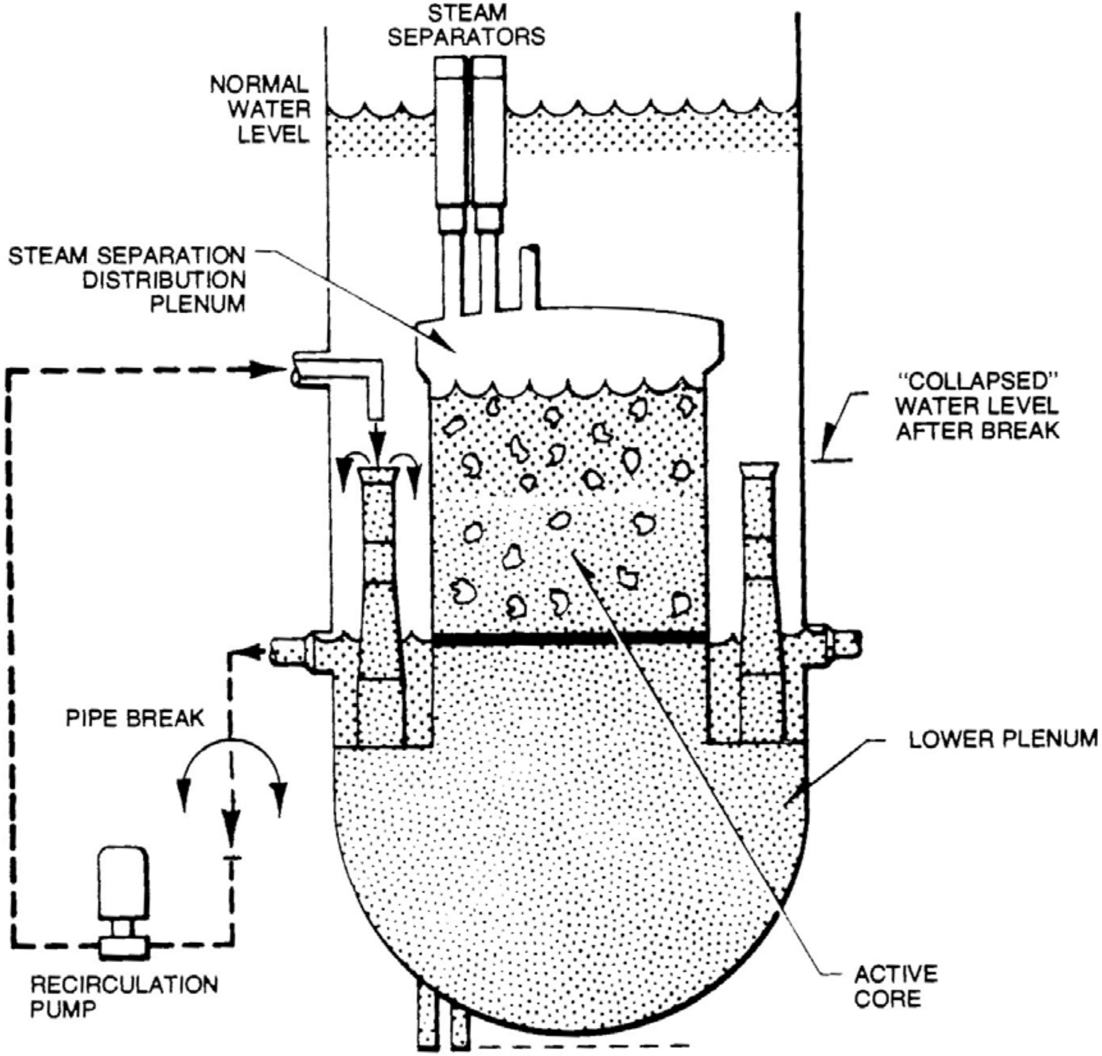


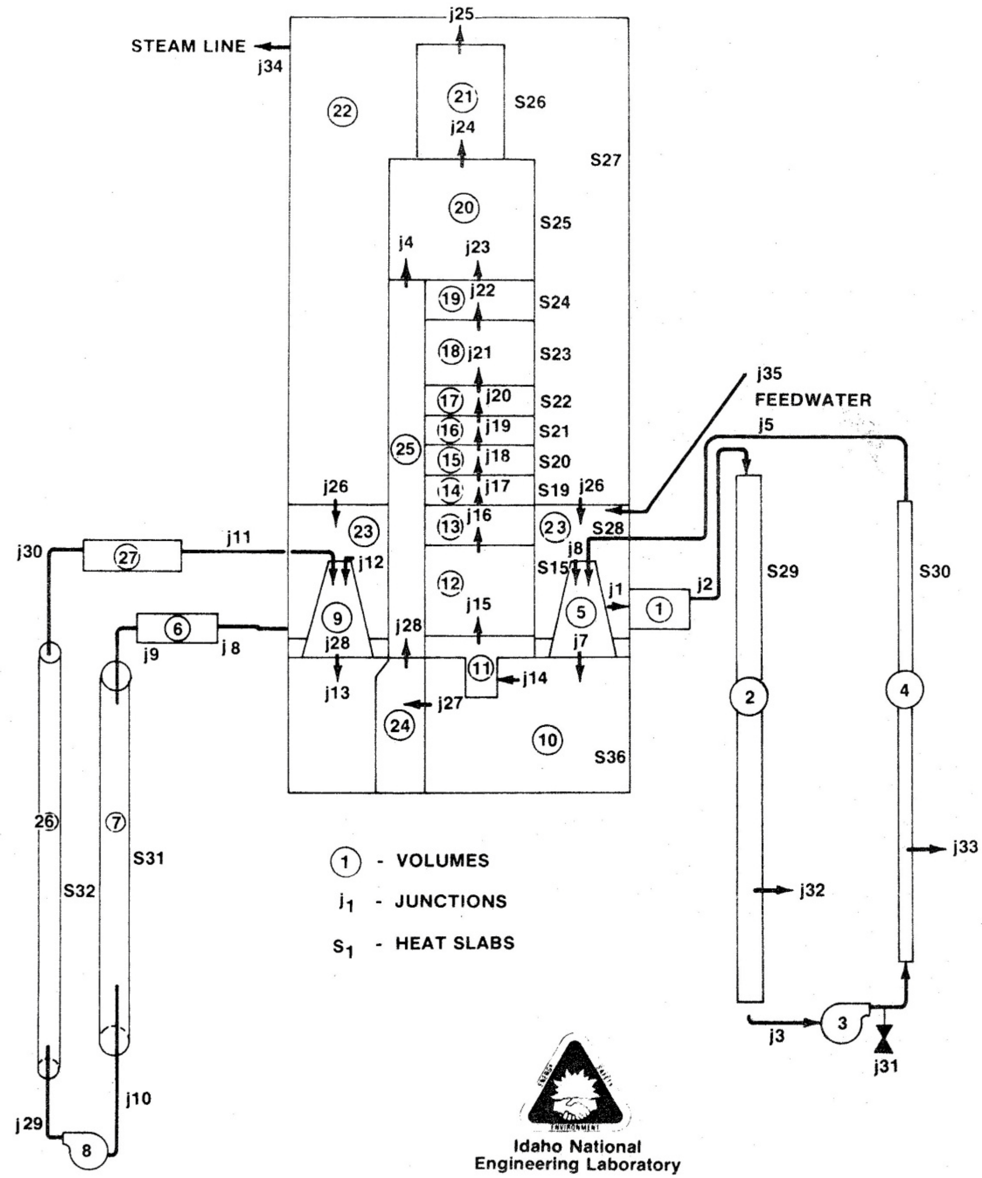

*Figure 3-5. Pressure Drop Components in a Typical Reactor Channel*

# *7. Behavior of System During Accident*

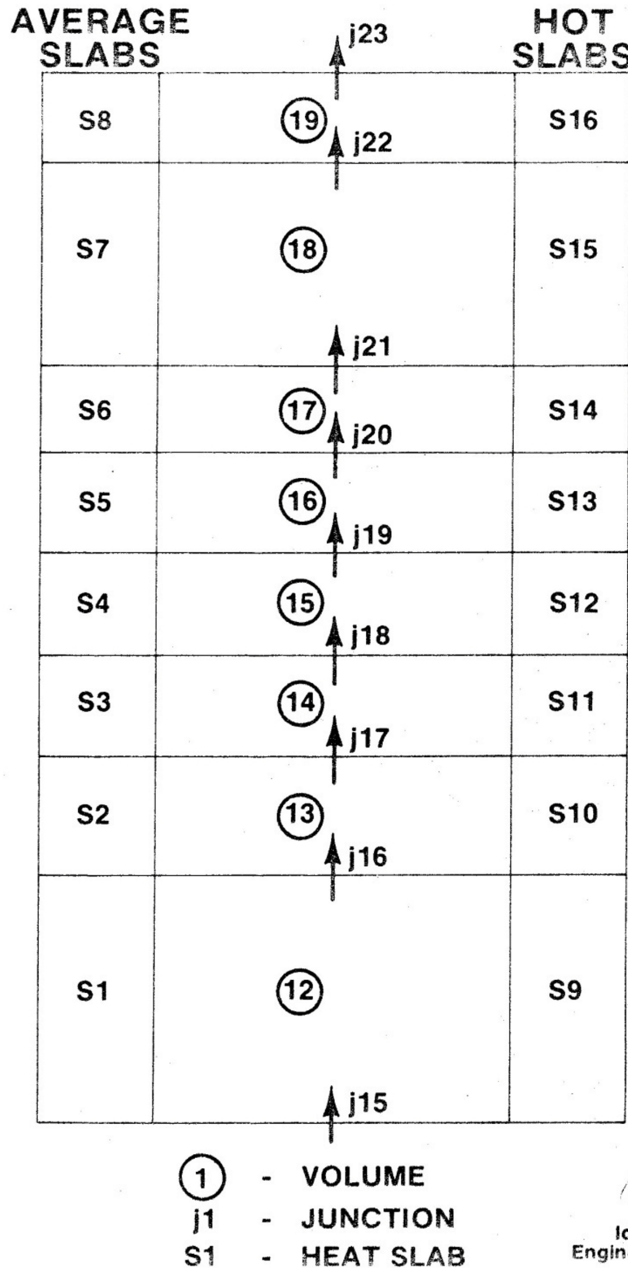
# BWR Break Water Level



# RELAP4 TLTA SYSTEM MODEL

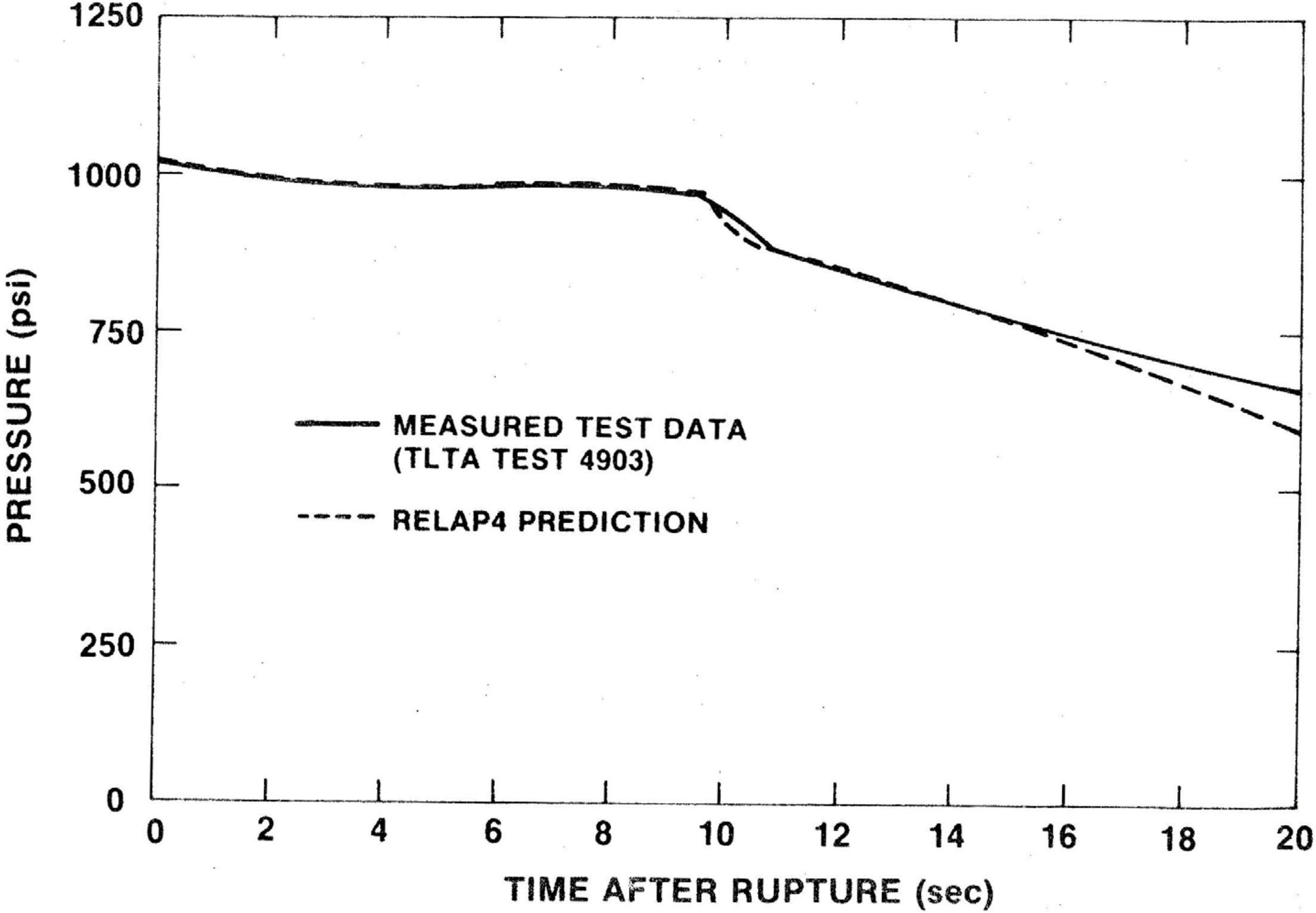


# RELAP4
# TLTA CORE MODEL

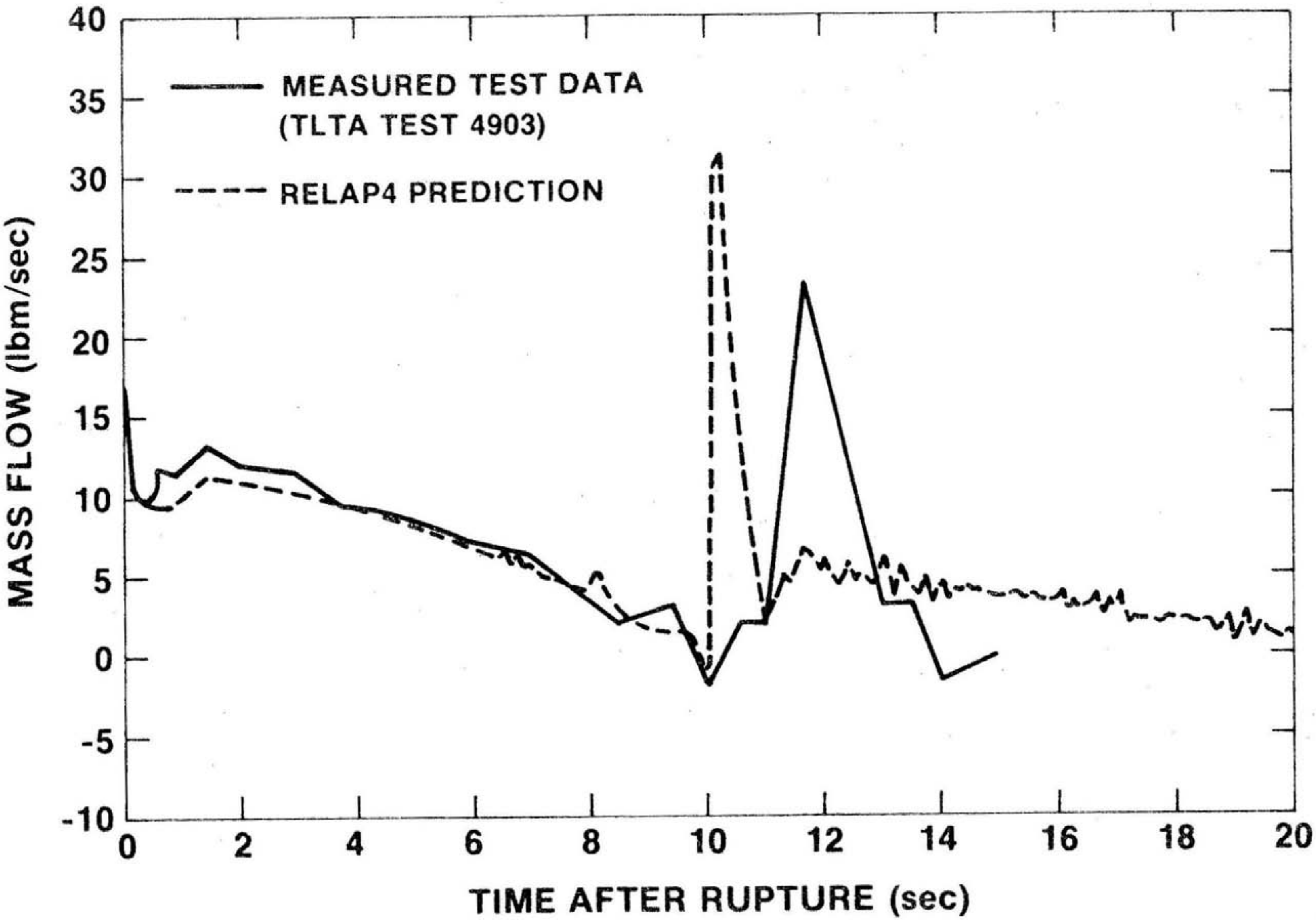



Idaho National Engineering Laboratory

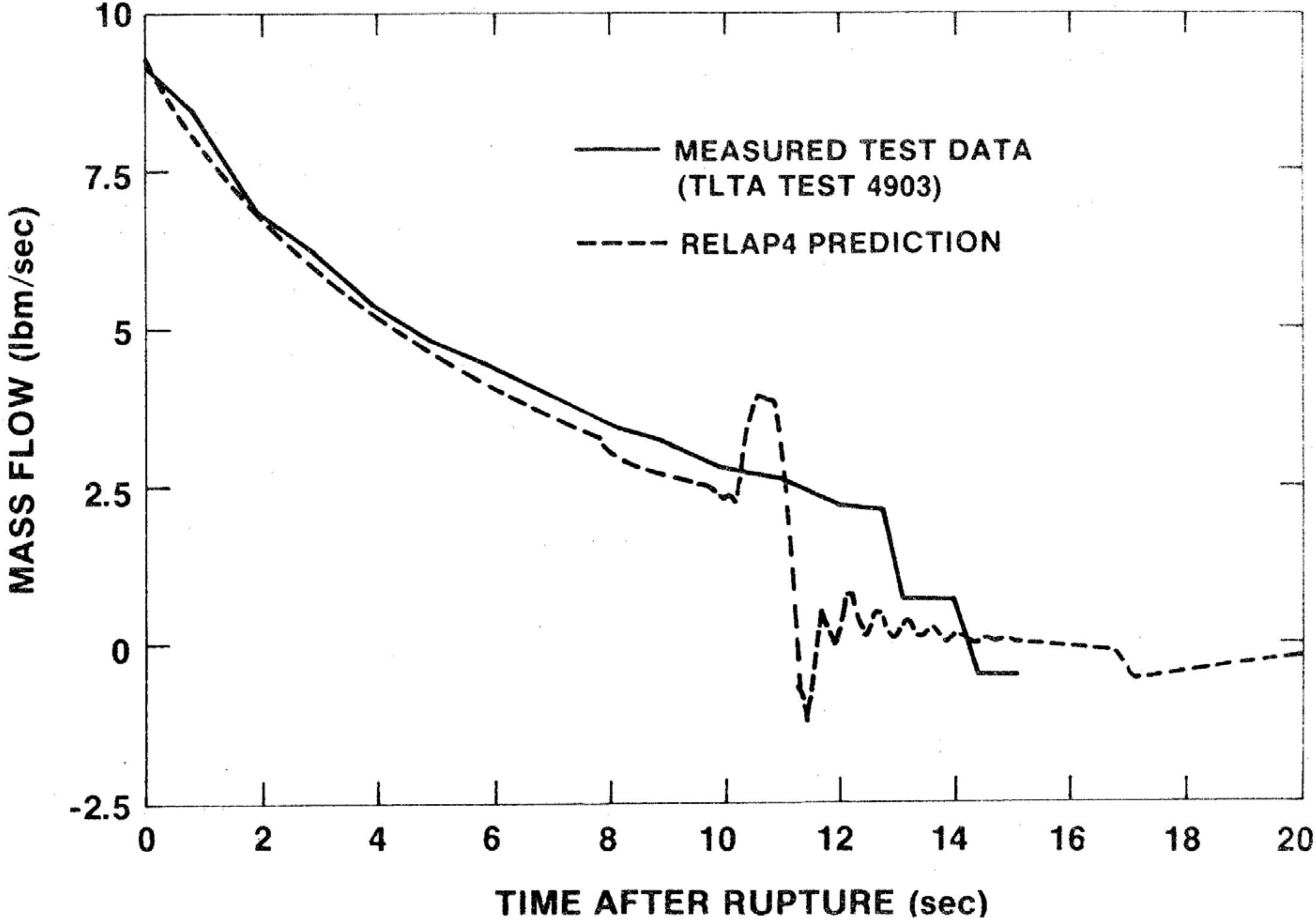
# STEAM DOME PRESSURE



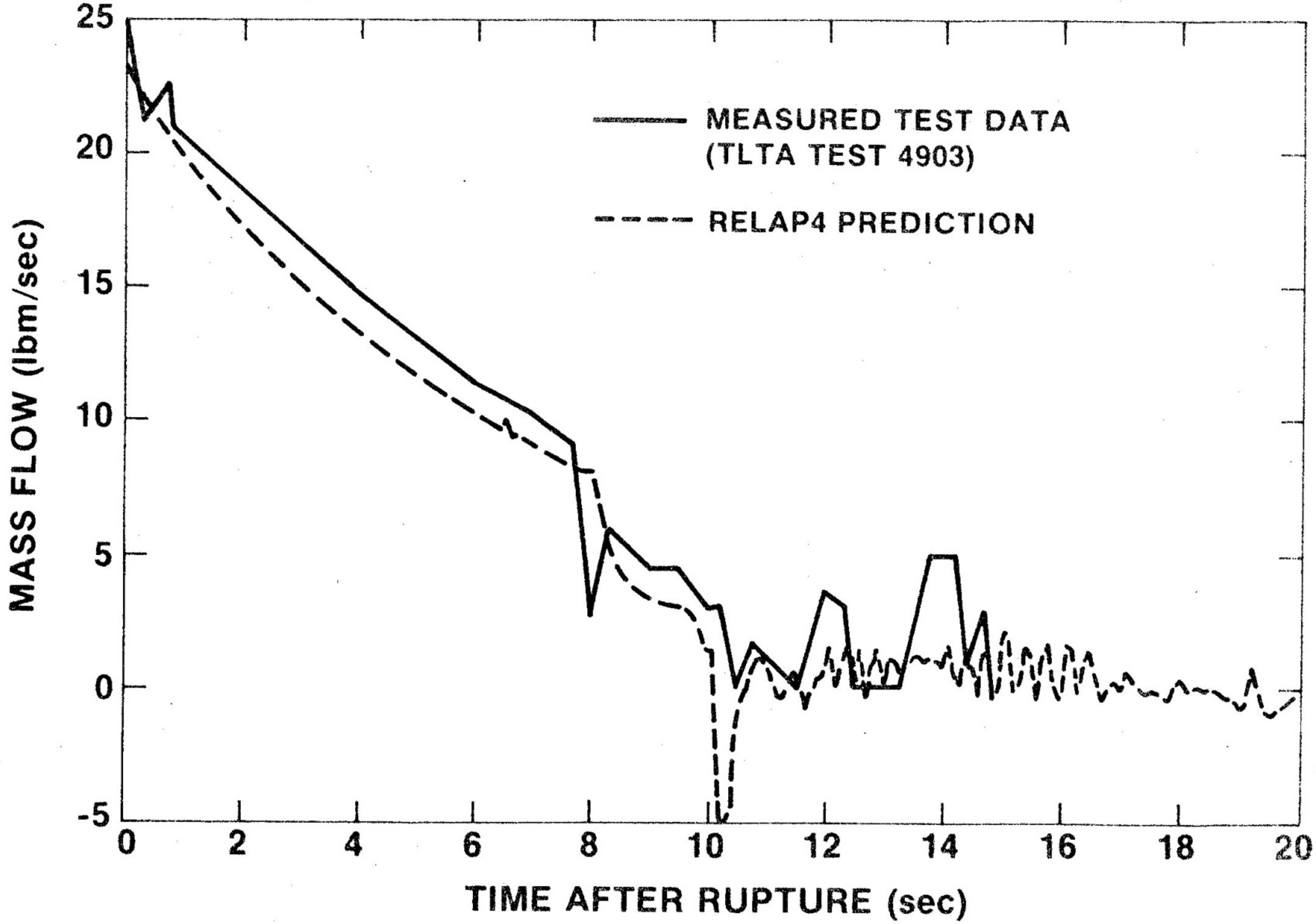
# CORE INLET MASS FLOW RATE



# DRIVE PUMP NO. 1 MASS FLOW RATE

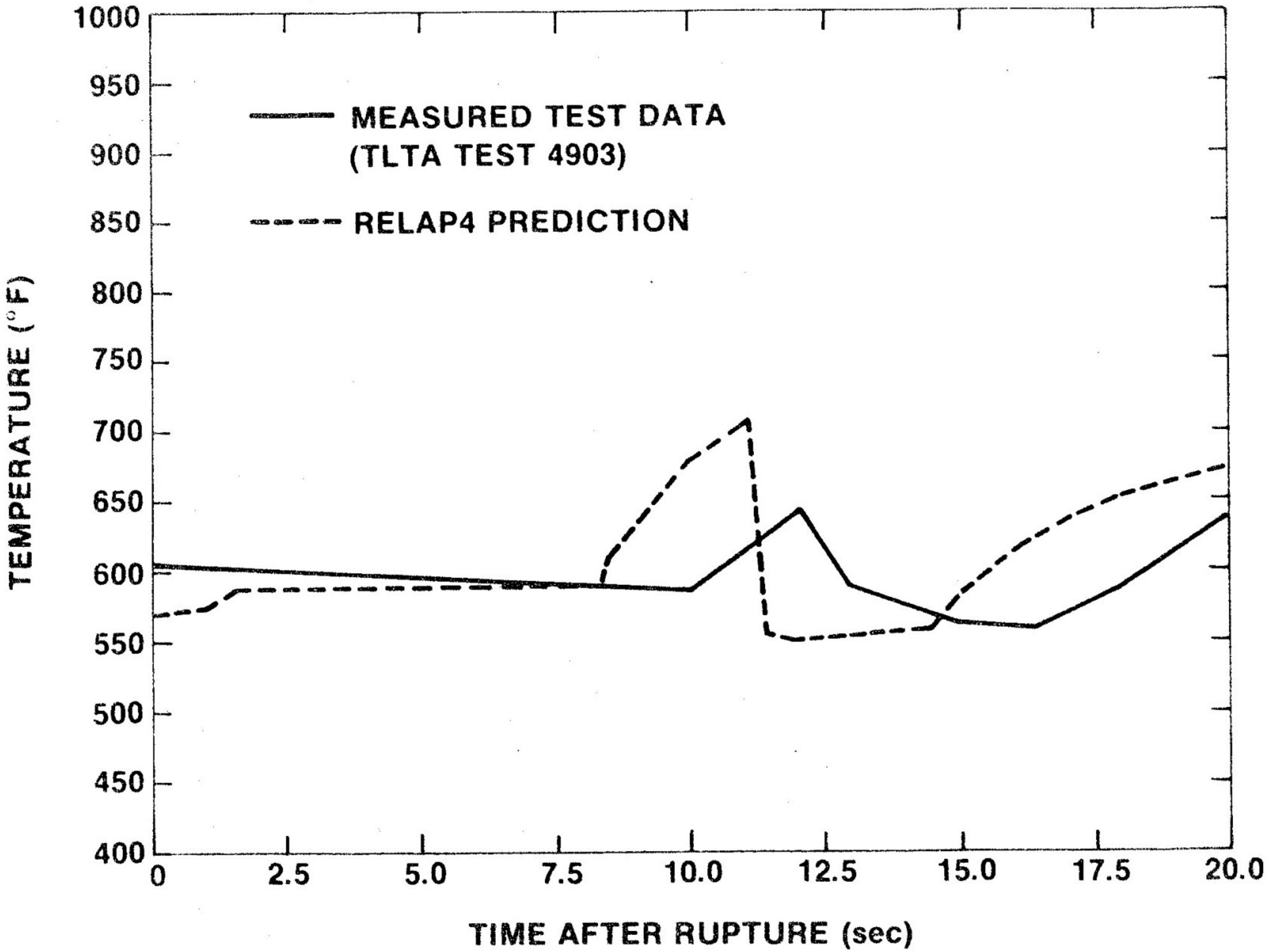


# INTACT LOOP JET PUMP EXIT MASS FLOW RATE



# CLADDING TEMPERATURE

## 118" ELEVATION



# *Summary*

- Heat transfer in BWR fuel channels can be evaluated using approaches based on convective heat transfer based experimental data.
- Heat flux models exist for all heat transfer regimes. These are complicated correlations based on experiments.
- Pressure drops due to two phase flow are greater than those found for single phase flow.
- Fluid Flow Transient and LOCA Situations are evaluated using Large Computer Programs such as RELAP5, TRAC and TRACE

# Important Links

- Some good course material two-phase flow to review - http://www2.et.lut.fi/ttd/studies.html
- Basic Nuclear Energy - http://www.nrc.gov/reading-rm/basic-ref/students.html
- Basic BWR - http://www.nrc.gov/reading-rm/basic-ref/teachers/03.pdf

# References


- **Wallis G.B., *One-dimensional two-phase flow*, McGraw-Hill Book Company, New York (1969)**
- **Lahey R.T., Moody F.J., *The thermal-hydraulics of a boiling water nuclear reactor*, 2nd Ed., American Nuclear Society (1993).**
- **N. Todreas and M Kazimi, ”Nuclear Systems I – Thermal Hydraulic Fundamentals”,Taylor & Francis”, (1993).**
- **N. Todreas and M Kazimi, ”Nuclear Systems II – Elements of Thermal Hydraulic Design”,Hemishere Publishing Corporation, (1990).**
- **Bird, R.B., Steward, W.E., and Lightfoot, E.N., ”Transport Pheomena”, New York: Wiley, (1960)**
- **ASME Steam Tables**
- **L.S. Tong & Joel Wiesman, ”Thermal Anhalu=ysis of Pressurized Water Reactors”, third edition, American Nuclear Society, La Grange Park, Il, (1996)**

# References cont)


- M.M. El-Wakil, “Nuclear Energy Conversion”, American Nuclear Society, La Grange Park, Il, Third Printing, (January 1982)