The Abdus Salam International Centre for Theoretical Physics

United Nations Educational, Scientific and Cultural Organization

IAEA International Atomic Energy Agency

2152-5

# Joint ICTP-IAEA Course on Natural Circulation Phenomena and Passive Safety Systems in Advanced Water Cooled Reactors

*17 - 21 May 2010*

## SELECTED EXAMPLES OF NATURAL CIRCULATION FOR SMALL BREAK LOCA AND SOME SEVERE ACCIDENTS

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IAEA Training Course on Natural Circulation in Water-Cooled Nuclear Power Plants,
International Centre for Theoretical Physics , Trieste, Italy
17th to 21st May, 2010, Paper ID. T21b

# SELECTED EXAMPLES OF NATURAL CIRCULATION FOR SMALL BREAK LOCA AND SOME SEVERE ACCIDENTS

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## KEY WORDS

Thermal-hydraulics, Small break loss-of-coolant accident, Severe accidents, Natural circulation.

## ABSTRACT

In all light water reactors (LWRs), natural circulation is an important passive heat removal system. The March 1979 accident at TMI-2 brought into question the capability of natural circulation cooling to remove core decay heat, especially during accident situations. Because natural circulation is expected to be an essential core heat rejection mechanism during certain kinds of accidents or transients in a PWR (e.g., small break LOCAs or operational transients involving loss of pumped circulation), a thorough understanding of natural circulation processes and factors that influence the natural circulation response of the reactor system is necessary.

In this paper, natural circulation and related major phenomena are discussed with examples for small break LOCA and severe accident cases, e.g., TMLB station black-out. Descriptions of three modes of natural circulation are provided: Single-phase natural circulation, two-phase natural circulation, and reflux condensation/boiling condensation. The basic phenomena associated with the three types of natural circulation being considered for severe accidents are also addressed: In-vessel natural circulation, hot leg countercurrent flow, coolant loop flows.

## LECTURE OBJECTIVES

Lecture on this subject will provide some idea about the importance of natural circulation phenomena for plant transients and severe accident cases (e.g. small break LOCA, TMLB-Station black-out, etc.).

## 1. INTRODUCTION

The natural circulation cooling is an essential means of removing shutdown decay heat in PWRs following the loss of forced circulation by the reactor coolant pumps (RCPs) during operational transients or following accidents. The loss of forced circulation may result from the loss of offsite power, pump failure, or operator action based on operating procedures for abnormal conditions. Natural circulation in PWRs refers to primary coolant flow within the loops of a closed primary system. Natural circulation flow is driven by differences in the average coolant density within the primary system. These density differences result from the heating of the reactor coolant in the core and the subsequent cooling of the reactor coolant in the steam generators, which are elevated relative to the core, resulting in a loop gravitational driving force. This paper will concentrate on PWRs having plant designs utilizing U-tube steam generators and it will provide overview on the importance of natural circulation for small break LOCA and severe accident cases with emphasize on natural circulation phenomena.

The paper provides a short overview on small break LOCA issue (section 2) as well as looking at the small break LOCA scenario and related main phenomena (section 3). Natural circulation and its importance are addressed in section 4. Severe accident scenario in PWRs leading to natural circulation is described in section 5. Section 6 describes and shows the importance of natural circulation during severe accidents. The final section includes concluding remarks. A list of references is given for further reading and information basis.

## 2. ORIGIN OF SMALL BREAK LOCA ISSUE (SYSTEM THERMAL HYDRAULICS BEFORE AND AFTER TMI-2)

In early 1970's, former U.S. Atomic Energy Commission convened a public hearing to explore the safety question in relation to the effectiveness of systems to mitigate the consequences of a loss of coolant accident in a nuclear reactor, in case it happens. Ultimately, after extensive public hearings, in 1974, the interim regulations were modified to provide a set of specific requirements for computer codes for ECCS analyses in Appendix K and a new section, 10 CFR 50.46 [1], requiring ECCS meet established standards. This included a definition that LOCAs are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system. The safety criteria prescribed in 10 CFR 50.46 are applicable to both large and small break LOCAs. That is to say the limits on peak cladding temperature, cladding oxidation and hydrogen generation must not be exceeded in a design basis accident. Calculations of ECCS performance using the conservative prescriptions of Appendix K resulted in the large break LOCA generally being the most limiting accident. At the time, there was a major safety research program to support code development for large break LOCA and also some limited work on small break LOCA.

The March 1979 accident at the Three Mile Island Unit 2 (TMI-2) reactor led to an extensive reorientation of light water reactor safety research programs and also regulatory changes. The TMI-2 accident was a small break LOCA, an event given significantly less attention because of the major emphasize on the large break LOCA at the time. Consequent to TMI-2, small break LOCA and plant operational transients received major attention. The experimental simulation of the natural circulation phenomena in the primary loops, including those in the two-phase stratified and counter-current flow regimes, is of primary importance to the thermalhydraulic response of a nuclear power plant during such transients. Since these phenomena are significantly dependent on facility scale, and geometry, large-scale tests for a primary system geometry representative of operational nuclear power plants are required. Either operational facilities were modified to carry out small break LOCA experiments or there were new facilities designed and constructed. It is to be noted that unlike the large break LOCA, the sequence of events following a small break LOCA can evolve in a variety of ways. Operator actions, reactor design, ECCS set points, break size and location will have a bearing how the small break LOCA scenario unfolds. Therefore in order to predict the integral system behaviour during a small break LOCA, a best estimate code must have sufficient modelling capabilities to take these factors into account. These codes are also needed to be assessed against integral system tests. After having been successfully assessed against data from a large number of scaled test facilities, best estimate codes become the ultimate repository of all previous thermal hydraulic safety research.

## 3. SMALL BREAK LOCA SCENARIO IN A PWR WITH RELEVANCE TO NUCLEAR REACTOR SAFETY AND MAIN PHENOMENA

The major characteristic difference between a small break and a large break LOCA are in the rates of coolant discharge and pressure variations with time. A typical PWR system and elevations are illustrated schematically in Figure 1. In general, small break LOCA's are characterized by an extended period (this can be tens of minutes to several hours at the lower end of the break spectrum) after the occurrence of the break, during which the primary system remains at a relatively high pressure and the core remains covered (Figure 2). As soon as the pumps are tripped, either automatically or manually, gravity controlled phase separation occurs and gravitational forces dominate the flow and distribution of coolant inside the primary system (Figure 3). The subsequent sequence of events, whether or not the core uncovers and is recovered or reflooded, depends not only on the location, shape and size of the break, but also on the over-all behaviour of the primary and secondary systems. This behaviour is strongly influenced by both automatic and operator initiated mitigation measures. In general, the reactor system response to a small break is slower compared to events after a large break. This allows more time, and different possibilities, for operator interventions. Another principal difference is the domination of gravity effects in small breaks versus inertial effects in the large breaks.

Breaks can occur in a number of ways, i.e., pump seal failure, steam generator tube failure, power-operated relief valve (PORV), automatic depressurisation system (ADS), or safety valves sticking open, and instrument-tube failure, in addition to simple pipe break or leaks. The PORV and ADS may be activated intentionally to reduce pressure and/or remove energy. The location of the break may be such as to enhance or reduce the loop flow rate due to natural circulation alone.

It is to be noted that there is no unique path of development of events following a small break LOCA in PWRs. The scenarios may change drastically by many factors such as the reactor design (e.g., U-tube or once-through steam generators, such as TMI-2), the break size, the size core bypass (allowing some fraction of the inlet cold leg flow directly into the core upper structure without passing through the core), and most importantly, by different operator interactions. As an example, the primary circulation pumps may be shut down early in a small break LOCA transient or they may be allowed to run and circulate the coolant through the core for a long time. These alternative actions can make a large difference in the nature of discharge flow, early heat removal from the core, and the liquid inventory in the system after one hour or so in the transient.

Another important possibility of different interactions is through the steam generators. The secondary side of steam generators can be isolated (no feed water flow) or they can be used for a controlled heat removal. It is also possible to cool the reactor through the so-called "feed-and-bleed" process (on the primary side). Either of these actions will have a major effect on the course of the transient. It is not the intent in this section to provide a catalogue of all possible scenarios following small break LOCA accidents. But it is important to note that an adequate set of modelling capabilities for any of the plausible scenarios will be equally adequate for all other scenarios. This is because the phenomena and processes are the same but their interactions and timing of various developments changes in different operations. Therefore in order to predict the integral system behaviour during a small break LOCA, a best-estimate code must have sufficient modelling capabilities to take these factors into account.

During a PWR small break LOCA, there is the potential for three distinct core heat ups. A typical transient history for primary system pressure and reactor vessel water level following a small break LOCA at the cold leg in a PWR is shown in Figure 5. The first heat up is caused by loop seal formation and the manometric core liquid level depression (Figure 4). Naturally occurring events including loop seal clearing and break uncovery mitigate this heat up. The second heat up occurs following the core quench caused by loop seal clearing and is caused by a simple core boil-off. During this period the primary pressure is decreasing to the accumulator set point and the steam produced by the core boil-off leaves the system via the break. Any heat ups that occur during this period are mitigated by the reflood from the accumulator water. The third possible heat up can occur following depletion of the accumulator tanks and before LPIS injection begins. One drawback to the reflood process accompanying the accumulator injection is a decrease in the ongoing depressurisation process such that another possible heat up occurs before the LPIS primary pressure set points are reached and long-term cooling is provided. Various factors affect the magnitudes of the three potential core heat ups. Some examples are break size, break direction and location, availability of HPIS, and the degree of upper head to downcomer bypass flow. Although the magnitudes of the core heat ups may vary, ECCS performance must be such that the criteria, e.g., 10. CFR 50.46 [1] is not exceeded. The interested readers can obtain further details on small break LOCA in reference [2].

## 4. NATURAL CIRCULATION AND ITS IMPORTANCE DURING SB LOCA

The March 1979 accident at TMI-2 brought into question the capability of natural circulation cooling to remove core decay heat, especially during accident situations. Because natural circulation is expected to be an essential core heat rejection mechanism during certain kinds of accidents or transients in a PWR (e.g., small break LOCAs or operational transients involving loss of pumped circulation), a thorough understanding of natural circulation processes and factors that influence the natural circulation response of the reactor system is necessary. Characterization of the natural circulation cooling processes requires:

- Identifying conditions under which natural circulation will occur,
- Determining the effectiveness of natural circulation in removing core decay heat and recovering the plant (i.e., what are natural circulation cooling limitations)
- Identifying how changing plant conditions affect natural circulation cooling.

Natural circulation will occur in a PWR primary loop (in the absence of pumped flow) whenever buoyant forces caused by differences in loop fluid densities are sufficient to overcome the flow resistance of loop components (steam generators, primary coolant pumps, etc.). The fluid density differences occur as a result of heating fluid in the core region (causing the liquid become less dense) and cooling fluid in the steam generators (causing the fluid to become denser).

The buoyant forces resulting from those density differences cause fluid to circulate through the primary loops, providing a means of removing the core decay heat. Depending on the primary loop fluid inventory, natural circulation consists of three distinct modes of cooling [3]:

- Single-phase
- Two-phase (liquid continuous)
- Reflux condensation (or boiler-condenser mode for once-through steam generators)

Progression from the single-phase mode through the two-phase and reflux condensation modes occurs as primary system liquid inventory decreases (Figure 6). Natural circulation flow in PWRs is driven by temperature induced density gradients, enhanced by a thermal center elevation difference between the hot (core) and cold (steam generator) regions in the primary loop. This density gradient produces a buoyancy effect that drives the natural circulation flow. Thus, single-phase natural circulation is the flow of an essentially subcooled primary liquid driven by liquid density differences within the primary loop. Two-phase natural circulation is normally defined as the continuous flow of fluid and vapor. In this mode of natural circulation, vapor generated in the core enters the hot leg and flows along with the saturated liquid to the steam generator, where at least some of the vapor is condensed.

Hence, density gradients are affected in two-phase mode not only by temperature differences, but also mainly by the voids in the primary loop. In both single-phase and two-phase natural circulation, the mass flow rate is the most important heat removal parameter. In contrary, during reflux condensation, the loop mass flow rate has a negligible effect because the primary mechanism of heat removal is vapor condensation. In the reflux condensation mode, single-phase vapor generated in the core flows through the hot leg piping, is condensed in the steam generator, and flows back to the core as a liquid. In summary, the three modes of natural circulation are distinguishable based upon characteristic mass flow rates, loop temperature difference behavior, and basic phenomenological differences.

The dominant heat transfer mechanism in single-phase natural circulation cooling is convection, making the loop flow rate the most important parameter governing heat removal. Heat generated by the core is convected away from the reactor vessel through the hot leg to steam generators (heat sink) via the subcooled primary liquid. Heat is transferred from the primary side to the secondary side in the steam generator. The cooling cycle is completed when the cooled primary fluid flows back to the reactor vessel. The amount of heat removed from the core through single-phase natural circulation cooling is normally the amount produced by decay heat power levels (about 5% core power).

Since the study of natural circulation cooling in PWR systems became of interest, work on the characterization of natural circulation has been focused in several areas, including effects of both primary and secondary liquid inventory and distribution on natural circulation effectiveness, the stability of the various natural circulation modes and transitions between modes as well as the possibilities of natural circulation flow interruption due to instabilities, countercurrent-flow limiting in the hot leg [4] and steam generator tubes, and the effects of noncondensables on natural circulation process. With respect to the primary side liquid inventory issue, natural circulation will provide decay heat removal at significantly reduced primary side inventory. The concern here is identifying the minimum primary side liquid inventory at which natural circulation will continue to provide adequate cooling of the core. Similarly, the steam generator secondary will continue to be a heat sink for the primary a significantly reduced secondary liquid inventories. Again, the concern is identifying the minimum secondary inventory that will ensure continued natural circulation flow in the primary loop. An additional concern with respect to the secondary inventory is the instabilities in primary side natural circulation flow caused by severely reduced secondary side liquid levels. These secondary side induced flow instabilities could reduce the effectiveness of natural circulation cooling.

Generally, the stability of the different modes of natural circulation cooling, as well as the stability of transients between the modes, is of concern because natural circulation will be the primary mechanism for core decay heat rejection for certain kinds of PWR accidents or transients, and instabilities in natural circulation process could lead to an interruption of natural circulation flow with a corresponding reduction in the removal of core decay heat. Thus an understanding of factors that influence the onset of flow instabilities as well as the effects of the instabilities on decay heat removal is necessary.

The presence of noncondensable gas in the primary side of a steam generator operating in the two-phase or reflux condensation mode of natural circulation is of concern because the gas may have a large effect on the condensation process occurring in the steam generator. The noncondensable gas in the steam generator tubes can cause a redistribution of the condensation locations as well as influence the amount of liquid being carried to the down side of the U-tubes. Thus the potential exists for the noncondensable gas to have a considerable influence on the natural circulation process. The presence of noncondensables in the primary loop may impede or even stagnate the natural circulation flow, thereby significantly reducing or terminating the heat removal capability of the steam generators both for single-phase and two-phase natural circulation cooling. Noncondensable gases can be introduced into the primary system through safety injection and by fuel degradation. As examples, hydrogen from the pressurizer vapor space, air dissolved in the refueling water, and nitrogen from accumulators (once they are depleted of water), are several sources of noncondensable gases. In addition, helium may enter the primary flow system if breaching of cladding occurs. Research has concentrated on determining whether noncondensables will migrate to and collect in the upper elevations of the primary loop. There, they may interrupt the natural circulation flow and jeopardize the effectiveness of single-phase and two-phase natural circulation cooling.

In PWRs with U-tube SGs, reflux condensation occurs when single-phase vapor generated in the core flows through the hot leg piping to the SGs, and is condensed in both the upflow and downflow sides of the steam generator U-tubes. Condensate in the upflow sides of the steam generator U-tubes drains back to the hot leg and eventually back to the vessel along the bottom of the hot leg. A countercurrent flow of liquid and vapor exists in the upflow sides of the steam generator U-tubes and in the hot leg. Condensate in the downflow sides flows into the cold leg pump suction piping cocurrently with any uncondensed steam. The reflux condensation process is shown in figure 7 During reflux condensation, primary to secondary heat transfer is accomplished through vapor condensation in SGs. This heat transfer is very effective due to the high latent heat associated with condensation. Consequently, removal of decay heat from the core during reflux condensation does not require large mass flow rates or large primary to secondary temperature differences. Small mass flow rates and primary to secondary temperature differences are characteristic of the reflux condensation mode of natural circulation. The experiments carried in the PKL-III test facility [6] in a 4-loop configuration, which simulates PWR plant with U-tube steam generators, shows the major modes of energy transport observed in this facility for natural circulation cooling, as already discussed above (figure 8). These tests have been done in relation to boron dilution phenomena and some typical results seen in figure 8 were obtained at constant pressures of 12 and 30 bar respectively and a constant power of 600 kW (about 1.8% of nominal power).

## 5. SHORT DESCRIPTION OF SEVERE ACCIDENT SCENARIO IN PWR LEADING TO NATURAL CIRCULATION

There are innumerable accident sequences that lead to core damage, but not all of them can or will result in natural circulation playing an important role in the accident progression [7]. Those, in which natural circulation will be significant, do have several characteristics in common.

The reactor coolant pumps can not be running, because forced flow through the reactor coolant system (RCS) will preclude the existence of natural circulation flows. Along a similar line of thought, there should be no pumped emergency core coolant injection. While the initial injection of liquid may increase the natural circulation flow because there is now more steam being generated in the core, the temperature of the steam exiting the core will decrease as the core continues to be cooled. The decreasing temperature will result in a reduced temperature difference between hot and cooler vapor, thereby reducing the driving potential for natural circulation flows. Heat transfer from the now hotter structures in the RCS to the cooler steam leaving the core will also tend to reduce the natural circulation flow. Additionally, the presence of pumped injection flow likely means that the accident is being terminated, so that further core damage will not occur.

There should be no large breaks in the system. For example, opening of the safety relief valves (SRV) or PORVs on the pressurizer draws enough flow through the surge line that hot leg countercurrent flow disrupted, although in vessel circulation is not. Small enough breaks will still allow natural circulation flows to exist in the reactor vessel and the hot legs. For example, pump seal leaks are not large enough to preclude hot leg natural circulation flows.

As examples, two transients that would result in significant natural circulation flows are designated the TMLB and $S_2D$ sequences. The TMLB sequence refers to a transient in which all AC power is lost and no steam-driven auxiliary feedwater is supplied to the steam generators. The TMLB' station blackout sequence has traditionally been used in severe accident natural circulation studies. The steam generators receive no feedwater, there is no AC power available for the duration of the accident, and the core undergoes a high-pressure boil-off with relief valves cycling. If reactor coolant pump seal leaks are considered, the RCS pressure will depend on the size of the leaks. The pressure may still be controlled by the relief valves, or it may approach and fall below the accumulator pressure. As the core uncovered, heat is transferred from the fuel to the metal mass of the primary coolant system through a process of natural circulation. Superheated steam and hydrogen carries heat to structures, including the upper reactor vessel, the hot leg, inlet plenum of the SGs, and the SG tubes. In this specific scenario, the loop seals remain filled with water and full loop circulation blocked. A countercurrent natural circulation is expected during this phase of accident (Figure 7). The $S_2D$ sequence is a small break LOCA with no high-pressure coolant injection. This transient will result in a core boil-off somewhere above the accumulator pressure, with the RCS pressure depending on the size of the break.

## 6. DESCRIPTION AND IMPORTANCE OF NATURAL CIRCULATION DURING SEVERE ACCIDENTS

Three natural circulation flows can be important during severe accidents:

- In-vessel natural circulation
- Hot leg countercurrent flow
- Flow through the coolant loops

Figure 9 illustrates these flows. Each of these flows may be present during high-pressure boil-off transients such as the TMLB sequence mentioned in above section 5.

The primary effect of the natural circulation flows is to redistribute the energy being generated in the core. This energy redistribution will slow the heat-up of the core, which in turn may affect the damage progression or the extent of the core damage. Slowing the core damage would allow more time for systems to be recovered to mitigate or terminate the accident. However, the energy removed from the core will affect the structures in which it is deposited, in both the upper plenum and the coolant loops. The discussions below address the basic phenomena associated with the three types of natural circulation being considered, together with how the transient progression may be affected by the coolant flow. In addition, the importance of natural circulation flows will also be addressed in summary form.

### 6.1 In-Vessel Natural Circulation

In-vessel natural circulation begins when the core heat-up begins. Because the center part of the core is at a higher power than the periphery, the super-heated steam there is hotter and less dense, and a radial density gradient is established. The denser vapor in the outer part of the core tends to flow toward the center, replacing the hot vapor that rises into the upper plenum. This vapor plume rises to the top of the upper plenum, where it is turned radially outward to the core barrel, and then flows back down toward the top of the core. Heat transfer to the structures in the upper plenum cools the vapor, reinforcing the density gradient between the center of the vessel and periphery. The cooler steam reenters the core through the top of the peripheral fuel assemblies. As core uncovering continues and the liquid level drops, the recirculating flow extends farther down into the core. Depending on the axial power profile, the flow may eventually extend to the bottom of the core. The density gradient in the upper plenum also establishes a recirculating flow within the upper plenum. In reality, many natural circulation cells will probably be established in the core, especially during the core damage portion of the transient. These cells will exist between fuel assemblies, between parts of fuel assemblies, and even between subchannels. Wherever a radial temperature gradient exists, a natural convection flow may be established. The rapid increases in local temperature associated with the accelerated oxidation of zircaloy fuel road cladding, at a temperature about 1850 K, will in particular result in the establishment of these relatively small natural circulation cells. These smaller flow cells are assumed to exist within larger overall natural convection cells involving the entire core. It is these core-wide patterns that are being investigated, because it is the global core behavior that is of interest, not the individual fuel rods or assemblies.

The in-vessel natural circulation flow may affect the cladding oxidation. Because steam is being recirculated from the upper plenum back into the core, it is less likely that the oxidation reaction will become steam-starved. The slower cladding heat-up caused by the removal of some of the core energy to the upper plenum structures, combined with steam reach environment, may result in more extensive oxidation of the cladding at lower temperatures. This in turn may lead to smaller amounts of unoxidized zircaloy melting as the temperature increases, which could delay relocation of molten material and cause less dissolution of the fuel pellets.

Increased heating of the upper plenum structures resulting from the natural circulation flow could also lead to their oxidation or melting. This could add to the hydrogen generated in the RCS before vessel failure, and could add more material to the melt that flows from the vessel at the time of lower head failure. Higher vapor temperatures in the upper plenum will also make hotter vapor available to the hot legs. Flow through the hot leg and surge line to the pressurizer PORVs will heat the piping. If the pipe temperatures are high enough, creep rupture failure of these pipes may become a concern. Failure of the RCS piping before the vessel fails could allow the system to depressurize, initiating accumulator injection. If the depressurization continues far and fast enough, the RCS pressure at the time of vessel failure be low enough to preclude direct containment heating.

### 6.2 Hot Leg Countercurrent Flow

Single-phase countercurrent flow in the hot leg is significant in PWRs with U-tube steam generators, because these participate in the natural circulation flow and heat transfer. Natural circulation flow pattern in a hot leg of a PWR during severe accident conditions is shown in figure 9. Superheated vapor enters the top of the hot leg, displacing saturated vapor, which then flows back to the reactor vessel along the bottom of the hot leg. When the hotter vapor enters the steam generator inlet plenum, it will rise toward the steam generator tubes. Vapor enters some of the tubes, displacing the cooler steam that was in the tubes. The displaced vapor enters the outlet plenum, then reenters other steam generator tubes, forcing vapor into the inlet plenum. A density gradient is thus established between tubes. This density gradient then pulls hotter vapor into the tubes, displacing additional cooler steam. The process continues until a steady flow is established, with hot vapor flowing from the inlet plenum to the outlet plenum through some of the steam generator tubes, and cooler vapor returning to the inlet plenum through the remaining tubes.

Mixing in the steam generator plenum is a controlling phenomenon for the hot leg natural circulation flow. It limits the mass flow in the hot leg by increasing the temperature (and lowering the density) of the vapor returning from the steam generator along the bottom of the hot leg. It limits the heat transfer in the steam generator by reducing the temperature of the hot vapor entering the tubes. While accurate modelling of the mixing is important in providing realistic simulation of the hot leg flow behavior, neglecting the mixing in the steam generator inlet plenum in the analyses will yield steam temperature tube temperatures and hot leg mass flow rates that are higher than would be expected.

Hot leg countercurrent flow can affect the structural integrity of the RCS piping. Heating of the pipes and steam generator tubes may lead to melting and creep rupture failure of those components. Steam generator tubes are very thin compared to the loop or surge line piping, and can be quickly heated if exposed to high temperature vapor. Should the tubes fail, a direct path outside of containment (through the steam line relief valves) becomes available to any fission products carried in the coolant. While flow through pressurizer relief valves in some transients will draw hot vapor into the hot leg and surge line, the countercurrent flow may accelerate the heating and possible failure of these pipes. For lower pressure transients in which relief valves are not open, the hot leg countercurrent flow may result in hot leg piping failures that would not otherwise be expected to occur.

Fission product behavior may also be affected by the flow to the steam generators. An extremely large surface area is available on the steam generator tubes for deposition of fission products. If the tubes remain cool, deposited species may remain there and not be released from the RCS. If the tubes continue heat-up so that revolatilization occurs, the flow may simply carry the resuspended fission products to cooler parts of the tubes, where they would again be deposited. The mixing in the steam generator inlet plenum may also play a part in the fission product behavior. The countercurrent flow in the hot leg itself may also affect the fission product transport. If gravitational settling is an important mechanism for fission product deposition in the hot leg, fission products falling from the flow that is heading toward the steam generators would enter the return vapor stream, where they would be carried back toward the reactor vessel, rather than away from it.

### 6.3 Coolant Loop Flow

If the loop seals clear of liquid during a transient with the reactor coolant pumps off and the steam generators removing heat from the RCS, loop natural circulation would be reestablished. In contrast to the natural circulation that occurs following the reactor coolant pump coast down early in a transient, the fluid flowing through the coolant loops would now be superheated vapor. Loop natural circulation flow is a buoyancy-driven one-dimensional flow with heat addition in the core and heat rejection primarily in the steam generators. However, in this situation heat would be transferred to the piping through out the coolant loops. Because of the resulting large vapor density differences and the height of the steam generator, this flow is generally large enough that it disrupts any multidimensional natural circulation flows that might exist in the hot leg or reactor vessel.

The high flow rate and large amount of metal structures available as heat sinks result in a much slower core heat up. The slower heat up rate could result in complete oxidation of the cladding before any of the zircaloy melts. Fuel road relocation would be delayed for several hours. Failure of the piping anywhere in the RCS is possible, although the steam generator tubes would be particularly susceptible because they are much thinner than the hot or cold legs, as already mentioned in section 6.2. Heating of all the piping will also tend to reduce the extent of fission product retention in the RCS.

### 6.4 Importance of Natural Circulation Flows

The significance of natural circulation flow in severe accidents is that it transfers energy from the core to the other regions of RCS. This energy transfer both slows the core heat-up, delaying fuel damage, and increases the temperature of structures elsewhere in the RCS (upper plenum, hot leg and surge line piping, SG tubes, etc.) so they may get hot enough to melt or fail.

Slowing the core heat-up provides additional time for system recovery or operator actions, either of which could terminate the transient by returning the core to a covered, cooled state. The core heat-up rate also affects the composition of the molten core. A slower core heat-up allows more complete oxidation of the fuel rod cladding in place, so that the melt contains mostly ceramic materials. A thick oxide shell builds up in the cladding, preventing relocation until ceramic melting temperatures are reached. A rapid core heat-up allows cladding melt and relocation much earlier, as molten zircaloy dissolves the thinner oxide shell. The relocation material in this case contains more metallic cladding, which can still be oxidized in either the molten pool or in containment through core/concrete interaction.

If the upper plenum structures melt, their mass is added to the core debris that may eventually reach the lower head. If the RCS piping gets hot enough to fail, either through melting or creep rupture, a breach of the RCS pressure boundary results that would allow the system to blow down to the containment. This could significantly alter the containment response at the time of vessel melt-through, in that the dispersion of molten core material is minimized if there is no pressure difference between the RCS and the containment (no DCH). Natural circulation flows may result in unintentional depressurization of the RCS by causing failures of the surge line or hot leg. If the failure location is the steam generator tubes, a containment bypass is established. Fission products released from the fuel can flow through the failed tubes to the secondary side of the SG, and from there through atmospheric dump valves or safety relief valves to the environment, bypassing the containment.

## 7. CONCLUDING REMARKS

Natural circulation plays an important role in the accident progression during small break LOCA. Single-phase natural circulation is generally an affective and dependable means for removing decay heat in PWRs. Nonuniform flow, noncondensable gases, and secondary side conditions can significantly influence the single-phase mode of natural circulation. Although it can be much more complex than single-phase natural circulation, the two-phase natural circulation can provide an adequate method for removing decay heat from the core. Two-phase flow is more tolerant for noncondensable gases than is single-phase natural circulation. But the effects of noncondensable gases can still significantly influence two-phase natural circulation cooling effectiveness. The reflux condensation is also very efficient way of heat removal mechanism.

In all modes of natural circulation, the heat sink must remain active. The effects of noncondensable gases, secondary side conditions, and nonuniform flow behavior do affect, to a certain degree, the different modes of natural circulation.

Natural circulation in severe accidents redistributes energy from the core to structures in the upper plenum and coolant loops. This energy transfer not only slows the heat-up of the core, it also contributes to potential failure of the RCS piping. Such a failure, if it happens early enough in the transient, may allow the RCS sufficient time to depressurize before lower head failure that high-pressure melt ejection will not occur, minimizing the effects of direct containment heating (DCH).

Information from separate effects tests and integral test facilities [8], operating plant data and analysis have made significant contributions to the knowledge base concerning natural circulation cooling in PWRs. Many important natural circulation phenomena have been identified through these studies. It is extremely important that the commonly used thermal-hydraulic system codes together with the CFD computer codes (used for detailed specific phenomena, e.g., mixing and natural circulation at the inlet and outlet of U-tube steam generators, etc.) are able to simulate natural circulation during transients and accidents, because they will be relied upon to predict full-scale plant behavior.

## NOMENCLATURE

| | |
|---|---|
| AC | Alternate Current |
| ADS | Automatic Depressurization System |
| CFD | Computational Fluid Dynamics |
| DCH | Direct Containment Heating |
| ECCS | Emergency Core Cooling System |
| HPIS | High Pressure Injection System |
| LOCA | Loss Of Coolant Accident |
| LPIS | Low Pressure Injection System |
| LWR | Light Water Reactor |

| | |
|---|---|
| PORV | Power Operated Relief Valve |
| PWR | Pressurized Water Reactor |
| RCP | Reactor Coolant Pump |
| RCS | Reactor Coolant System |
| $S_2D$ | Small Break LOCA with no HPIS |
| SG | Steam Generator |
| TMI-2 | Three Mile Island Unit 2 |
| TMLB | Station Blackout Sequence |

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[3] R. B. Duffey, J. P. Sursock, "Natural Circulation Phenomena Relevant to Small Breaks and Transients", Proceedings of Specialists Meeting on Small Break LOCA Analysis in LWRs, Vol. 1, Pisa, Italy, 23-27 June 1985

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[8] N. Aksan "The CSNI Separate Effects Test and Integral Test Facility Matrices for Validation of Best-estimate Thermal-hydraulic computer codes" Lecture to be presented at the IAEA Course on Natural Circulation in Water Cooled Nuclear Power Plants (Paper ID. T16), International Centre for Theoretical Physics (ICTP), Trieste, Italy, 28 June – 2 July, 2004

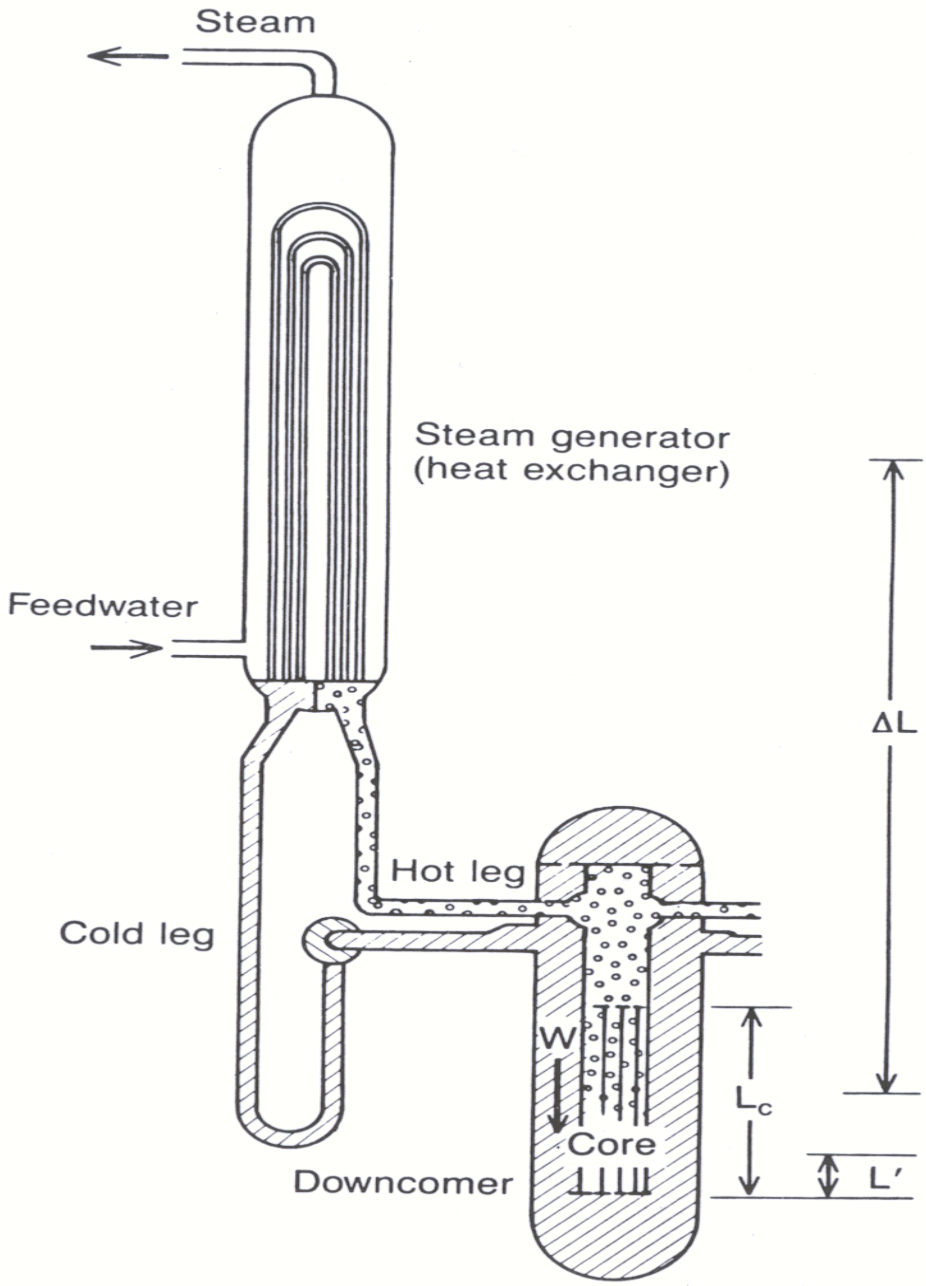


Figure 1: Schematic of a typical PWR system and elevations

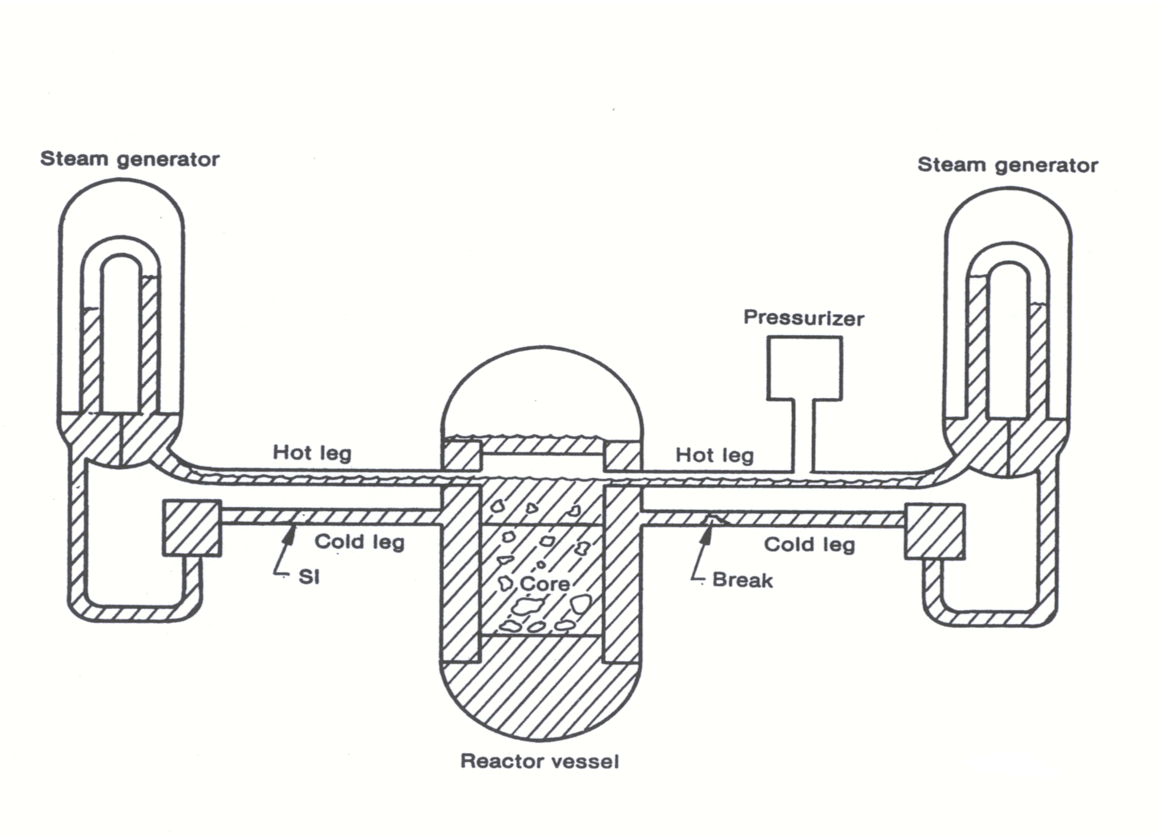


Figure 2: Formation of a continuous vapor phase at the top of the steam generator U-tube and vessel during a small break LOCA in PWR

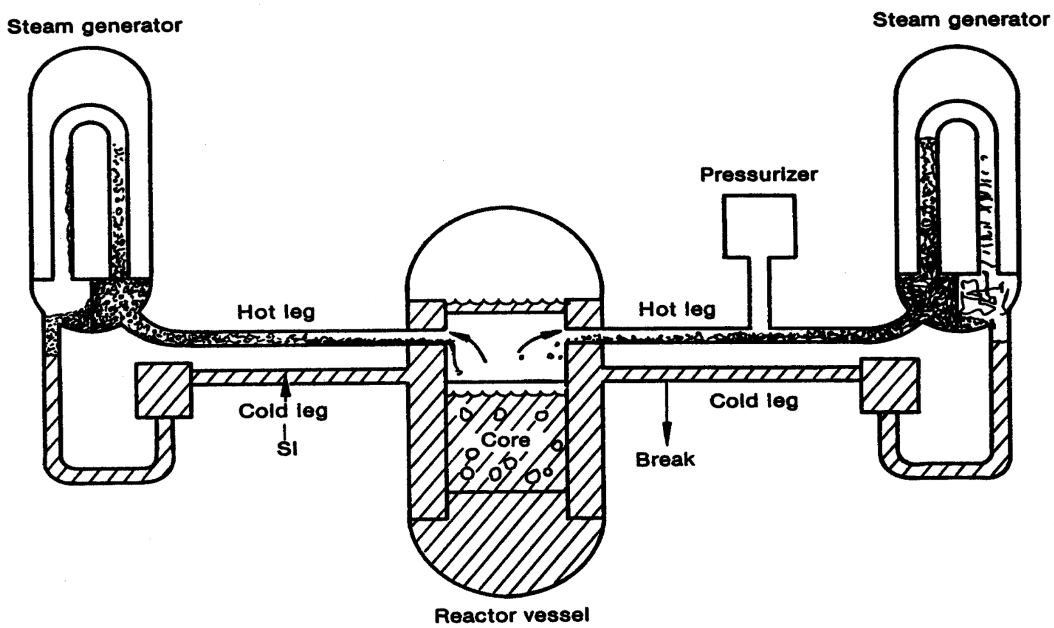


Figure 3: Initiation of core uncovery during a small break LOCA in PWR

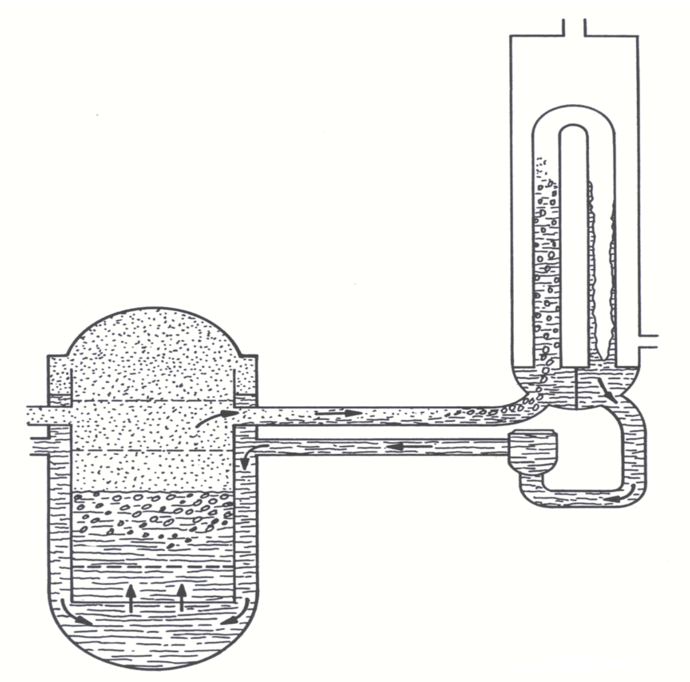

Figure 4: Liquid inventory trapped outside of the reactor vessel due to a loop seal

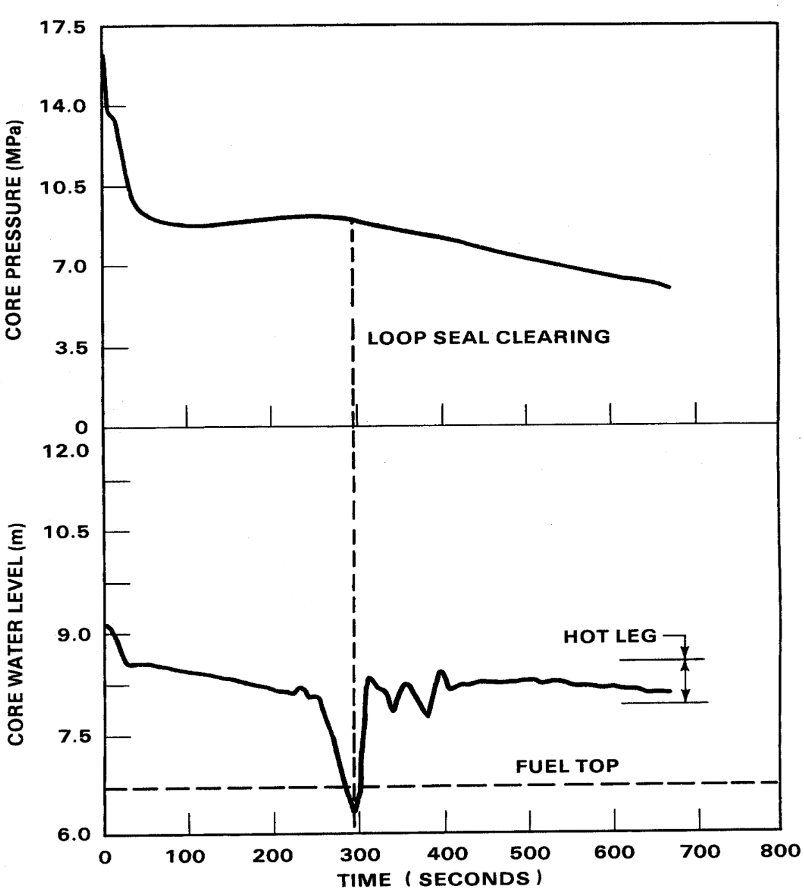


Figure 5: Typical transients of the primary system pressure and reactor vessel water level following a small break LOCA at the cold leg in a PWR

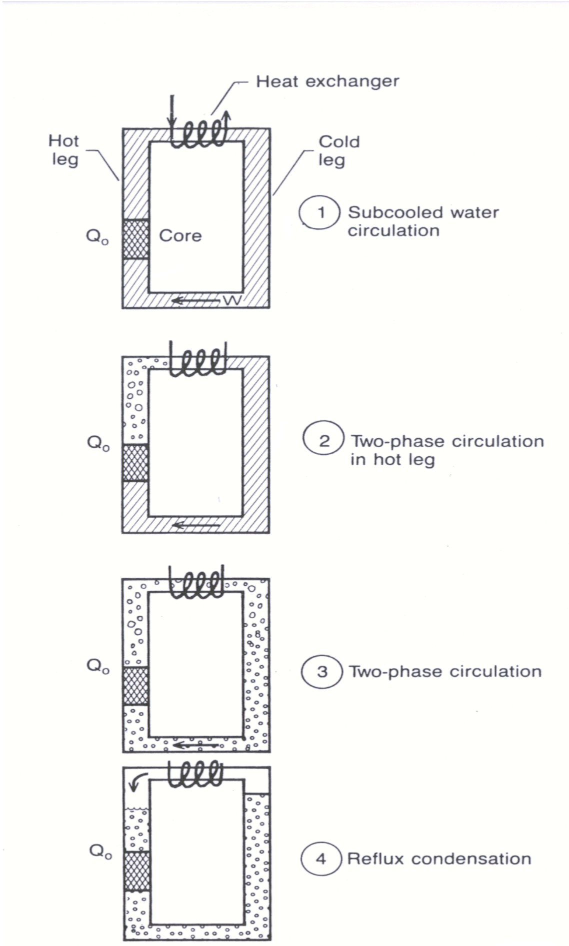


Figure 6: Different modes of natural circulation cooling in a PWR

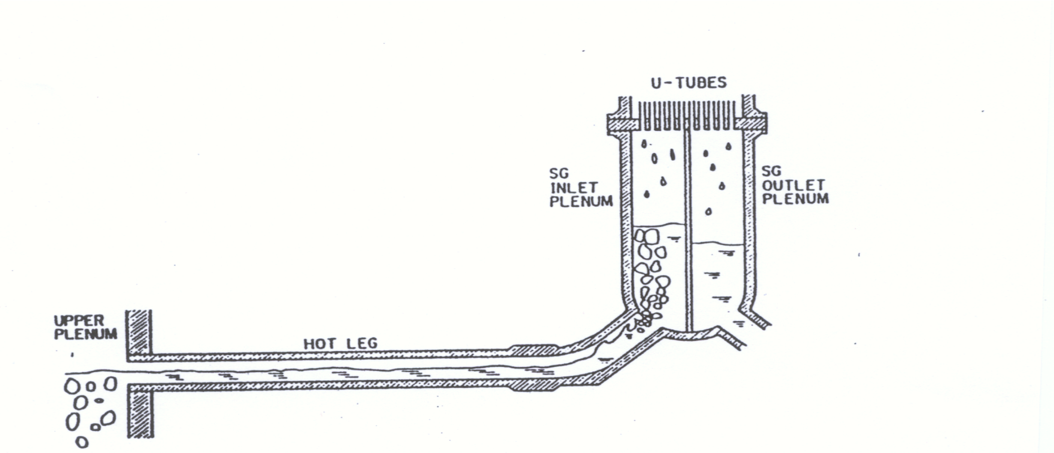


Figure 7: Schematic of liquid distribution during reflux condensation in PWRs

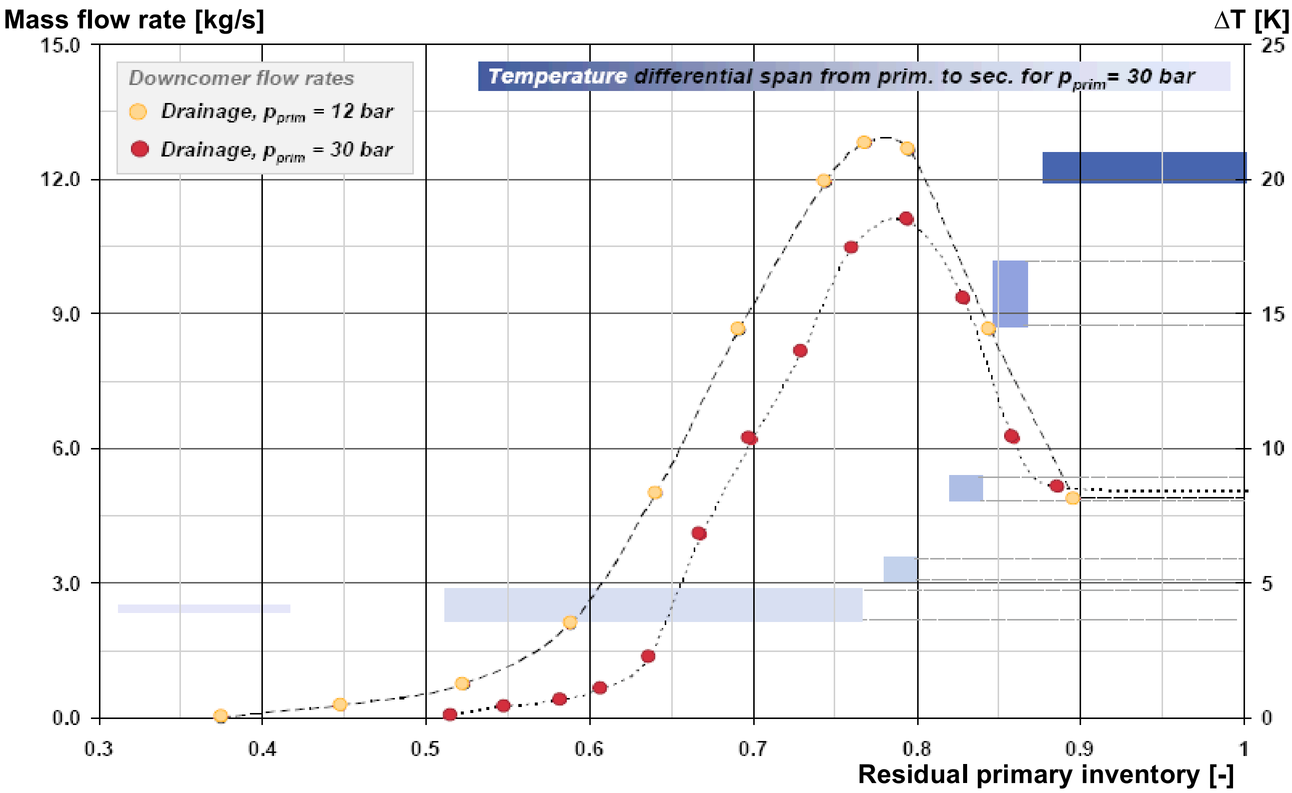


Figure 8: Downcomer mass flow rates during drainage and primary to secondary temperature differences as functions of residual primary mass inventory for PKL-III test facility (Power = 600 kW, Pprim = 12 and 30 bar) [6]

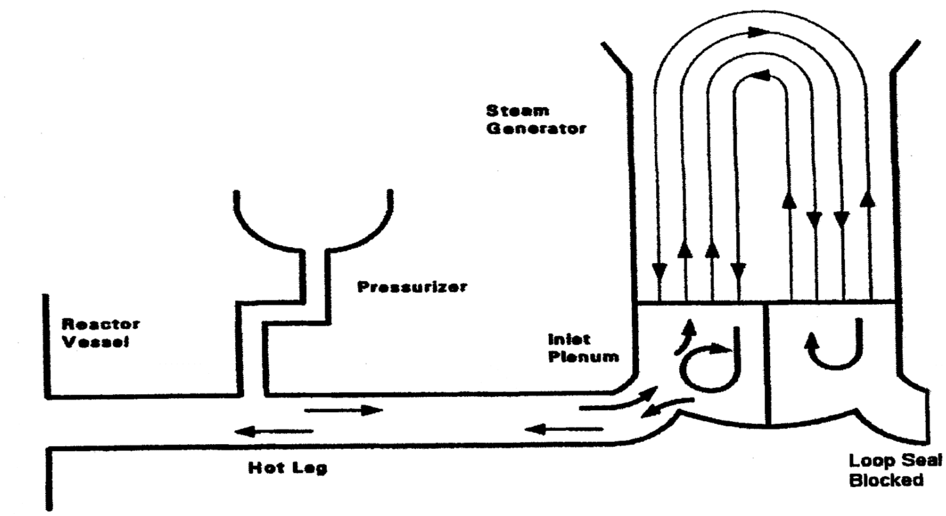


Figure 9: Overview of natural circulation flow pattern in a hot leg of a PWR during severe accident conditions