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 reaction system, derivation of $k_{eff}$

Low Power Reactor Dynamics
- Describes the $\sigma$-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
BWR Flow Paths

- Steam Dryers
- Steam Separators
- Main Steam Flow to Turbine
- Main Feed Flow from Turbine
- Driving Flow
- Core
- Jet Pump
- Recirculation Pump
- Valve (Typical)
PWR 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:
  \[
  \chi = \frac{\text{mass of steam}}{\text{total mass of steam} + \text{liquid}}
  \]
- In static (non-flowing system) void fraction: \( \alpha \) is defined:
  \[
  \alpha = \frac{\text{volume of steam in mixture}}{\text{total volume of steam} + \text{liquid}}
  \]
- Void fraction can be expressed in terms of steam quality and specific volumes (from *Steam Tables*) as follows:
  \[
  \alpha = \frac{\chi v_g}{v_g + \chi v_{fg}} = \frac{1}{1 + [(1 - \chi)/\chi] v_f/v_g}
  \]
  Where:
  - \( v_g \) is specific volume of steam in \( \text{ft}^3/\text{lb-m} \)
  - \( v_f \) is specific volume of liquid in \( \text{ft}^3/\text{lb-m} \)
  - \( v_{fg} = v_g - v_f \) is difference in specific volumes

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

\[
S = \frac{V_g}{V_f}
\]

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: $W \text{ (lb-m/sec)}$
- Steam flow rate is: $\chi W$
- Liquid flow rate is: $(1- \chi)W$
- Phase volumetric velocity of steam is:
  $$V_g = v_g \frac{W \chi}{A_g}$$
  - where: $A_g$ is relative cross sectional area of steam in two phase column
- Phase volumetric velocity of liquid is:
  $$V_f = v_f \frac{W(1- \chi)}{A_f}$$
Definition of Slip in Terms of Steam Quality and Void Fraction

- Slip: \( S = \frac{V_g}{V_f} \)
- Slip defined in terms of steam quality by combining:
  \[
  V_g = v_g \frac{W \chi}{A_g} \\
  V_f = v_f \frac{W(1-\chi)}{A_f}
  \]
- This yields:
  \[
  S = \left(\frac{\chi}{1-\chi}\right)\left(\frac{A_f}{A_g}\right)\left(\frac{v_g}{v_f}\right)
  \]
- Noting that in small slice of column, ratio of steam to total mixture is: \( \alpha = \frac{A_g}{A_g + A_f} \), rearranging this:
  \[
  \left(\frac{A_f}{A_g}\right) = \frac{1 - \alpha}{\alpha}
  \]
- Slip can then be expressed:
  \[
  S = \left(\frac{\chi}{1-\chi}\right)[(1 - \alpha)/\alpha] \left(\frac{v_g}{v_f}\right)
  \]
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:

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

• 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

- Bubbly
- Wavy
- Plug
- Slug
- Stratified
- Annular
Flow Patterns in Vertical Flow

Bubbly

Slug

Churn

Wisy-annular

Annular
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)
  \[ h_{\text{film}} = \left( \frac{k}{D_h} \right) 0.023 \, \Pr^{0.4} \, \Re^{0.8} \]
• Above subcooled region – different \( h_{\text{film}} \) model would apply
• ALSO: When modeling cooling on tube:
  \[ h_{\text{film}} = \left( \frac{k}{D_h} \right) 0.023 \, \Pr^{0.3} \, \Re^{0.8} \]
• Example problem: heat transfer in subcooled region
• Assume: \( 1000 \, psia, \ T_{in} = 515^\circ F, \ V_f = 6.8 \, ft./sec. \)
• Use standard dimensions of GE 8x8 Fuel Bundle
• Calculate \( h_{\text{film}} \)
Example $h_{\text{film}}$ Calculation for 8x8 BWR Fuel

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

Fuel Assembly Pitch in cm.: $p := 1.62$

Clad Outer Radius in cm: $R_c := \frac{1.25}{2}$

Hydraulic diameter in cm: $D_e := 2R_c \left[ \frac{4}{\pi} \left( \frac{p}{2R_c} \right)^2 - 1 \right]$, $D_e = 1.423$

Hydraulic diameter in in: $D_h := D_e \cdot 0.394$, $D_h = 0.561$

Conductivity of Water in BTU/ft.hr.F: $k_w := 0.3435$ For 515 F, 1000 psi $k_w = 0.305$ BTU/hr ft F

Prandtl Number:
515 F, 1000 psi $Pr := 0.87$

Kinematic Viscosity in ft$^2$/sec. 515F, 1000 psi $\nu := 1.38 \cdot 10^{-6}$ Taken from Table 4.4-1 River Bend FSAR

Flow velocity in ft/sec: $v_f := 6.82$

Reynolds Number (lengths computed in ft):

$Re := \left( \frac{D_h}{12} \right) \frac{v_f}{\nu}$, $Re = 2.309 \cdot 10^5$

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:

$hf_{\text{film}} := \frac{k_w}{D_h} \cdot Nu$, $hf_{\text{film}} = 260.309$
Nucleate Boiling

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

\[ q_{THOM} = 0.05358 \times \exp\left(\frac{P}{630}\right) \times (T_c - T_{sat}(P))^2 \]

- \( q_{THOM} \) 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:

\[ q''_{\text{CHF}} = \frac{1}{0.0036} \left( \frac{A F_A - \chi_{\text{in}}}{C F_c F_g \frac{F_{\text{nu}}}{G}} + \frac{h - h_{\text{in}}}{0.0036 \cdot q'' h_{\text{tg}}} \right) \]

with:

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

\[ C = 1.6151 \cdot P_r^{1.4090} \cdot (0.0036 \cdot G)^{(-0.4643 - 2.0749 P_r)} \]

and:

- \( q''_{\text{CHF}} \) = critical heat flux (Btu/s/ft²),
- \( q' \) = local heat flux (Btu/s/ft²),
- \( G \) = coolant mass flux (lbm/s/ft²),
- \( P_r \) = critical pressure ratio (= system reference pressure/critical pressure),
- \( h \) = local enthalpy (Btu/lbm),
- \( h_{\text{in}} \) = inlet enthalpy (Btu/lbm),
- \( h_{\text{tg}} \) = vaporization enthalpy (Btu/lbm).

\( F_A, F_C, F_g \) and \( F_{\text{nu}} \) 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:

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

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

\[ F_g = 1.3 - 0.3 C_g \]

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

\[ F_{\text{nu}} = 1.0 + \frac{Y - 1}{1 + 0.0036 \cdot G} \]

\[ Y = \int q'(X) \, dX \]

\[ Y = \frac{q''(X) \, X}{q''(X)} \]

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:

\[ q''_{TB} = \frac{1}{2} \sqrt{T_w - T_{sat}} \left( T_w - T_{sat} \right) \]

where:

\[ C_1 = \frac{q''_{CHF} - q''_{FB}}{T_{CHF} - T_{sat}} \]

\( q''_{CHF} \) = 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

\[ q_{FB}^* = H_{FB} (T_w - T_{sat}) \]

\[ H_{FB} = 0.052 \frac{k_g}{D_h} Re_{hom}^{0.688} Pr_f^{1.26} \frac{1}{\gamma^{1.06}} \]

\[ \gamma = 1.0 - 0.1 (1 - x) \frac{\rho_f}{\rho_g} - 1 \]

\[ Pr_f = \frac{C_{pv} \mu_v}{k_v} \]

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

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/ft³),
- \( \rho_g \) = saturated vapor density (lbm/ft³),
- \( \mu_g \) = dynamic viscosity of saturated vapor (lbm/s/ft),
- \( G \) = coolant mass flux (lbm/s/ft²),
- \( C_{pv} \) = 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 $T_{\text{sat}}(P)$ at $H_{\text{BOIL}}$:
  \[
  T(z, P) = T_{\text{in}}(P) + \int_{-H_{\text{eff}}/2}^{z} \frac{q(z)}{W C_{p}(P)} \, dz
  \]
  \[
  h(z, P) = h_{\text{in}}(P) + \int_{-H_{\text{eff}}/2}^{z} \frac{q(z)}{W} \, dz
  \]
- Above $H_{\text{BOIL}}$ further heat addition only increases steam content, not temperature
- Enthalpy rise: $h(z, P) = h_{\text{sat}}(P) + \chi(z) h_{fg}(P)$
  \[
  \text{where: } \chi(z) = \int_{H_{\text{BOIL}}}^{z} \frac{q(z)}{W h_{fg}(P)} \, dz
  \]
Simulation of Uniform Linear Power Density

Channel inlet flow in lb-m/sec.:
Reactor Inlet enthalpy: assuming 1000 psia, T=515 F
Saturated liquid enthalpy: assuming 1000 psia, Tsat=544.58 F
Vaporization enthalpy: assuming 1000 psia, Tsat=544.58 F
Peak axial power in BTU/sec.ft
Axial Power Distribution in BTU/sec ft:
Routine to generate Subcooled Axial enthalpy profile:
\[ f(z) = \text{hin} + \frac{1}{W} \int_{-\text{Heff}}^{z} qax(z) \, dz \]
\[ h(z) = \begin{cases} f(z) & \text{if } f(z) \leq \text{hf} \\ \text{hf} & \text{otherwise} \end{cases} \]
Routine to find the point where boiling starts:
\[ F(z) = f(z) - \text{hf} \]
\[ \text{Hboil} = \text{Heff} + \frac{\text{Heff}}{2} \]
\[ \text{Height above bottom of core: } \text{Hbot} = \frac{\text{Heff}}{2} + \text{Hboil} \]
\[ \text{Hbct} = 3.479 \]
Routine to generate the Axial Quality Profile and Axial Enthalpy:
\[ \chi(z) = \begin{cases} 0 & \text{if } f(z) \leq \text{hf} \\ \int_{Hboil}^{z} \frac{qax(z)}{W \cdot \text{hfg}} \, dz & \text{if } f(z) > \text{hf} \end{cases} \]
\[ h(z) = \begin{cases} f(z) & \text{if } f(z) \leq \text{hf} \\ \text{hf} + \chi(z) \cdot \text{hfg} & \text{otherwise} \end{cases} \]
Simulation of Cosine Linear Power Density

Channel inlet flow in lb-m/sec:

Reactor Inlet enthalpy:
assuming 1000 psia, T=815 F

Saturated liquid enthalpy:
assuming 1000 psia, Tsat=544.58 F

Vaporization enthalpy:
assuming 1000 psia, Tsat=544.58 F

Peak axial power in BTU/sec-ft

Axial Power Distribution in BTU/sec-ft:

Routine to generate Subcooled Axial enthalpy profile:

\[ f(z) := \sin \left( \frac{z}{H_{eff}} \right) \]

\[ q_{ax}(z) := \frac{W}{0.6} \int_{-H_{eff}}^{h_{f}} q_{ax}(z) \, dz \]

\[ h(z) := \begin{cases} f(z) & \text{if } f(z) \leq h_{f} \\ h_{f} & \text{otherwise} \end{cases} \]

Routine to find the point where boiling starts:

\[ F(z) := f(z) - h_{f} \]

\[ H_{boil} = \sqrt{\frac{F(z) - h_{f}}{2}} \]

Height above bottom of core:

\[ H_{bot} = \frac{H_{eff}}{2} + H_{boil} \]

\[ H_{bot} = 3.848 \]

Routine to generate the Axial Quality Profile and Axial Enthalpy:

\[ \chi(z) := \begin{cases} 0 & \text{if } f(z) \leq h_{f} \\ \int_{z}^{H_{boil}} \frac{q_{ax}(z)}{W \cdot h_{f}} \, dz & \text{if } f(z) > h_{f} \end{cases} \]

\[ h(z) := \begin{cases} f(z) & \text{if } f(z) \leq h_{f} \\ h_{f} + \chi(z) \cdot h_{fg} & \text{otherwise} \end{cases} \]
Simulation of Cosine Linear Power Density

BWR Fuel Data:
Fuel Rod Length in ft.:

Fuel Pellet Radius in in.:

Uranium Dioxide thermal conductivity in BTU/hr.ft°F:

Gap Conductance in BTU/hr.ft²F:

Zircaloy Cladding outer radius in in.:

Zircaloy thermal conductivity in BTU/hr.ft°F

\[ T_{\text{in}} = 515 \quad C_p = 1.17 \]

Coolant Temperature vs. height distribution:

\[
T(z) = \begin{cases} 
T_{\text{in}} + \frac{z}{w \cdot C_p} \int_0^z q_{\text{in}}(z) \, dz & \text{if } z < H_{\text{boil}} \\
T_{\text{sat}} & \text{otherwise}
\end{cases}
\]

Clad Surface Temperature vs. height distribution:

\[
T_{\text{clad}}(z) = T(z) + \frac{q_{\text{in}}(z) \cdot 3600}{2 \cdot \pi} \left( \ln \left( \frac{R_c}{R_c} \right) + \frac{1}{R_c \cdot h_{\text{film}}} \right)
\]

Heat Flux calculated based upon temperature difference:

\[
q_f(z) = \frac{2 \cdot \pi \cdot R_c \cdot h_{\text{film}} (T_{\text{clad}}(z) - T(z))}{3600}
\]

Fuel Centerline Temperature vs. height distribution:

\[
T_f(z) = T(z) + \frac{q_{\text{in}}(z) \cdot 3600}{2 \cdot \pi} \left( \frac{1}{2 \cdot k_{\text{fuel}}} + \frac{1}{R_c \cdot h_{\text{gap}}} + \ln \left( \frac{R_c}{R_o} \right) + \frac{1}{R_c \cdot h_{\text{film}}} \right)
\]

Graphs showing coolant, clad, surface, and fuel temperatures as functions of core height.
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

- Channel Enthalpy vs Height
  - $W = 0.3\ \text{lb-m/sec.}$  $H_{bot} = 2.66\ \text{ft.}$
  - $W = 0.6\ \text{lb-m/sec.}$  $H_{bot} = 3.848\ \text{ft.}$
  - $W = 0.9\ \text{lb-m/sec.}$  $H_{bot} = 4.825\ \text{ft.}$
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:

\[
\Delta P_{\text{friction}} = \left( \frac{fL}{D_h} \right) \frac{\rho \nu^2}{2} + \sum_i \left( K_i \frac{\rho \nu_i^2}{2} \right)
\]

• 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:

\[
\bar{R} = \frac{\Delta P_{2\phi}}{\Delta P_{1\phi}}
\]

• -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: \( \Delta P_{2\phi} = \bar{R} \times \Delta P_{1\phi} \)
- Where:

![Diagram](image_url)
Figure 3-4. Ratio of Two-Phase to Single-Phase Flow Resistance Versus Quality and Pressure
BWR Fuel Channel Pressure Drop
(Pressure Drops Due to Grid Spacers, Inlet/Outlet Geometry Would Need to be Added!)

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

Calculation of Pressure Drop Across Fuel Bundle:

\[ f = \frac{0.184}{Re^{0.2}} \]

Hydraulic Diameter in in.: \( Dh = 0.56 \)

Dynamic viscosity in \( \text{ft}^2/\text{sec} \):
\[ \mu = 1.38 \times 10^{-6} \]

Flow velocity in ft/sec.: \( V_f = 6.82 \)

Density in lb/ft\(^3\):
\[ \rho = 46.64 \]

Height of subcooled length in ft.: \( H_{bot} = 3.848 \)

Conversion constant from lbm to lbf:
\[ gc = 4.17 \times 10^8 \]

Presssure drop in subcooled portion of fuel channel in psi:
\[ \Delta P_{sec} = \left( \frac{fH_{bot}}{Dh} \right) \frac{\rho(V_f^2)}{2gc144} \]

Length of fuel channel in ft. above boiling point:
\[ L = Heff - H_{bot} \]

Exit Quality:
\[ \chi_{exit} = \chi \left( \frac{Heff}{2} \right) \]

Pressure drop in saturated portion of fuel channel in psi:
\[ \Delta P_{sat} = R \left[ \left( \frac{fL}{Dh} \right) \frac{\rho(V_f^2)}{2gc144} \right] \]
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:

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

- 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 \( \chi_{\text{exit}} \)

\[
\Delta p_a = \left[ (1 - \chi_{\text{exit}}) m_{\text{tot}} V_f + \chi_{\text{exit}} m_{\text{tot}} V_g - m_{\text{tot}} V_{\text{in}} \right]
= \left( m_{\text{tot}}/A_c \right) \left[ (1 - \chi_{\text{exit}}) V_f + \chi_{\text{exit}} V_g - V_{\text{in}} \right]
\]

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

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

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

\[
V_f = m_f v_f / A_f = \left( 1 - \chi_{\text{exit}} \right) m_{\text{tot}} v_f / A_f = \left( 1 - \chi_{\text{exit}} \right) m_{\text{tot}} v_f / \left( 1 - \alpha_{\text{exit}} \right) A_f
= \left( 1 - \chi_{\text{exit}} \right) G v_f / \left( 1 - \alpha_{\text{exit}} \right)
\]
Acceleration Pressure Drop

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

\[ V_g = \chi_{exit} G v_g / \alpha_{exit} \quad V_{in} = G v_{in} \approx G v_f \]

• Substituting these – acceleration pressure drop becomes:

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

\[ = G^2 v_f \left[ \frac{(1- \chi_{exit})^2}{(1- \alpha_{exit})} + \chi_{exit}^2 v_g / v_f \alpha_{exit} - 1 \right] \]

• An overall acceleration Multiplier \( R_a \) can now be defined:

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

• Acceleration pressure drops would be calculated:

\[ \Delta p_a = G^2 R_a \]
Example Acceleration Pressure Drop Calculation

Assume 1000 psi, \(T_{in} = 515 \degree F, \ x_{exit} = 0.199, \ S = 2.5\)

Specific volume of liquid in ft³/lb: \(v_f = 0.02159\)
Specific volume of steam in ft³/lb \(v_g = 0.44596\)

Definition of Void Fraction in terms of Steam Quality and Slip Ratio:
\[\alpha(x,S) := \frac{1}{1 + \left[\frac{1 - x}{x} \frac{vf}{v_g} S\right]}\]
\[\alpha(0.199,2.5) = 0.672\]

![Void Fraction vs Steam Quality graph]

**Acceleration Pressure Drop Multiplier**

Acceleration Multiplier definition:
\[Ra(\alpha, x) = \frac{(1 - x)^2}{(1 - \alpha)} + \frac{x^2 v_g}{\alpha \cdot vf} - 1\]
\[Ra(0.672,0.199) = 0.047\]

Inlet fluid velocity in ft/hr:
\(V_f = 6.82 \times 3600\)

Inlet fluid density in lb-m/ft³
at 1000 psi, 515 F:
\(\rho_{in} = 48.2393\)

Mass flow rate per unit cross-sectional area in lb-m/ft²:
\(G = \rho_{in} \cdot V_f\)
\(G = 1.184 \times 10^6\)

Conversion constant:
\(g_c = 4.17 \times 10^8\)

Definition of acceleration pressure drop in psi
\[\Delta Pa := \frac{Ra(0.672,0.199) G^2}{144 \cdot g_c}\]
\[\Delta Pa = 1.096\]
2-Phase Expansion and Contraction Losses

- Recall in treatment of single phase pressure drops:
  \[ \Delta P = K \rho V \text{in}^2 / 2 \]

- 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.
Figure 3-5. Pressure Drop Components in a Typical Reactor Channel
7. Behavior of System During Accident
BWR Break Water Level

[Diagram of a BWR break water level system with labels for normal water level, steam separators, steam separation distribution plenum, pipe break, recirculation pump, lower plenum, active core, and "collapsed" water level after break.]
DRIVE PUMP NO. 1 MASS FLOW RATE

- MEASURED TEST DATA (TLTA TEST 4903)
- RELAP4 PREDICTION

MASS FLOW (lbm/sec)

TIME AFTER RUPTURE (sec)
INTACT LOOP JET PUMP EXIT
MASS FLOW RATE

![Graph showing mass flow rate over time after rupture. The graph compares measured test data (TLTA TEST 4903) with RELAP4 prediction.](image)

- **MEASURED TEST DATA**
  - TLTA TEST 4903

- **RELAP4 PREDICTION**

**Y-axis:** MASS FLOW (lbm/sec)

**X-axis:** TIME AFTER RUPTURE (sec)
CLADDING TEMPERATURE
118" ELEVATION

- MEASURED TEST DATA
  (TLTA TEST 4903)

- RELAP4 PREDICTION

TEMPERATURE (°F)

TIME AFTER RUPTURE (sec)
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

• ASME Steam Tables
References cont)