TRACE Modeling Issues and Guidelines

Information Systems Laboratories, Inc.

Presented at

Nuclear Regulatory Commission
TRACE/SNAP User Workshop
Idaho Falls, Idaho
September 30 – October 3, 2014

Objectives

- Familiarize students with proven modeling techniques and guidelines
- Introduce various important modeling considerations

Outline

- Discussion of LOCA modeling and analysis
- L/D modeling considerations for one-dimensional flow paths
- PWR loop seal nodalization
- Frictional pressure drop modeling
- Loop elevation closure
- Steam generator modeling
- BREAK component geometry input considerations
- Discussion of the reflood process

Discussion of LOCA Modeling and Analysis

Loss of Coolant Accident (LOCA) analyses are performed using plant models with thermal hydraulic systems codes, such as TRACE and RELAP5

10 CFR 50.46 requires an analysis be performed with an approved evaluation model demonstrating that the performance of the ECCS satisfies the following criteria

- The calculated maximum fuel element cladding temperature does not exceed 2,200 ^{o}F (1,200 ^{o}C) at any location in the core

- Fuel cladding oxidation does not exceed 17% of the total cladding thickness at any location in the core

- The total amount of hydrogen generated by chemical reaction of fuel cladding with water and steam does not exceed 1% of the total hypothetical hydrogen that would be generated if all of the cladding in the core were to react

- Changes in core geometry are such that the core remains amenable to cooling

- After successful initial operation of the ECCS, the core remains cool and the decay heat is removed for an extended period of time

Discussion of LOCA Modeling and Analysis

TRACE LOCA models are developed in order to calculate the transient fuel cladding temperature behavior

Given the reactor configurations and safety system designs, if the peak cladding temperature requirement is satisfied then generally the fuel cladding oxidation and core geometry requirements are also satisfied

Long-term cooling is typically analyzed using simple models for evaluating the process of switchover to sump injection and the effectiveness of boron in solution for sustained control of the core reactivity

A background discussion of TRACE modeling for LOCAs is provided in this presentation because:

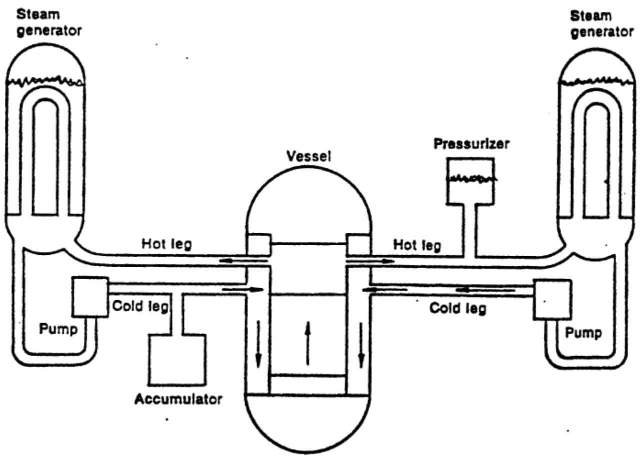
(1) A majority of the analyses performed using TRACE is related to LOCAs, and

(2) A TRACE exercise later in this workshop includes the modeling, calculation and analysis of a PWR small break LOCA

For additional information on the topics discussed here see the workshop materials from “Analysis of Small Break Loss-of-Coolant Accidents in Conventional Pressurized Water Reactors,” Information Systems Laboratories, Inc., September 20-22, 2011, Bethesda, Maryland.

Overview and Terminology of Conventional PWR Systems

Westinghouse (W), Combustion Engineering (CE) and Babcock and Wilcox (B&W)



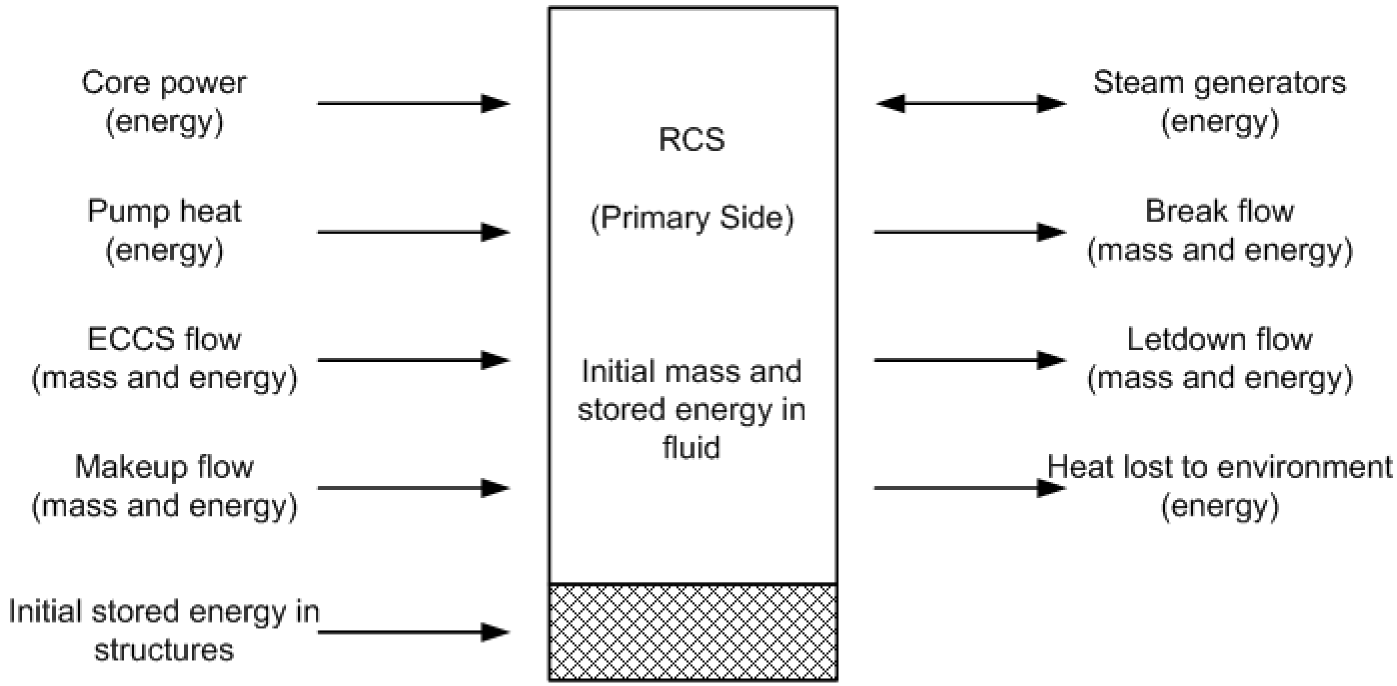
Schematic of Reactor Coolant System (RCS) for Two Loops of a Westinghouse PWR

PWR LOCA analysis focuses on core cooling and the thermal-hydraulic response of the RCS:

Reactor vessel containing the core
Hot leg piping from reactor vessel to steam generators (SGs)
Pressurizer and surge line
SG primary coolant system
SG secondary coolant system, and the feedwater and steam systems
Crossover leg piping (also called the “loop seal”) between SG and reactor coolant pump (RCP)
Cold leg piping from RCP to the reactor vessel
Emergency core cooling systems (ECCS)
- Charging pumps – CVCS (W, CE), makeup (B&W)
- High pressure safety injection pumps (HPSI, HPI, HHSI)
- Accumulators (W), Safety Injection Tanks (CE), Core Flood Tanks (B&W)
- Low pressure injection pumps (LPSI, LPI, LHSI)

Mass and Energy Balances During LOCAs
Consider the RCS Fluid to be a Large Control Volume

- The RCS fluid can be thought of as a single large control volume
- Mass and energy can enter or leave the control volume as shown below



Addition of Mass and Energy to the RCS

- Core power (energy)
  - Fission power prior to reactor trip
  - Decay power, 3.4% at 1 minute after reactor trip, 1.3% at 1 hour after reactor trip
- Pump heat addition (energy)
  - Typically around 12 to 15 MW total for a PWR
  - Goes away when RCPs are tripped
- ECCS flow (mass and energy)
  - Injections from HPI, accumulator and LPI systems
- Makeup flow (mass and energy)
  - Used for RCP seal injection and pressurizer level control during normal operation
  - May or may not be safety grade (allowing it to be credited in LOCA analyses)
- Initial stored energy of structures (energy)
  - Fuel rods, maximum fuel centerline temperature ~3,500 ^oF (~1927 ^oC)
  - Piping and vessel walls, ~580 ^oF (~304 ^oC)
- Steam generators (energy)
  - Generally, SGs remove heat, however the SGs can transfer heat into a depressurized and cooled RCS

Removal of Mass and Energy from the RCS

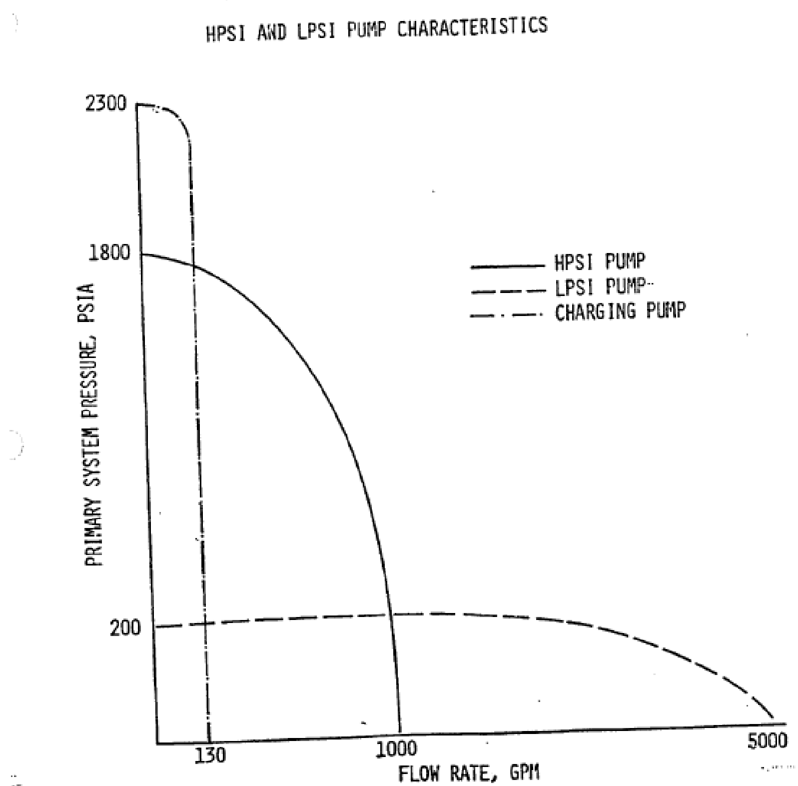
- Steam generators (energy)
  - Remove energy until the tubes void and loop natural circulation stops
  - Cause and effect, steam generator cooling/coolant loop natural circulation flow
  - After loop natural circulation ends, SGs can still transfer heat from the RCS via <u>reflux cooling</u> (countercurrent flow in hot legs, steam from core condensed inside SG tubes, condensate drains downward and returns to core via reverse flow through the hot legs, process limited by CCFL effects)
- Break flow (mass and energy)
  - Density of liquid is high, but its energy is low (liquid break flow significantly depletes RCS inventory but does not effectively remove RCS heat)
  - Density of steam is low, but its energy is high (steam break flow depletes RCS inventory very little but removes RCS heat very effectively, promoting RCS depressurization and leading to increased ECCS flows)
- Letdown flow (mass and energy)
  - Generally a small flow rate and typically not modeled for LOCAs (system is isolated at time of safety injection signal)
- RCP seal leakage (mass and energy)
  - Generally a small flow rate (~20 gpm per pump) and typically not modeled for LOCAs
- Heat lost to the containment (energy)
  - Typically ~8 MW, conservatively not modeled for LOCAs (all RCS exterior vessel and piping walls are typically assumed perfectly insulated)

Safety Injection System Flows

The charging, HPSI and LPSI pumps are typically of centrifugal design (except for positive displacement charging pumps in some plants)

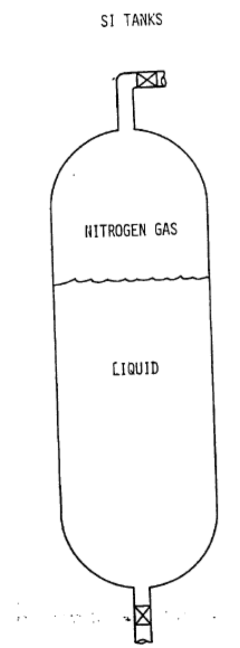
The safety injection flow is delivered into the cold legs at a rate that is a function of RCS pressure

No flow is delivered at RCS pressures above the pump shut-off heads



Safety Injection System Flows

The accumulator injection rate is a function of the nitrogen bubble expansion process, the injection line piping flow loss, and the RCS cold leg pressure



LOCA Behavior Bottom Line: The safety injection system flows increase as the RCS pressure declines. The RCS must depressurize sufficiently for the safety systems to replenish the coolant lost out the break. As the break size increases, the RCS pressure declines more, which increases the injection flow but also increases the rate of coolant loss out the break. Therefore, the RCS behavior and the peak cladding temperature response are functions of break size.

Characterization of LOCA Break Sizes

A LOCA results from a rupture in the primary RCS pressure boundary, for which the rate of coolant inventory loss exceeds the flow capability of the charging/makeup pumps at the plant operating conditions

Smallest LOCA break size typically analyzed is ~0.0005 ft^2, ~0.3-inch equivalent diameter

From the licensing perspective, PWRs must be designed for safety following breaks at any location on the primary RCS and for breaks as large as a double-ended rupture of the main coolant piping

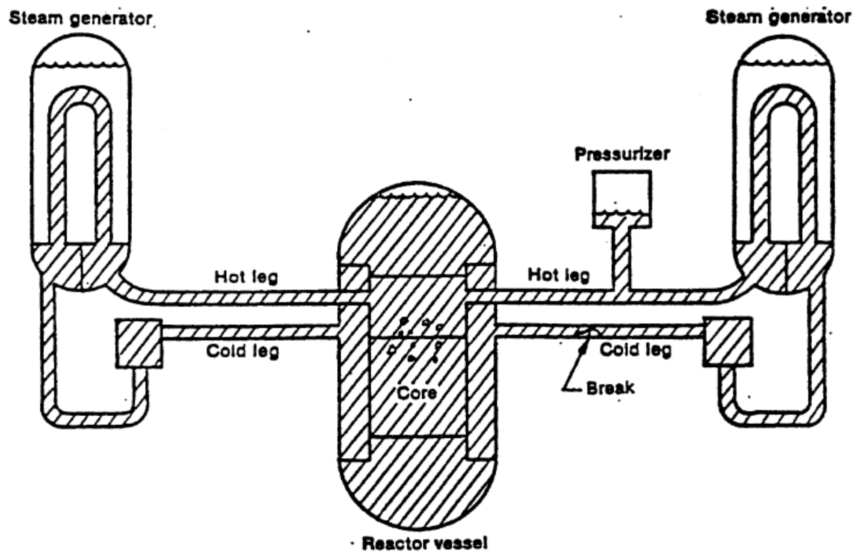
Largest LOCA break size requiring analysis is ~10.0 ft^2, two breaks of ~30.0 inch equivalent diameter

Because of different controlling phenomena and plant systems, LOCAs are subdivided into two categories, large break LOCAs (LBLOCAs) and small break LOCAs (SBLOCAs)

1.0 ft^2 (~13.5-inch) < **LBLOCAs** < 10.0 ft^2 (double-ended 30.0-inch)

0.0005 ft^2 (0.3-inch) < **SBLOCAs** < 1.0 ft^2 (~13.5-inch)

RCS Inventory Behavior for Small Cold Leg Break



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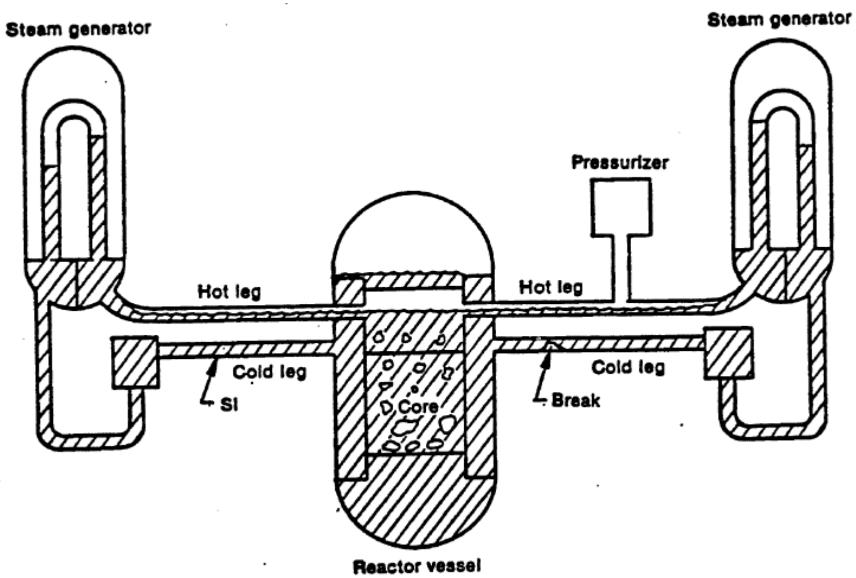
Break Opening

Reactor Trips, RCS is Liquid Filled but Break Flow Rate Exceeds Injection Flow Rate



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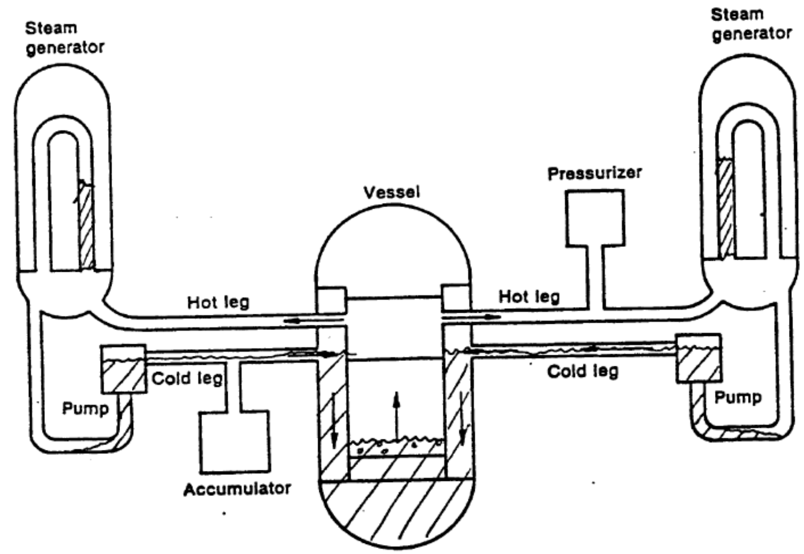
Interruption of Loop Natural Circulation

RCS Draining Results in Voiding at the Tops of SG U Tubes, Breaking the Siphon Paths for the Loop Flow

RCS Inventory Behavior for Small Cold Leg Break



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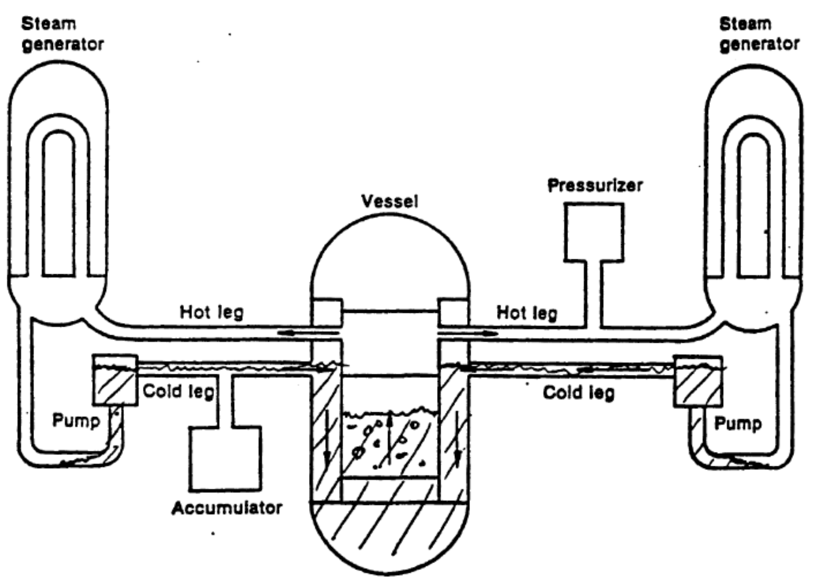
RCS Draining

Steam Cannot Reach Break Because of Water in Loop Seals, Differential Draining of SG Tubes Caused by Reflux Cooling and CCFL Effects Depresses Core Level, Uncovering Top of the Core



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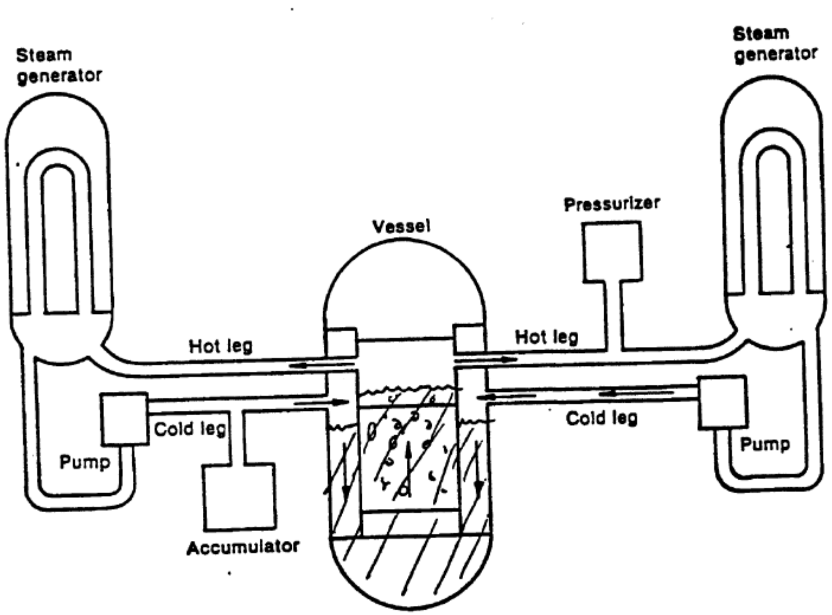
Loop Seal Clearing

RCS Pressure Remains Elevated Until Loop Seals Clear, Permitting Steam to Reach the Break. Loop Seal Clearing Process Also Depresses the Core Level

RCS Inventory Behavior for Small Cold Leg Break



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Post Loop Seal Clearing

Steam Flow Out the Break Depressurizes the RCS, Increased Injection Flows Recover and Cool the Core

RCS conditions are stabilized. Injection mass flow makes up for break mass flow. Core decay heat removed from RCS by two-phase flow out the break.

Effect of SBLOCA Pipe Break Location

- <u>Hot Leg or Surge Line Break</u>
  - Reactor vessel water level drops to hot leg elevation
    - Could be at a higher elevation if the break is in the surge line
  - SG tubes void
    - Natural circulation flow stops
  - Loop seal piping remains filled
  - ECCS flow all goes towards the downcomer, then through the core before exiting at the break
  - Steam generated in the core exits through the break
  - Generally excellent core cooling for hot leg or surge line breaks

Effect of SBLOCA Pipe Break Location

- <u>RCP Crossover Leg (Loop Seal) Break</u>
  - Reactor vessel water level drops to the hot/cold leg elevation
  - SG tubes void
    - Natural circulation flow stops in the intact loops
  - Intact loop crossover leg piping remains filled
  - ECCS flow from the intact loops goes towards the downcomer
  - Some of the ECCS flow from the broken loop goes towards the RCP and some goes towards the downcomer
    - Amount of flow split is break size dependent
  - Break flow from the cold leg side is limited due to the large resistance of reverse flow through the RCP
  - Steam generated in the core flows through the broken loop hot leg and SG towards the break
  - Generally good core cooling for RCP crossover leg breaks

Effect of SBLOCA Pipe Break Location

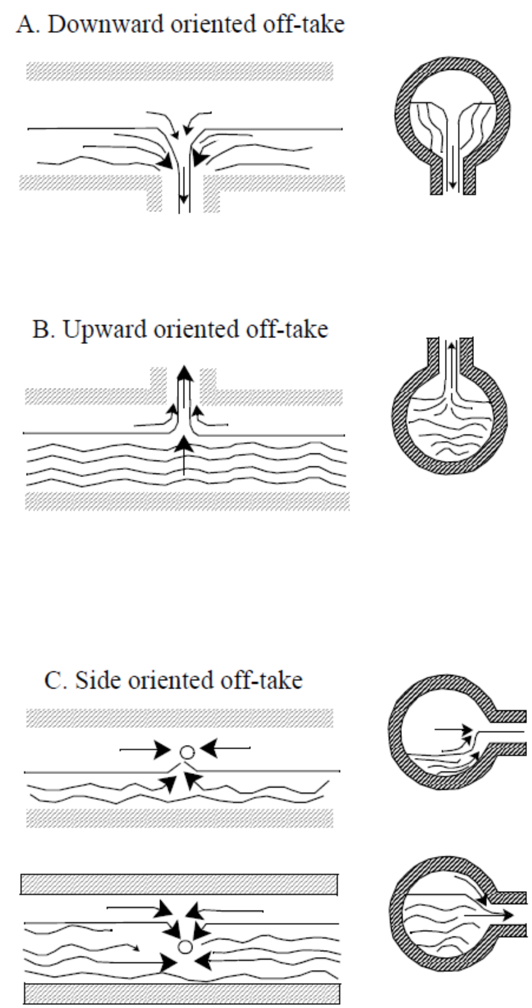
- <u>Cold Leg Break</u>
  - Reactor vessel water level drops to cold leg elevation
  - SG tubes void
    - Natural circulation flow stops
  - Loop seal piping initially remains filled
  - Majority of the ECCS flow in intact loops goes towards the downcomer
    - Some of the flow can go backwards towards the RCP
    - Some of the flow goes around the downcomer and exits the break
  - Some ECCS flow in the broken loop spills out the break
    - For a large double ended break, all the ECCS flow in the broken loop is typically assumed to exit the break

Effect of SBLOCA Pipe Break Location

- <u>Cold leg break (continued)</u>
  - Significantly less ECCS flow available for core cooling as compared with a hot leg break
  - Steam generated in the core raises the pressure in the upper regions of the RPV relative to the remainder of the RCS
    - This pressurization results in core liquid level depression
  - Break flow exceeds injection flow so the loop seal(s) eventually clear
    - Broken loop will clear first
    - Natural circulation two-phase flow will begin in cleared loops
    - RCS pressure declines and ECCS is injected at higher rates
  - **Core cooling is most challenged by cold leg breaks, which are typically most limiting for peak cladding temperature**

Effect of SBLOCA Break Location on the Circumference of the Pipe

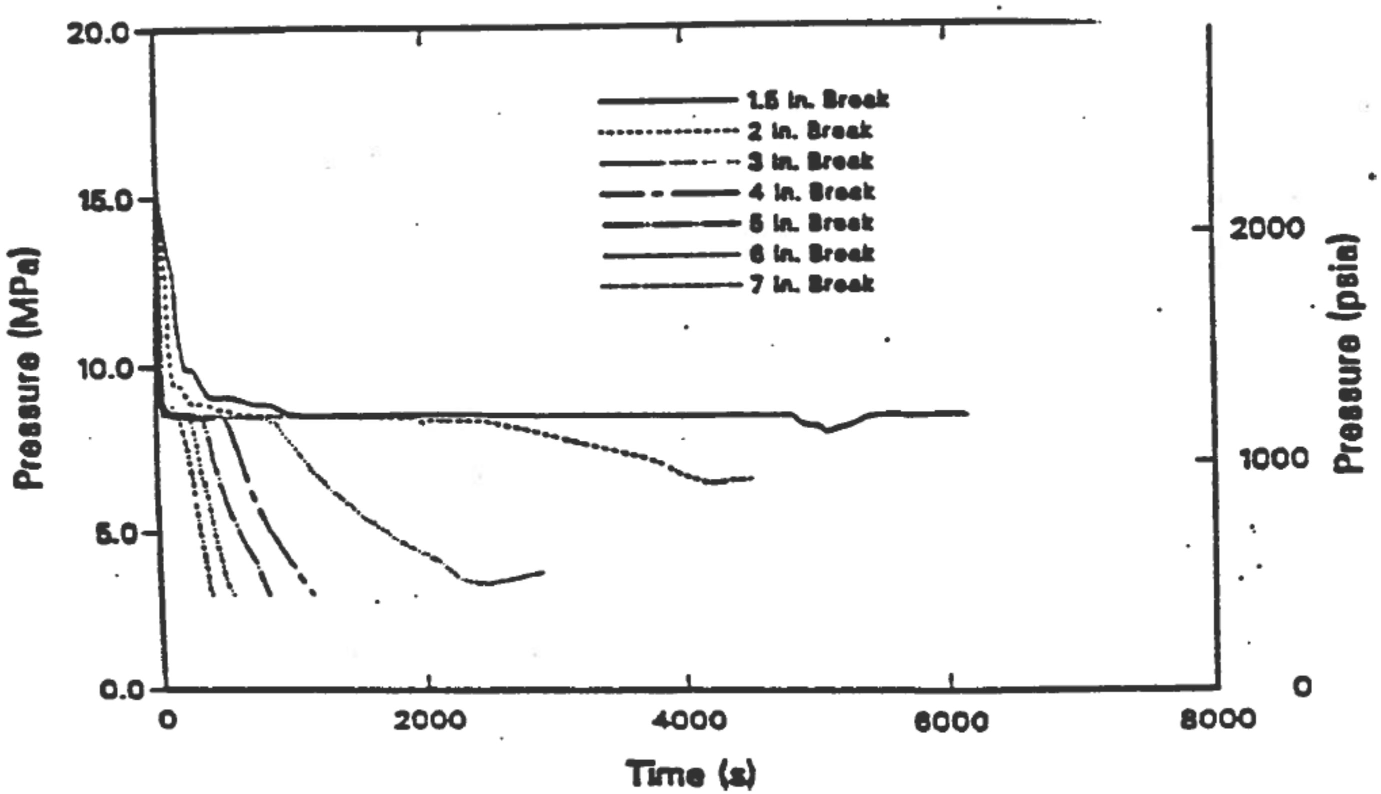
- Breaks can occur on the top, side or bottom of a pipe
  - Break elevation determines maximum water level in reactor vessel
    - Hot and cold legs are typically between 28” and 42” diameter
  - Flow stratification
    - Allows only steam to exit a top break
    - Allows only liquid to exit a bottom break
  - Top breaks
    - Allows for more steam to exit the break
      - Results in more energy removal and a quicker pressure decline which leads to higher ECCS flow rates
  - Bottom break
    - Allows for more liquid to exit the break
      - Results in more mass being lost through the break
      - Less energy removal and slower pressure decline than a top break (lower ECCS flows)
  - **Limiting breaks are typically pipe bottom breaks**



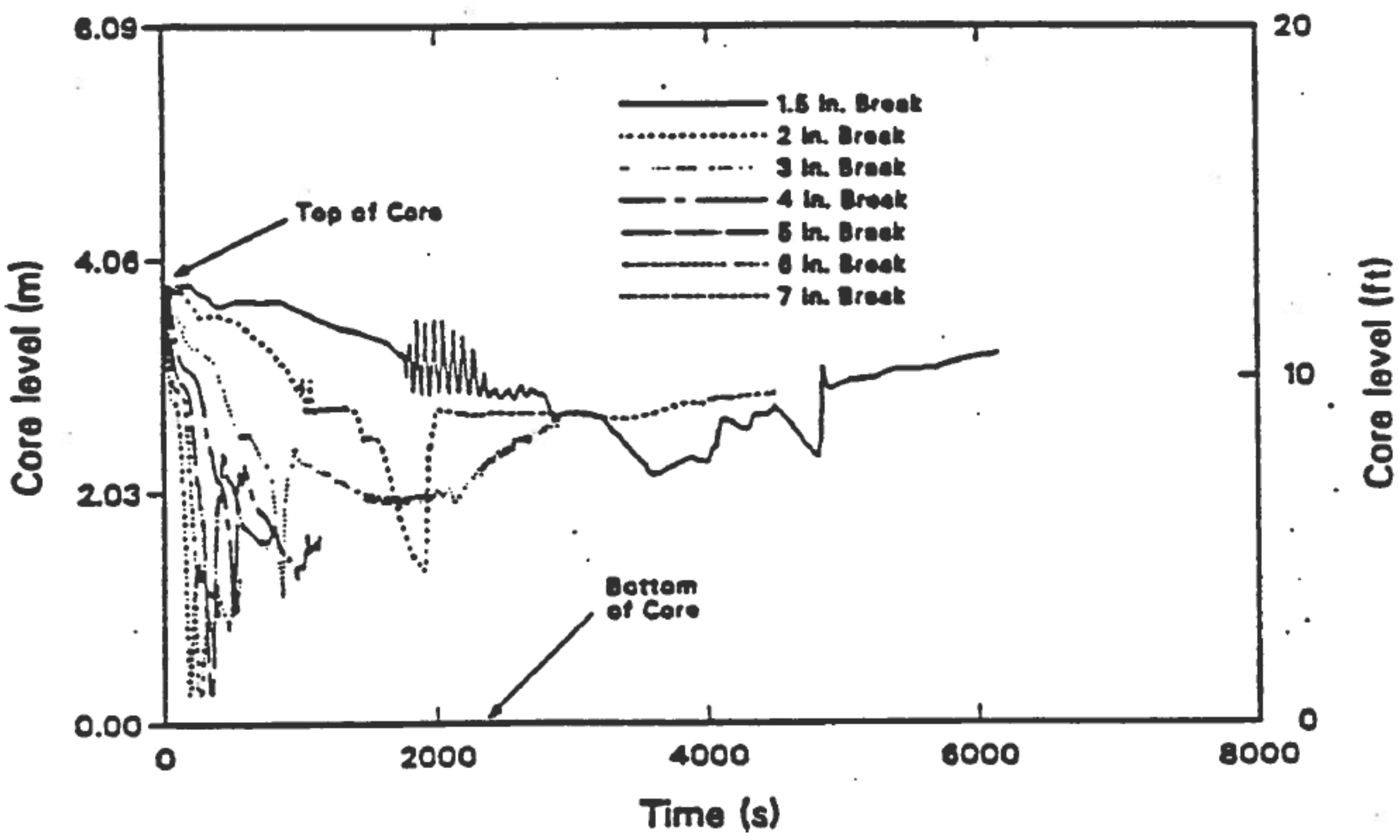
Example SBLOCA Analysis Results

- INEL study performed for the Westinghouse 4-loop RESAR-3S plant
  - Results shown are from RELAP5 calculations documented in NUREG/CR-4384
- Break spectrum evaluated (diameter of cold leg centerline breaks)
  - 1.5 in (3.8 cm)
  - 2.0 in (5.1 cm)
  - 3.0 in (7.6 cm)
  - 4.0 in (10.2 cm)
  - 5.0 in (12.7 cm)
  - 6.0 in (15.2 cm)
  - 7.0 in (17.8 cm)

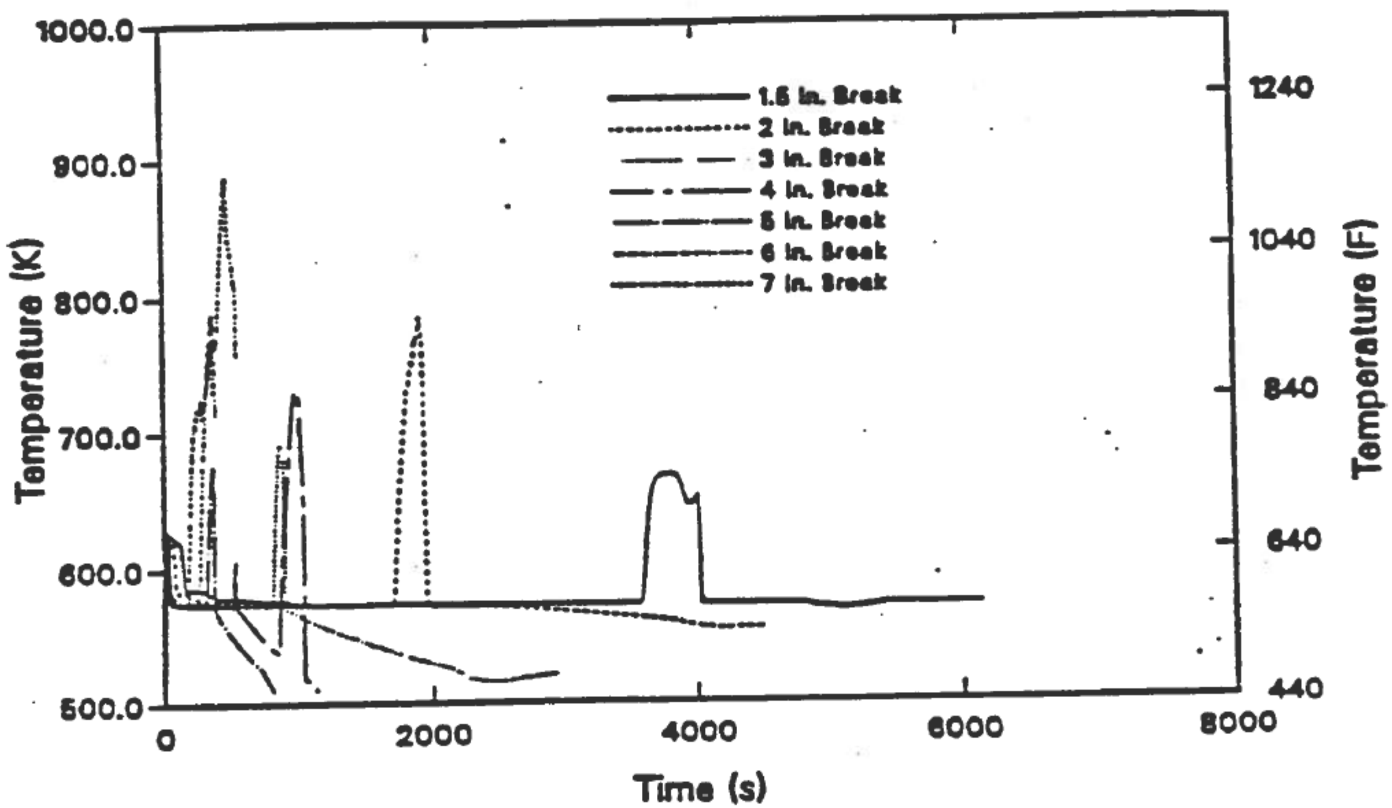
Example RCS Pressure Responses



Example Core Collapsed Liquid Level Responses



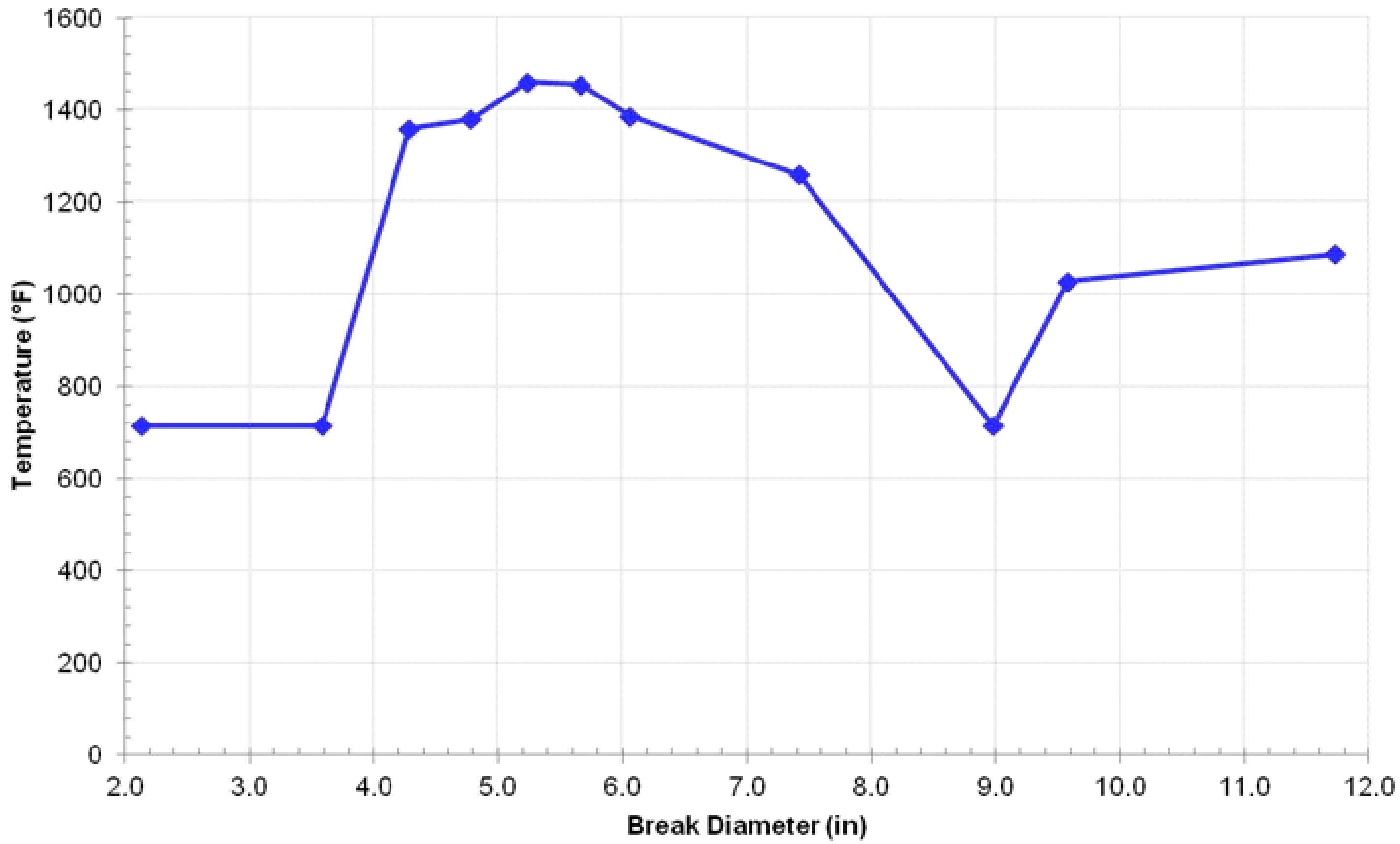
Example Peak Cladding Temperature Responses



PCT is for 6 inch diameter break

Example Break Spectrum Analysis Results

- The peak cladding temperature is shown as a function of break diameter for the Oconee nuclear power plant (B&W design)
  - Data from Oconee Updated Final Safety Analysis Report, PCT is for 5.5 inch diameter break



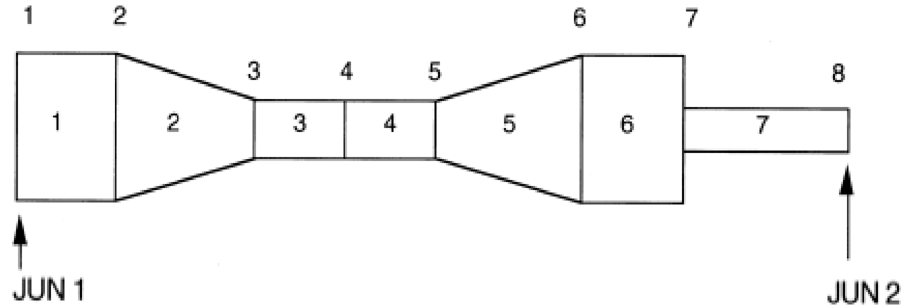
Summary of Expected SBLOCA Analysis Results

- The "limiting" SBLOCA break is the one which results in the highest peak cladding temperature
- The limiting break is typically located on the bottom of a cold leg
- The limiting break size typically:
  - Is large enough so that SG heat removal capability is degraded or lost due to voiding in the upper regions of the RCS
  - Is small enough so that the core decay power cannot be removed from the RCS through the break except by the flow of steam
  - Results in an extended period when RCS inventory loss continues while RCS depressurization is minimized

Length-to-Diameter (L/D) Modeling Considerations for One-Dimensional Flow Paths

- Examples of applications where one-dimensional modeling is used: PWR hot legs and cold legs, PWR and BWR steam lines, BWR recirculation lines.

- Use of longer cells increases the Courant time step limit (calculated: cell length divided by the cell fluid velocity) allowing for improved running times, but the detail in the solution is reduced.



Length-to-Diameter (L/D) Modeling Considerations for One-Dimensional Flow Paths

- Use of shorter cells provides more solution detail but increases the running time.
- Recommend making cell lengths as long as practical, based on the acceptability of using a set of average conditions to represent the fluid over the cell **length**.
- A general rule of thumb is to use cell L/D ratios of about 5.
- Recommend using shorter cells in regions where changes in fluid conditions are expected to be greater per unit of cell length, and gradually increase/reduce lengths from one cell to the next.

Length-to-Diameter (L/D) Modeling Considerations for One-Dimensional Flow Paths

- Compromises using L/D < 5 may be needed to adequately model the physical flow and/or thermal behavior.

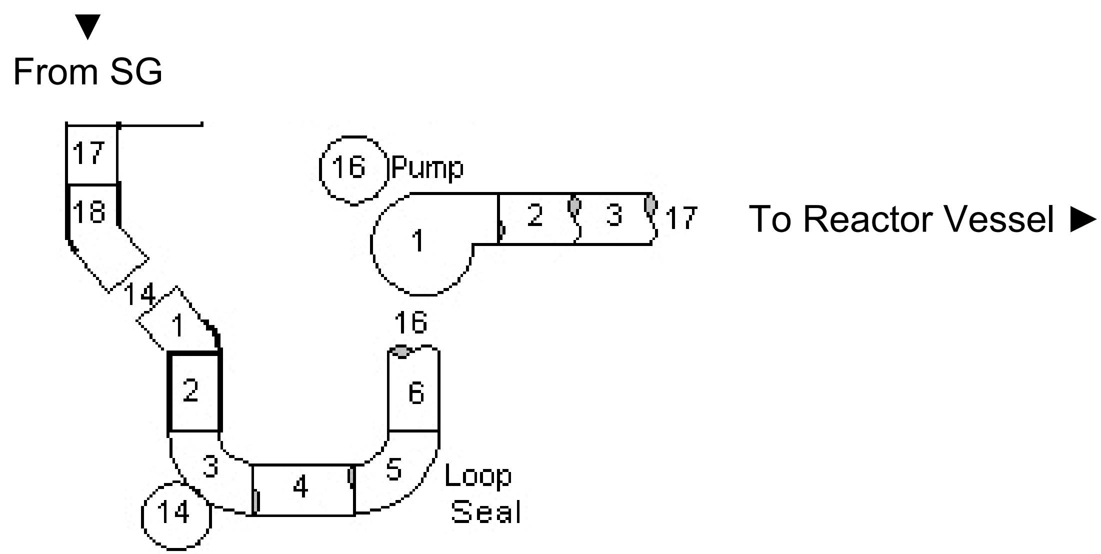
- Example compromise: Axial noding for PWR SGs selected based on heat transfer modeling considerations. Same axial noding used for tube primary and boiler secondary regions, regardless of cell L/D considerations.

- A basic assumption for one-dimensional modeling is that it must be acceptable to use average fluid conditions to represent the actual range of fluid conditions over the cell **width**.

- For consistency with that assumption, cell L/D should generally be greater than 1.

PWR Loop Seal Nodalization

- “Loop seals” are the primary coolant system pipes from the steam generator outlet to the reactor coolant pump inlet in conventional U. S. PWRs.



- Loop seal behavior is key for the response of PWRs during LOCAs.

PWR Loop Seal Nodalization

During PWR LBLOCAs:

- Reactor coolant system depressurization is rapid.
- Steam produced by flashing can easily reach the break, resulting in a complete system depressurization.
- Low system pressures result in high safety injection flows from the HPI, LPI and accumulator systems that lead to refilling of the core.
- Steam produced by flashing in hotter and higher elevation regions of the reactor coolant system forces the water from the loop seals into the reactor vessel, which increases core inventory and enhances core cooling.

PWR Loop Seal Nodalization

During PWR SBLOCAs:

- Reactor coolant system depressurization is slow.
- Steam produced by flashing cannot reach a break located in a cold leg, causing the depressurization to stall near the SG secondary relief valve opening setpoint pressure.
- Core decay heat cannot be fully removed from the reactor coolant system by flow of water out the break, so some SG heat removal is needed.
- At this relatively high reactor coolant system pressure, safety injection flow from the HPI system is insufficient to replace the water flowing out the break.

PWR Loop Seal Nodalization

During PWR SBLOCAs (continued):

- Reactor coolant system inventory continues declining until a loop seal can clear, providing a path for steam to flow through a cold leg break.

- Once a loop seal clears, reactor coolant system depressurization can resume, allowing for increased HPI system flow, initiation of accumulator and LPI system flows, and increased core inventory.

- The mechanism of loop seal clearing results in depression of the core level, potentially leading to a fuel heat-up in the top of the core.

- Because of its significance for safety, an adequate simulation of the loop seal clearing process is needed.

PWR Loop Seal Nodalization

Recommendations for Loop Seal Modeling

- Nodalization should be set up to permit horizontal flow regimes to be represented in the bottom of the loop seal.
- Allows for simulation of steam escaping over liquid, which affects the timing of clearing and provides a capability for loop seals to remain partially filled following “loop seal clearing”
- Flow resistance through a partially-filled loop seal can be much greater than through an empty loop seal.

Frictional Pressure Drop Modeling

- Traditionally, TRACE K loss coefficients could only be input as constants. A recent code improvement permits the input of Reynolds Number dependent loss coefficients:

  K = B/Re**C,

  where B and C are entered as constants

- By using constant, fully-turbulent, user-input K loss coefficients for bends, orifices, valves, etc., a TRACE plant model can underpredict the total physical flow loss at near-stagnant flow conditions (illustrated for pipe wall friction in the diagram on the next slide).

Frictional Pressure Drop Modeling

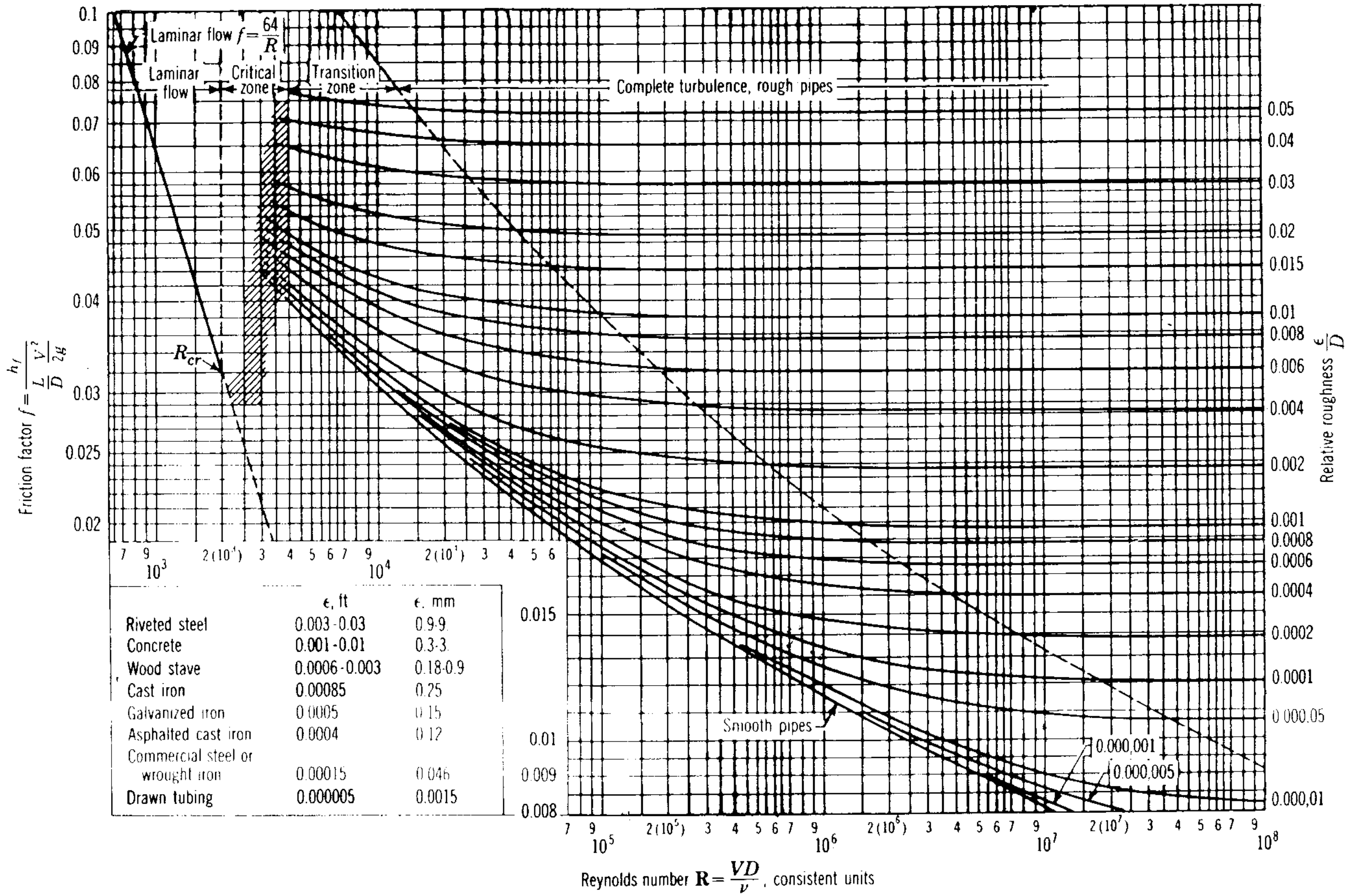


Figure 5.21 Moody diagram.

Reference: V. L. Streeter and E. B. Wylie, *Fluid Mechanics*, Eighth Edition, McGraw-Hill, 1985.

Frictional Pressure Drop Modeling

- The Reynolds Number dependent flow loss option permits the user to better simulate the greater frictional resistance at very low flows.
- This approach provides for more physical and stable solutions under near-stagnant conditions.
- Frictional pressure drop modeling is a frequent source of errors.
- The following slide summarizes how the various components of the total frictional pressure drop are combined together.

Frictional Pressure Drop Modeling

The total frictional flow loss at a cell face is the sum of:

(1) The user-input FRIC or K flow loss (constant or Reynolds Number dependent),

(2) The TRACE-calculated abrupt area change FRIC flow loss (if requested), and

(3) The TRACE-calculated wall friction flow loss, where f_{wall} is the homogeneous-flow friction factor that TRACE calculates as a function of hydraulic diameter, wall surface roughness and flow rate.

When entering FRIC data, the total frictional pressure drop is:

$$\Delta P = [\, FRIC_{User} + FRIC_{Abrupt} + f_{wall} \,] * [\, L / D \,] * [\, \rho * V^2 / 2 \,]$$

When entering K data, the total frictional pressure drop is:

$$\Delta P = [\, K_{User} + (FRIC_{Abrupt} * L / D) + (f_{wall} * L / D) \,] * [\, \rho * V^2 / 2 \,]$$

where L is the sum of the half-cell lengths on each side of the cell face, D is the hydraulic diameter of the cell face and velocity V is based upon the flow area of the cell face.

Regarding the input for D, note that the SNAP "calculate hydraulic diameter" option is only applicable for situations where parallel flow paths are not lumped together (such lumping is done, for example, when modeling all steam generator tubes using a single TRACE PIPE component).

Loop Elevation Closure

TRACE contains limited provisions for assuring that elevations are closed around the thermal-hydraulic flow loops of a system model.

A TRACE Type 5 Steady State is a static check option which calculates whether zero loop flows are achieved with all pumps tripped and all wall heat transfer inactive

Hand calculation checks for elevation closure also can be done, but typical TRACE system models contain many connected flow loops and the required hand calculations are very tedious.

SNAP provides assurance that elevation closure errors do not introduce artificial driving heads for flows through the loops of a system model.

With previous SNAP versions, false indications of loop closure were provided, especially for situations where flow loops included VESSEL component source connections.

Recent SNAP experience indicates that difficulties with the elevation checker have been resolved and that users can be confident that all loops are closed when SNAP indicates that is the case.

Calibration of System Model Pressure Drops

- After a system model has been developed and checked for loop elevation closure, it may be run with pumps and heat sources powered, so as to initialize the fluid and heat structures at the desired set of conditions.

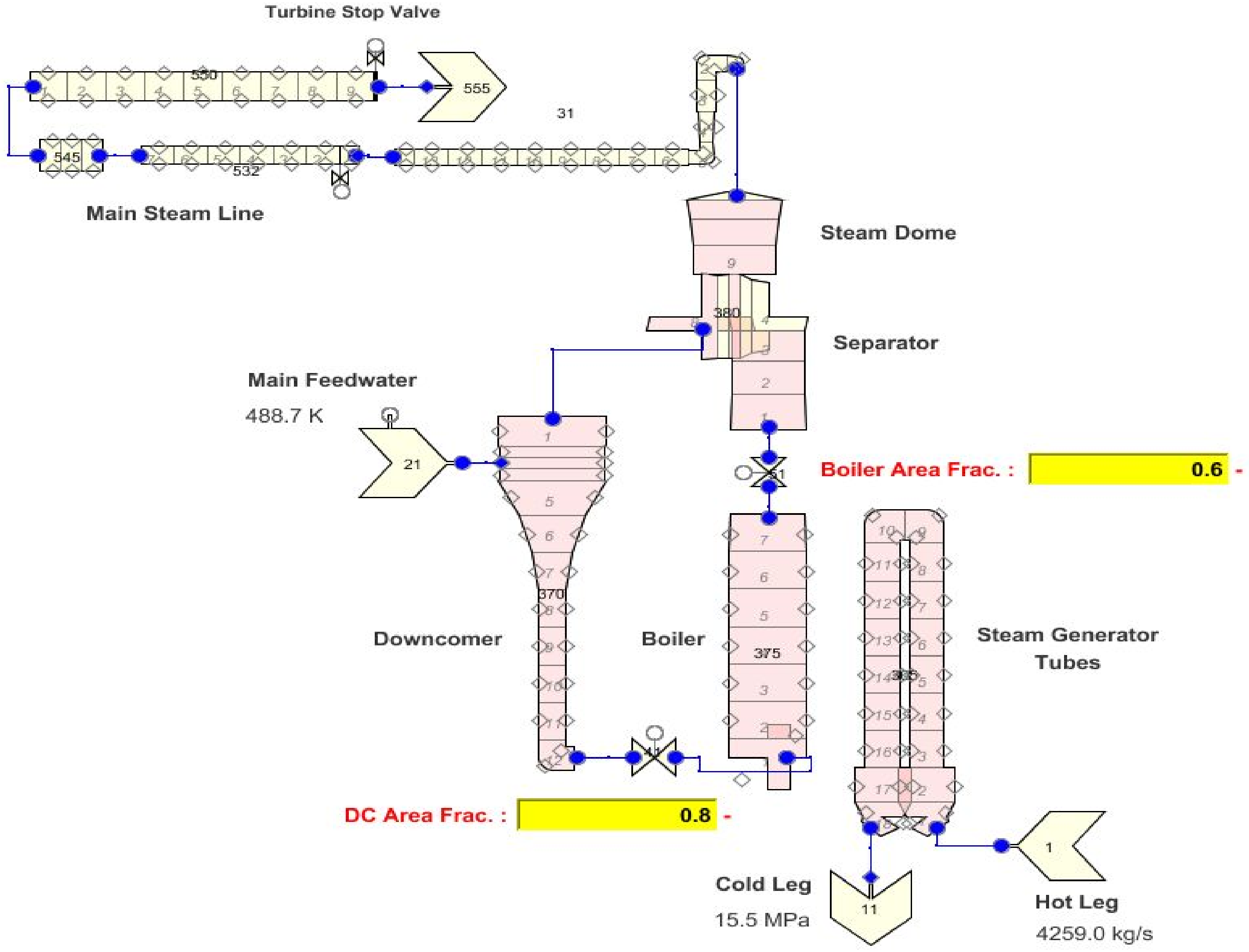
- When the desired loop flow rates have been achieved, the analyst should compare the calculated differential pressures with the available data from the prototype facility. For PWRs, prototype data is often available for the differential pressures across the core, reactor vessel, steam generators and reactor coolant pumps. For valid comparisons, the analyst needs to be aware of the basis for the prototype data (especially whether flow and/or elevation effects are included in the data).

- For PWRs, the pressure gained at the reactor coolant pump is that required to drive the loop flow at the desired rate through the reactor vessel, hot leg piping, steam generators and cold leg piping.

- Adjustments are typically required in the FRIC or K input in order for an adequate match to be achieved between all of the TRACE and prototype differential pressures. Best to make these adjustments at locations where the geometry and flow behavior is particularly complex and uncertainty in the flow loss is the greatest (for example, the turning and cross-flow through internals in reactor vessel lower plenum).

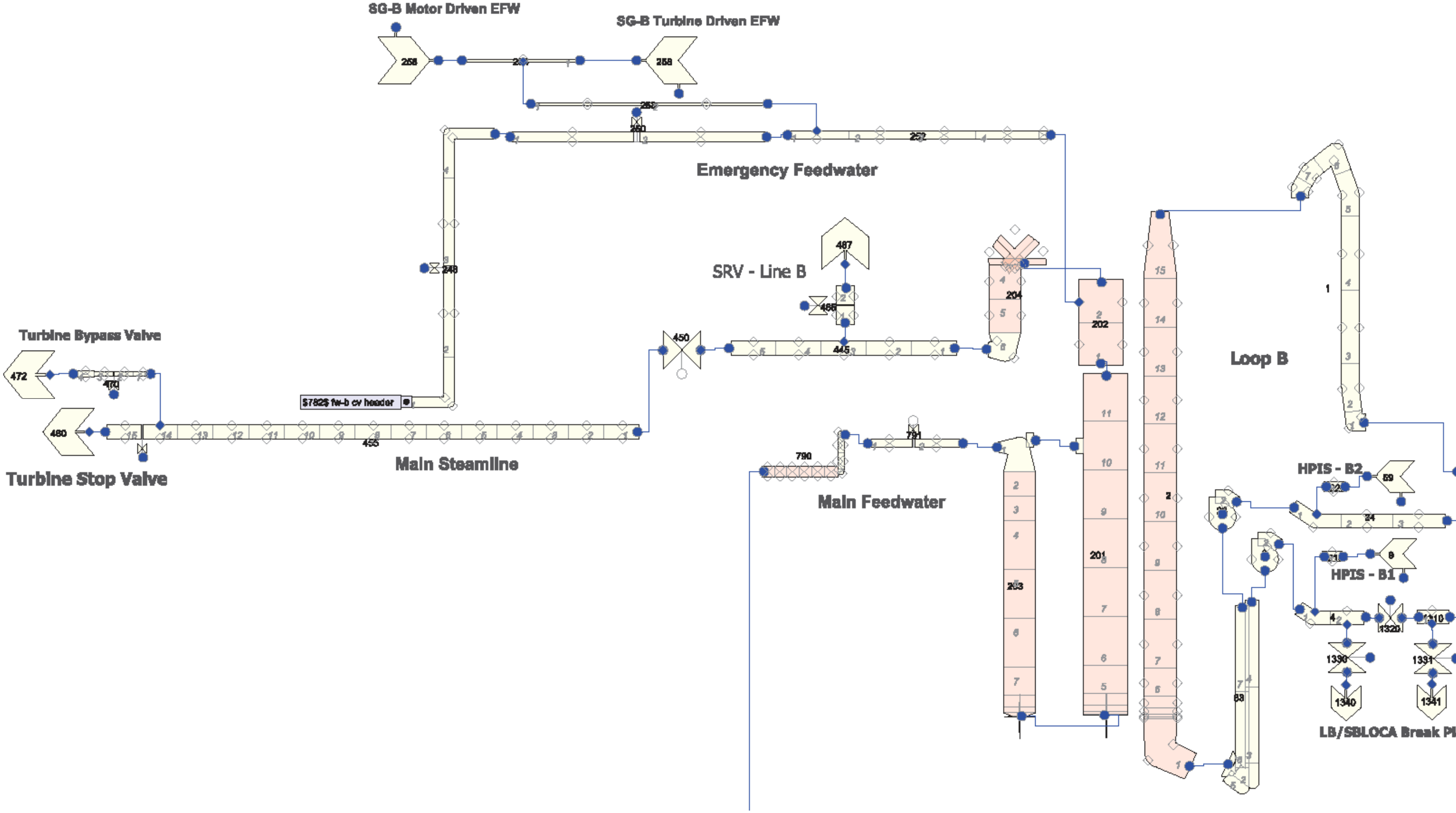
Steam Generator Modeling

Example U-Tube Steam Generator Nodalization



Steam Generator Modeling

Example Once-Through SG Nodalization



Steam Generator Model Balancing

Obtaining satisfactory agreement between TRACE-calculated and desired values for the SG portions of plant system models is a significant part of the effort needed to develop a new plant model or revise an existing plant model.

It is highly recommended that the SG model be first balanced separately from the remainder of the plant system model and then integrated into it once acceptable SG model performance has been attained.

Re-balancing of existing SG models is needed to account for:

- The effects of TRACE code revisions implemented since a SG model was last used.
- Revisions in the target SG conditions, such as required for a model supporting a plant power up-rate.

Parameters Important for Balancing a SG Model

Key Primary System Parameters Affecting SG Steady-State Performance

- Hot leg fluid temperature
- Cold leg fluid temperature
- Coolant average fluid temperature
- Coolant loop flow rate
- Power removed from the primary coolant system

Key Secondary System Parameters Affecting SG Steady-State Performance

- Main feedwater liquid temperature
- Main feedwater/main steam flow rate
- SG pressure (determines SG tube secondary side heat sink temperature, and significantly influences SG tube heat transfer rate and SG power)

Parameters Important for Balancing a SG Model

Additional Secondary System Parameters Affecting SG Transient Behavior

- Secondary side liquid mass

  Affects SG dry-out behavior during a station blackout accident

- Downcomer level

  Many plant actions, such as main and auxiliary feedwater trips, are based on SG narrow-range and wide-range level setpoints

- Recirculation ratio, which is defined as the mass flow rate from the separator into the downcomer divided by the main feedwater/main steam mass flow rate

  Model parameters selected to match a desired recirculation ratio during steady state operation also influence boiler region flow and SG heat removal behavior during transient and accident events

Steam Generator Balancing Methods

Successful balancing or re-balancing of SG models involves finding a compromise solution in which most, but typically not all, of the calculated parameter values closely match the desired prototype conditions

Compromise is needed because of code and plant model limitations for predicting the complex flow behavior in the SG boiler region

1-D modeling
Swirling axial and cross flows in the tube bundle
Local effects in regions of tube support plates

Steam Generator Balancing Methods

Compromise is also needed because of inconsistencies and uncertainties in the set of desired steady state conditions:

The basis for the desired loop mass flow rate condition typically is not clear

Generally *small* uncertainties: primary system temperatures and pressures, SG power

Generally *moderate* uncertainties: secondary system pressure, downcomer level, main feedwater temperature, main feedwater flow rate and steam flow rate

Generally *large* uncertainties: secondary system mass and recirculation ratio (ambiguity due to two possible definitions, with desired values often listed without indicating the corresponding definition)

Steam Generator Balancing Methods

The parameters selected for compromise should be based on the expected principal application of the plant system model

For LOCA applications the primary system parameters should not be compromised (loop average temperature, loop flow rate, SG power)

For station blackout, main steam line break, steam generator tube rupture and secondary-related non-LOCA transients, the secondary system parameters should not be compromised (secondary pressure, downcomer level, secondary mass, recirculation ratio)

Based on the significance of the compromises, consider that more than one SG model may be required in order to satisfactorily address all application needs

BREAK Component Geometry Input Considerations

For both SBLOCAs and LBLOCAs, the reactor coolant system component is connected to a BREAK component representing the fluid conditions in the containment.

The BREAK component imposes a pressure boundary condition one cell away from the TRACE reactor coolant system component.

Studies were performed to develop guidelines for the BREAK component geometry input.

These studies and the results are summarized as follows…

BREAK Component Input Guidance for Unchoked Flow Conditions

BREAK component geometry affects the break flow behavior.

The geometry is input as follows:

1. DXIN specifies the BREAK component length.
2. VOLIN specifies the BREAK component volume.

These two input parameters determine the volume-centered flow area for the BREAK:

A = VOLIN / DXIN

BREAK Component Input Guidance for Unchoked Flow Conditions

Four modeling situations might call for the use of the BREAK component:

1. Modeling the outflow from a piping system into a large volume, such as might occur during SBLOCAs and LBOCAs.

2. Modeling the inflow from a large volume into a piping system, such as might occur from a containment if the pressure in the primary system drops below the containment pressure.

3. Modeling a test section with a pressure tap downstream of the area of interest and the pressure tap location is modeled with the BREAK (the time-varying pressure solution is known). The flow through the break is outflow. For example, this might be the case for a critical flow separate effects experiment.

4. Same as 3, except the pressure tap is upstream of the flow conditions being modeled so that the BREAK would be modeling inflow. For this case, the BREAK forces flow into a system based on differential pressure (rather than as might be done by specifying the inflow rate using a FILL component).

BREAK Component Input Guidance for Unchoked Flow Conditions

For system outflow into the BREAK, the flow at the junction is determined by the pressure difference between the BREAK and the connected volume.

• If the connecting junction is choked, the junction thermodynamic properties are set equal to the conditions in the upstream connected volume.

• If the connecting junction is unchoked, the junction thermodynamic properties are defined as the length-weighted average of the connecting volume and the BREAK component.

  - Averaging of the BREAK and connecting volume thermodynamic properties for use in the junction momentum equation can be eliminated by using a very small DXIN for the BREAK component (thereby giving the connecting volume the dominating weight).
  - A large DXIN relative to the DX of the connecting volume will provide more weighting to the BREAK volume in calculating the thermodynamic properties of the connecting junction.

The input BREAK length, and therefore the averaged properties, can have a pronounced effect on the interfacial and wall drag closure models.

This same thermodynamic property averaging scheme applies when a BREAK is used for inflow to a system.

BREAK Component Input Guidance for Unchoked Flow Conditions

The pressure condition specified in the BREAK component always represents a static pressure. TRACE will internally calculate a dynamic pressure for the BREAK volume using the BREAK flow area and the length-weighted junction thermodynamic conditions:

$$P_{\mathrm{dynamic}} = P_{\mathrm{static}} - m^2_{\mathrm{junction}}/(2rho_{\mathrm{junction}} * A^2_{\mathrm{break}})$$

Where: P_{dynamic} is the BREAK volume dynamic pressure
P_{static} is the BREAK volume static pressure (input)
m_{junction} is the mass flow for the connecting junction
A_{break} is the BREAK volume area.

BREAK Component Input Guidance for Unchoked Flow Conditions

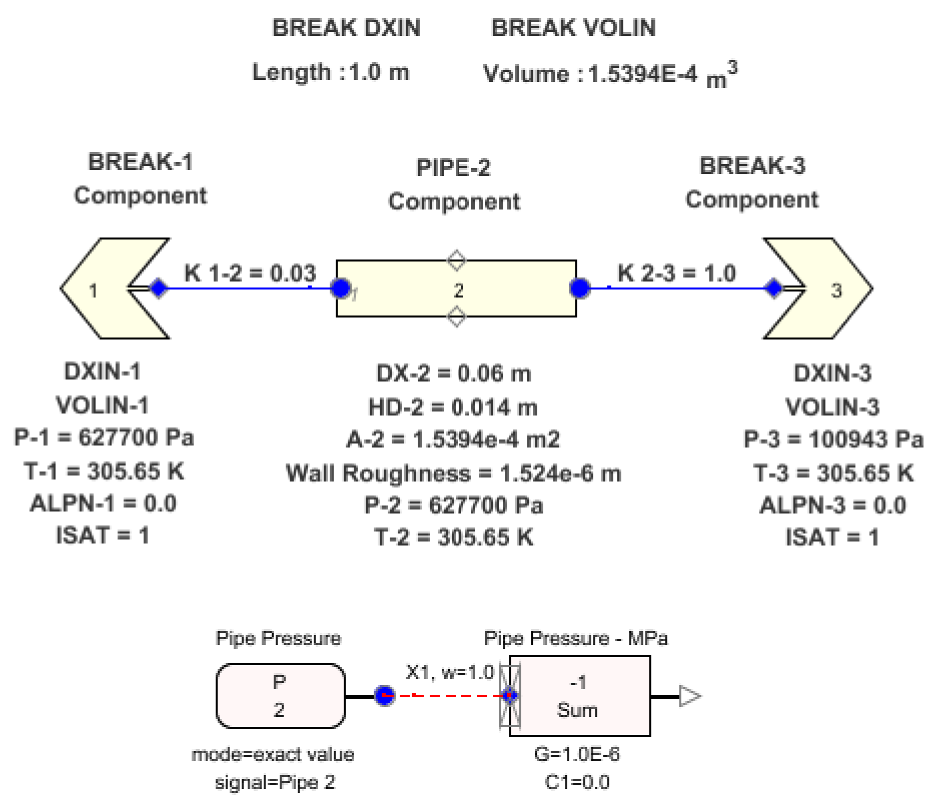
To provide guidance for the DXIN and VOLIN inputs for an unchoked BREAK component, an assessment of the TRACE calculated steady-state results of a simple BREAK-PIPE-BREAK model were compared to hand calculations using the Bernoulli equation to solve for the flow and thermodynamic conditions.

This assessment is documented in the TRACE V5.0 User's Manual, Volume 2, see the section under "Modeling Guidelines, Thermal-Hydraulic Components, Break Flow Modeling".

- The model employed represented a steady-state liquid flow condition between two BREAK components.
- BREAK DXIN and VOLIN were varied and the TRACE results were compared with the hand calculations.

BREAK Component Input Guidance for Unchoked Flow Conditions

BREAK Input Sensitivity Model:



BREAK Component Input Guidance for Unchoked Flow Conditions

Cases Evaluated and the Hand Calculation Results:

DXIN-1 and DXIN-3 (m)	VOLIN-1 and VOLIN-3 (m3)	A-1 and A-3 (m2)	Flow (kg/s)	Velocity V-2 (m/s)	P-2 (MPa)	Condition
1.0	1.5394e-4	1.5394e-4	4.67	30.47	0.5884	Static
0.1	1.5394e-5	1.5394e-4	4.67	30.47	0.5884	Static
0.01	1.5394e-6	1.5394e-4	4.67	30.47	0.5884	Static
0.06	9.2364e-6	1.5394e-4	4.67	30.47	0.5884	Static
0.06	0.0001	0.0016667	4.67	30.47	0.13034	Dynamic
0.06	0.001	0.016667	4.67	30.47	0.12643	Dynamic
0.06	0.01	0.16667	4.67	30.47	0.1264	Dynamic
0.06	10000.0	166667.0	4.67	30.47	0.1264	Dynamic

BREAK Component Input Guidance for Unchoked Flow Conditions

The assessment led to the following recommendations for unchoked flow:

1. If a TRACE volume, such as a PIPE component volume, is connected to a pressure source or sink which possesses the same flow area, the BREAK component values of DXIN and VOLIN should be set equal to the connecting volume values. For this situation, the pressure input for the BREAK component represents a static pressure and the BREAK dynamic pressure will be calculated by TRACE using the above equation.

2. If a TRACE volume, such as a PIPE component volume, is connected to a large pressure source or sink volume, the source or sink boundary volume should be modeled as a BREAK component with a small value for DXIN and a large value for VOLIN. This input results in a large flow area and a very small velocity in the BREAK component. Consequently, the pressure input for the BREAK component represents a dynamic pressure. That is, if the BREAK flow area is large, the BREAK input pressure equals the static pressure which equals the break dynamic pressure. A rule of thumb for this case is to use a DXIN equal to the adjacent cell DX and a VOLIN that is at least 100 times larger than the VOL of the adjacent cell.

BREAK Component Input Guidance for Choked Flow Conditions

A similar study was performed for choked flow conditions. The outcome of the study shows that for choked flow conditions:

- The flow rate is not dependent on the value of the BREAK VOLIN and DXIN
- The choked flow rate is determined by the flow area of the break junction connected to the BREAK component

The BREAK component input guidelines for unchoked flow described above also pertain for choked flow conditions.

Modeling of Small Breaks

The original recommendation for modeling SBLOCAs (going back to TRAC-P) was to use a short cell, with a length equal to the pipe wall thickness, between the ruptured pipe and the BREAK component representing the pressure boundary condition.

This modeling approach was needed in order to provide the critical flow model with correct fluid conditions immediately upstream of the break

This approach, however, led to difficulties because of the very high velocities in the short cell

Modeling of Small Breaks

With recent improvements in the TRACE momentum solution, the recommendation for SBLOCAs is now to use a simpler break modeling approach:

Single junction PIPE or VALVE components should be connected directly to the TRACE component representing the ruptured pipe

Prior difficulties with VALVE component inherent flow loss also have been resolved. The user may now remove inherent loss to prevent uncertainty and/or double counting of flow losses specified at coolant system breaks.

The Reflood Process

Reflood refers to the third phase of a LBLOCA event in a PWR. The reactor coolant system blowdown has completed, the emergency core cooling systems have refilled the reactor vessel lower plenum, and replenishment of water into the hot core (from the bottom upward) has begun.

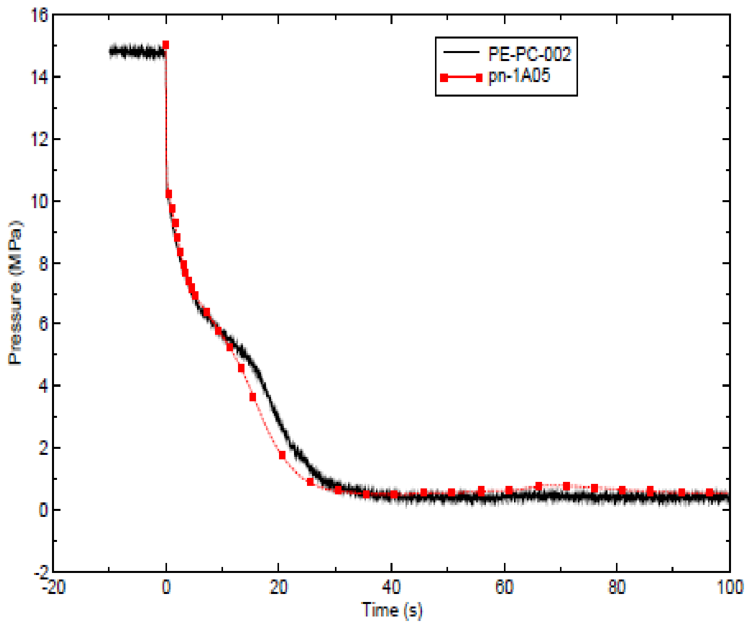


Figure C.1-37. L2-5 Intact Loop Hot Leg Pressure.

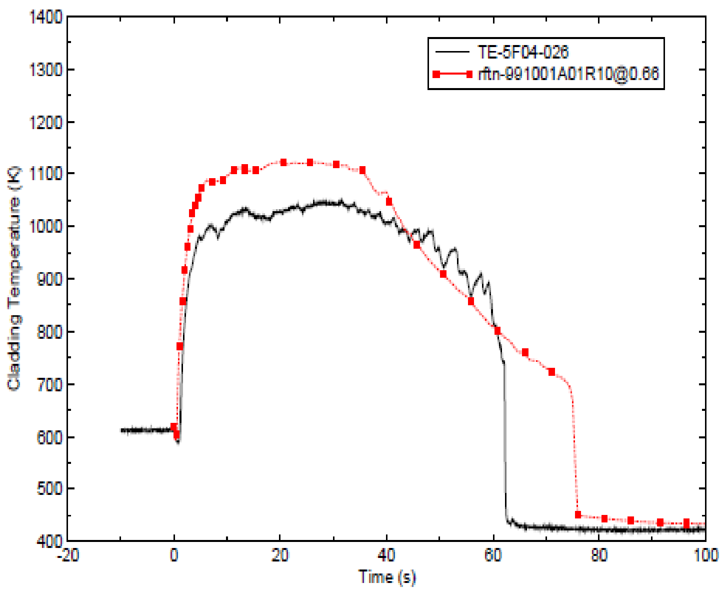


Figure C.1-53. L2-5 Cladding Temperature at 0.66 m.

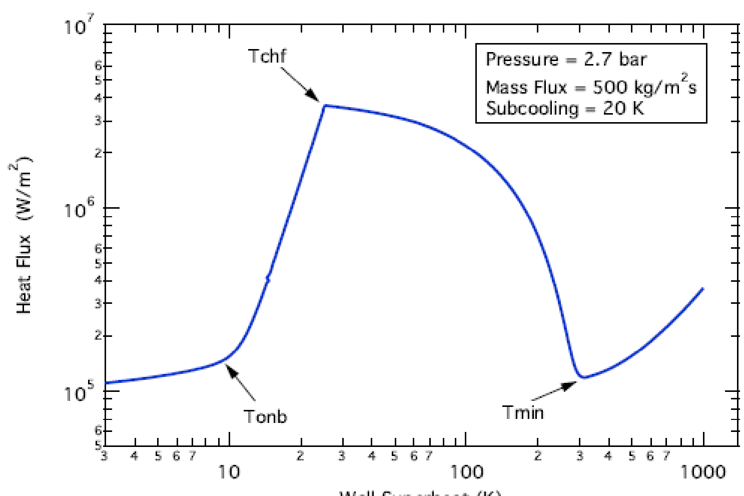
The Reflood Process

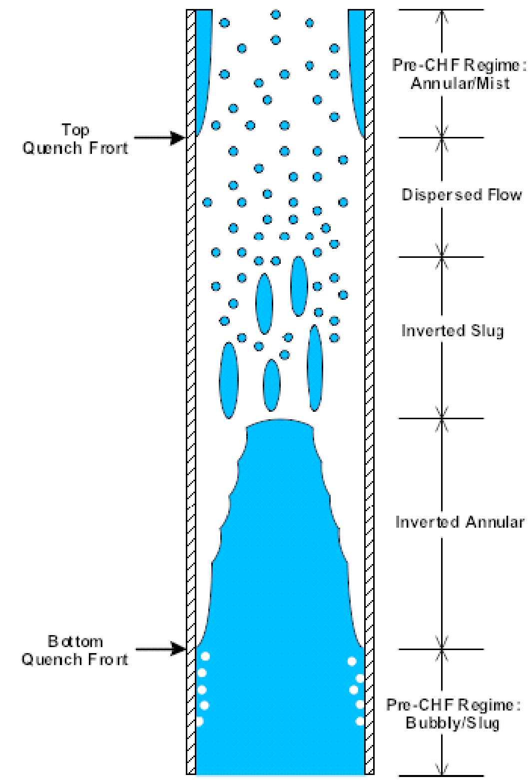
Fuel cooling from film boiling and dispersed flow heat transfer ahead of the quench front is minimal.

Rapid fuel cooling (from the return to nucleate boiling heat transfer) occurs at and below the quench front elevation.

Because the pressure and steam density in the core are so low, steam produced by quenching leads to very high steam velocities that provide a significant resistance against which reflood must proceed.

The reflood process is therefore relatively slow, with fuel temperatures in the upper core region remaining elevated for extended periods.





The Reflood Process

- TRACE contains a unified heat transfer package that includes models for reflood heat transfer.
- The exercise that follows will provide experience with TRACE reflood modeling.
- Any questions before we move on to the exercise?