LOSS OF COOLANT ACCIDENT, LOCA

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INTRODUCTION

The Loss of Coolant Accident (LOCA), is the design basis accident of most reactor concepts. It has different versions for each reactor design and its phenomenology depends on the size of the break.

We consider the small and large-break LOCA phenomenology for the PWR and BWR deigns. An analytical model estimates the rate at which a PWR core becomes uncovered in the case of a small-break LOCA, exemplified by the Three Mile accident.

LOCAs have occurred in light water and heavy water reactors as well as gas cooled and liquid metal cooled ones.

CORE HEAT-UP PHENOMENA

As the temperature increases in the core of a reactor in the analysis of a postulated reactor accident, different physical processes come into play. Their succession as the core temperature increases is shown in Table 1.

Table 1. Phenomena of interest during reactor core heat-up.

Physical phenomenon	Temperature [^oC]
Average cladding temperature during normal operation	350
Cladding is perforated or swells in volume as a result of internal pressure buildup. Some fission gases such as Kr, I and Xe are released. Solid reaction between stainless steel and Zircaloy start. Clad swelling and ballooning could block the coolant flow through some channels.	800-1,450
Steam can react with the Zircaloy cladding producing hydrogen and an energy release that exceeds that from the decay heat. The oxidation process embrittles the Zircaloy. Steel alloys melt.	1.450-1,500
The Zircaloy and steam reaction may become autocatalytic, feeding upon itself, unless the Zircaloy is quenched by immersion in the water coolant.	1,550-1,650
The Zircaloy cladding melts. Releasing of the fission products from the ceramic UO_2 fuel becomes significant above 2,150 K.	1,900
Both UO_2 and ZrO_2 melt.	2,700

These different phenomena lead to the release of different amounts of fission products as well as actinides depending on the reached temperatures in a LOCA accident.

The cumulative percentage fission products radioactive fission products releases are shown in Table 2. The vaporization data assume an exponential loss over 2 hours with a half-time of 30 minutes. If a steam explosion occurs first, only the core fraction not involved in the steam explosion can be vaporized.

Table 2. Cumulative radioactive fission products releases in a postulated PWR LOCA accident. Source: WASH-1400, Reactor Safety Study, 1975.

Fission products	Fuel-clad gap [percent]	Meltdown [percent]	Vaporization [percent]	Oxidation [percent]
Noble gases, Kr, Xe	3.0	90.0	100	90ab
Halogens, I, Br	1.7	90.0	100	90ab
Alkali metals, Cs, Rb	5.0	81.0	100	-
Tellurium, Selenium, Rubidium Te, Se, Rb	0.01	15.0	100	60ab
Alkaline earths Sr, Ba	0.0001	10.0	11	-
Noble metals Ru, Mo	-	3.0	8	90ab
Rare earths La, Sm, Pm Refractories Zr, Nb	-	0.3	1.3	-

a is the fraction of core involved in a steam explosion
b is the fraction of inventory remaining for release through oxidation

HYPOTHETICAL LARGE BREAK LOCA IN THE PWR

The hypothetical large-break LOCA is the classical design-basis accident for the PWR reactor concept.

To describe its phenomenology the following assumptions that represent the worst accident that could be conceived to happen in a water circuit, are usually made:

1. One of the inlet pipes from the circulating pumps is completely non-functional.
2. Free discharge of the primary coolant from both the broken ends. This is referred to as a "double ended guillotine" or "200 percent" break.

The following sequence of events is envisioned to happen in this case:

BLOWDOWN PHASE, 0-20 seconds

1. The large break in one of the cold legs leads to rapid depressurization.

2. A two-phase steam and water mixture is formed by the flashing of the water into steam as the pressure of the coolant is decreased to the saturation pressure.
3. The mass flow rate of the water-and steam mixture is slower than in the case of single phase flow, which leads to slower depressurization.
4. Within 10 seconds, the flow from the High Pressure Injection System, HPIS and Accumulator nitrogen-pressurized tanks is initiated into the Emergency Core Cooling System, ECCS line in the cold leg.

BYPASS PHASE, 20-30 seconds

1. A significant upward flow of steam still exists in the downcomer annulus through which the coolant normally flows.
2. This upward flow prevents the accumulators water from entering the plenum region below the core.
3. The coolant water bypasses the upper part of the inlet plenum and flows out through the break.

REFILL PHASE, 30-40 seconds

1. After the steam pressure is decreased enough, the ECCs water can now ingress into the lower plenum of the core.
2. The Low Pressure Injection System, LPIS is now initiated providing a higher coolant flow rate than the HPIS.
3. The refilling of the lower plenum starts at 23 seconds after the break initiation.
4. The filling of the lower plenum takes about 17 seconds.

REFLOODING PHASE, 40-250 seconds

1. The core would have been totally uncovered and dried out during the blowdown phase.
2. The fuel temperature would have risen rapidly to about 1,000 ^oC.
3. The fuel temperature falls slowly because of the steam, rather than liquid, flow in the core.
4. The fuel elements rupture releasing the fission products to the primary circuit as well as through the break to the containment structure.
5. In the reflooding phase, the fuel elements are rewetted from the bottom of the core to its top.
6. The excess ECCS water is overflowing through the break and is collected in the containment sump.

LONG TERM COOLING, > 250 seconds

1. After sufficient depressurization, the LPIS pumps water at a high flow rate into the cold leg.
2. The formed column of liquid drives the emergency cooling water through the core by natural circulation.
3. Steam may continue being generated in the core.

4. The steam is condensed by the containment spray system, collected in the containment sump, cooled in a heat exchanger, and recirculated for the long term cooling of the core without a containment breach.

STEAM BINDING PHENOMENON

During the rewetting and reflooding phase of the large-break LOCA in the PWR the steam-binding phenomenon can affect the reflooding process and id worthy of a detailed description:

1. As the fuel surface rewets, steam is formed and entrains the liquid droplets before the rewetting front into the upper core plenum.
2. The steam and water droplets mixture passes from the upper plenum through the steam generator, through the circulation pumps and back to the cold leg and out through the break.
3. The water droplets evaporate in the steam generators due to the backflow of water from the secondary side which still contains a hot fluid.
4. The resistance presented by the outflow route generates a backpressure in the upper plenum which restricts the rate at which the reflooding of the core takes place. This phenomenon is known as: "Steam binding."
5. The highest resistance to the flow into the upper plenum caused by steam binding exists when the circulating pump rotor becomes locked stationary if all the water droplets pass to the steam generator.
6. The resistance to the flow is reduced if the water droplets get deposited on the upper plenum structures and thus are not carried out of the pressure vessel, and the circulation pump rotor is still rotating.

HYPOTHETICAL LARGE BREAK LOCA IN THE BWR

This is considered as the design basis accident for the Boiling Water Reactor, BWR design.

It assumes a rupture of one of the pipes connecting the external circulating pump with the reactor vessel.

The BWR LOCA involves a more gradual depressurization than the PWR. The reason is that the main coolant pipe diameter in the BWR is 50 cm in diameter compared with 80 cm in the PWR.

The pipe rupture is followed by the core being uncovered with the jet pumps suction being uncovered.

The feed pump flow stops and the suction of the jet pumps stop, with the core rate of flow dropping to zero.

The water remaining in the lower plenum flashes into steam. Within 10 seconds, the core begins to dry out and increase in temperature.

The loss in moderation shuts down the chain reaction. However, the decay heat would damage the fuel if no supplemental cooling is provided. Steam binding in the core may occur.

As the core spray system is initiated the core gets reflooded to maintain a legislated USA Nuclear Regulatory Commission, USNRC temperature below an upper temperature value of 1,477 K.

SMALL-BREAK LOCA IN THE PWR

Before the occurrence of the Three-Mile reactor accident, most attention in safety analysis was concentrated on the postulated large-break LOCA, which is the design basis accident for the PWR.

The small-break LOCA is defined for piping breaks up to sizes where the reactor remains pressurized despite the occurrence of the break. This encompasses up to 12 cm diameter holes in the primary circuit piping.

At the Three-Mile Island accident, a small break equivalent LOCA was caused by a stuck-open power operated safety relief valve on top of the pressurizer unit. This led to an emphasis on the small-break LOCA as a likely accident deserving of detailed analysis.

In the small-break LOCA, the reactor depressurizes more slowly than in the large-break LOCA, following a different set of physical phenomena.

Since the core remains at high pressure for a long time in a small-break LOCA, it is not possible to activate the Low Pressure Injection System, LPIS, with its relatively large rate of coolant flow, until a late stage into the accident.

Most serious results are noticed when the break occurs in the cold leg bringing the feed water coolant flow into the core.

EMERGENCY CORE COOLING SYSTEM ALTERNATIVES

To mitigate the consequences of a postulated LOCA, alternative Emergency Core Cooling Systems, ECCSs have been used.

In a typical PWR, the ECCS water is injected into the coolant cold leg into the reactor pressure vessel only.

In the German design of a PWR, the ECCS water is injected into both the cold and hot legs. This alternative is claimed to lead to a lower peak temperatures in the case of a large-break LOCA and to a rapid quenching of the core.

In the case of a small-break LOCA, faster depressurization is claimed, allowing for early actuation of the accumulators and Low Pressure Injection System, LPIS with a high volume of low pressure water.

In the newer AP600 and AP1000 designs, warm water is directly injected into the pressure vessel to avoid its loss from a potential pipe break.

PHENOMENOLOGY OF THE SMALL-BREAK LOCA IN THE PWR

Even though the large-break LOCA is considered as the design basis accident for the PWR, the small-break LOCA is more likely to occur, as occurred in the Three Mile accident. This fact makes it worthy of detailed analysis. Its sequence of events proceeds as follows:

1. Following the initiation of the small-break LOCA, depressurization of the primary system occurs, yet at a much lower rate than in the large-break LOCA.
2. The automatic control system senses the pressure drops, inserts the control rods, and shuts down the fission power. The decay heat continues to be released, though.
3. As the pressure falls below 100 bars, the HPIS pumps water at a high pressure but a small flow rate.
4. As the pressure falls below 70 bars, the hottest liquid in the primary circuit flashes into steam.
5. The water in the pressurizer is the first to totally vaporize.
6. Steam bubbles form in the primary circuit and settle in the upper part of the reactor vessel because the pumps would be stopped by the operators. Controversy has arisen over whether the main coolant pumps should be stopped or not during a LOCA. If they are left in operation, they would assist in circulating the liquid and promote the loss of fluid process. The current rule is that the pumps should be stopped. Another reason for stopping them is that they could fail through the vibration caused by pumping a two-[hase steam and water mixture, for which they were not designed.
7. Steam collects in the upper head of the reactor as a result of depressurization and cannot escape.
8. The water coolant leaks quickly and drains to the level of the water inlet and outlet pipes within about 250 seconds. At this stage the high pressure still inhibits the initiation of the accumulators and the Low Pressure Injection System, LPIS with its desirable high coolant flow rate.
9. The steam generators eventually become voided of liquid and filled with steam.
10. The steam formed in the reactor core condenses in the steam generators and flows back in to the core conditional on the secondary side being at low pressure, and at a correspondingly low saturation temperature. For instance, if the secondary side pressure is at 70 bars or 1,000 psia, and the primary side pressure is also 70 bars, no condensation can take place.
11. It is imperative at this stage that the secondary side cooling or depressurization is carried out manually by well trained and informed operators using the power-operated relief safety relief valves at the top of the pressurizer.
12. The core dries from top to bottom while the system is still pressurized. This is the process of "core uncover."
13. The coolant circulation pumps have a pump loop seal in the form of a U bend under it that remains filled with water. The water trapped in the loop seal blocks the steam from flowing from the reactor vessel through the steam generator and the circulation pump to the location of the break. Only when the levels reach the bottom of the U bend, can the steam flow through the loop seal. At this juncture, the steam passes from the core through the pump and out along the cold leg to the location of the break. Only at this stage rapid depressurization occurs.
14. The water in the core vaporizes and a mixture of water and steam bubbles rewets the upper part of the core.
15. As the depressurization progresses the core may be dried out a second time as occurs in the large-break LOCA.
16. The depressurization now allows the activation of the accumulators and the LPIS with its capability to move the coolant at low pressure but a large flow rate.

17. The core is rapidly reflooded and brought to a cold condition.
18. In the longer term for a time larger than 350 seconds after in itiation of the accident, the decay heat is extracted in the same manner as the large-break LOCA.

ANALYSIS OF A SMALL BREAK LOCA IN A PWR

In a small break LOCA in a PWR, the core becomes uncovered at the top of the core and the remaining water continues to evaporate. It is of interest to estimate the rate at which the coolant is evaporating and at which the core is being uncovered.

The volume of water per unit length of the core in the wetted region, for a uniform heat flux distribution, is given by:

$$V_\ell = \frac{\pi D^2}{4}(1-\alpha)(1-\alpha_{fuel})[\frac{m^3}{m}] \qquad (1)$$

where:
- D is core diameter [m]
- α is the mean void fraction in the wetted region
- α_f is fraction occupied by the fuel in the core region

The heat release to the water from the decay heat from the submerged fuel is given by:

$$P_{decay} = P_{th} f \delta \ [W] \qquad (2)$$

where:
- P_{th} is the reactor thermal power [W]
- P_{decay} is the decay heat power [W]
- f decay heat power fraction after shutdown
- δ is the fraction of the fuel submerged

The mass water evaporation rate would be given by:

$$\dot{m} = \frac{P_{decay}}{L} = \frac{\text{Heat release rate [W]}}{\text{Latent heat of evaporation [J/kg]}}[\frac{kg}{\text{sec}}] \qquad (3)$$

The volumetric evaporation rate is given by:

$$\dot{V} = \frac{\dot{m}}{\rho}[\frac{m^3}{\text{sec}}] \qquad (4)$$

where: ρ is the coolant density $[\frac{kg}{m^3}]$

The core uncovery rate is given by:

$$\Omega = \frac{\dot{V}}{V_\ell} = \frac{\text{Volumetric evaporation rate}[\frac{m^3}{\text{sec}}]}{\text{Volume of water per unit length}[\frac{m^3}{m}]}[\frac{m}{\text{sec}}] \qquad (5)$$

EXAMPLE

Let us consider the following situation for a typical PWR:

Thermal power: P_{th} = 3,800 MWth
Half upper half of the core is already uncovered, δ = 1/2
Mean void fraction in the wetted region is: α = 0.50
The fuel occupies 40 percent of the core volume: α_{fuel} = 0.40
Core diameter D = 3.6 m
Core height H= 4 m
System pressure during the core uncover period P = 85 bars
Latent heat of vaporization at 85 bars, L = 1.46 x 10^6 [J/kg]
Coolant density ρ = 713 [kg/m^3]

The fraction of decay heat power after shutdown is summarized in Table 1.

Table 1. Fraction of decay heat power after shutdown.

Decay heat power percentage after shutdown f	Time after shutdown
6.5	1 s
5.1	10 s
3.2	100 s
1.9	1,000 s
1.4	1 h
0.75	10 h
0.33	100 h = 4.17 d
0.11	1,000 h = 1.39 m
0.023	8,760 h =1 y

$$\begin{aligned} V_\ell &= \frac{\pi D^2}{4}(1-\alpha)(1-\alpha_{fuel}) \\ &= \frac{\pi \times 3.6 \times 3.6}{4}(1-0.5)(1-0.4) \\ &= 3.054[\frac{m^3}{m}] \end{aligned}$$

At 1 hr after shutdown:

f = 1.4 / 100

$$P_{decay} = P_{th} f \delta \ = 3800 \times 10^6 \times \frac{1.4}{100} \times \frac{1}{2} = 26.6 \times 10^6 [W]$$

The water evaporation mass and volumetric rates are:

$$\dot{m} = \frac{P_{decay}}{L} = \frac{26.6 \times 10^6}{1.46 \times 10^6} = 19[\frac{kg}{\sec}]$$

$$\dot{V} = \frac{\dot{m}}{\rho} = \frac{19}{713} = 0.0266[\frac{m^3}{\sec}]$$

Consequently, the uncovery rate is:

$$\Omega = \frac{\dot{V}}{V_\ell} = \frac{0.0266}{3.054} = 0.0087[\frac{\text{m}}{\sec}] = 0.87[\frac{\text{cm}}{\sec}]$$

SAFETY DESIGN OF LIGHT WATER REACTORS, LWRs

Two types of computational models are used in the safety design of the Light Water Reactors, LWRs:

1. **EVALUATION MODELS**

These account for various phenomena which are represented by differential equations and use assumptions that are postulated the worst possible conceivable results in postulated reactor accidents.

For instance, it may be assumed that there is no penetration of the Emergency Core Cooling System, ECCS cooling water into the reactor vessel during the blowdown phase of a postulated accident.

These models are specified in the Code of Federal regulation 10CFR20, Part 20, Appendix K. They are required to obtain reactor operating licenses from the Nuclear Regulatory Commission, NRC.

The evaluation computational models are proprietary models developed by the architect engineers and the reactor manufacturers. They are not available in the public domain.

To perform the needed safety analyses, reactor operators have to pay substantial fees for the proprietary owners of the codes to perform the analyses required for licensing and operational needs.

2. BEST-ESTIMATE MODELS

These are public domain codes developed by the NRC with government funds. However, they still need to be licensed by the users at a nominal fee that is sufficient to maintain the codes for use on different computing platform by organizations that have opted to perform that function with NRC approval.

These codes use the best available physical models to simulate the various phenomena.

An attempt is made to calculate the system's behavior on the basis of these models. Such codes include RELAP, SCDAP, RETRAN, and TRAC for the analysis of reactor accidents in one and two dimensions for light water reactors.

The calculation models of two-phase flows are still at a continuing development stage. It would be unsatisfactory to rely on them as a basis of rector safety design, and reliance is weighted towards the evaluation models.

EXERCISE

1. If the water evaporation volumetric rate as a result of decay heat generation in a typical Light Water Reactor, LWR, Loss of Coolant Accident (LOCA) is 0.01 m^3/sec, its effective wetted core area is 3 m^2, and its core height is 4 m.

a. Calculate the core uncovery rate in cm/sec.

b. If the core is half filled with water, estimate the time in minutes for total core uncovery.

REFERENCE

1. George Voelz, "The SL-1 Reactor," Chapter 15, in: Suzan M. Stacy, ed. "Proving the Principle - A History of the Idaho National Engineering and Environmental Laboratory, 1949-1999, DOE/ID-10799, 2000.