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Decay Heat Removal and Operator's Intervention During A Very Small LOCA

Hee Cheon No
Korea Advanced Institute of Science and Technology
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매우 작은 규모의 냉각재 상실 사고 동안 잔열 제거와 운전자의 개입

노 희 천
한국과학기술원
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Abstract

Sample calculations were done for KORI-1 to develop a better understanding of what happens after very small LOCA ($\leq$0.05 ft^2). For a water-side break with the break size larger than 0.006 ft^2, fluid-loss through break exceeds the makeup. If the break size is larger than 0.008ft^2, decay heat can be completely removed through break. Based on these results, it was concluded that KORI-1 is fairly safe for the whole spectrum of sizes in very small LOCA. However, for the reactor with 900 MWe or 1200 MWe, a certain spectrum of sizes in very small LOCA should be carefully considered. In the accident sequence the transition from natural circulation to pool boiling or from pool boiling to natural circulation may be troublesome to the operator or in the safety analysis. Operator's intervention was discussed; primary pump shutoff, HPI pump shutoff, break isolation, and opening relief valve. It was proved that continuous operation of HPI pumps after shutdown will not threaten the integrity of the primary system.

요 약

매우 작은 규모의 냉각재 상실 사고후($\leq$0.05ft^2) 어떤 일이 일어나는 가를 더 잘 이해하기 위해 고리 1호기에 대한 샘플 계산을 수행하였다. 깨진 크기가 0.006 ft^2 보다 큰 사고에 대해서는 냉각재 상실이 보충되는 양을 초과한다. 0.008 ft^2 보다 큰 깨진 크기에 대해서는 잔열은 깨진 곳을 통해 완전히 제거된다. 이와 같은 결과에 비추어 고리 1호기는 매우 작은 규모의 냉각재 상실 사고의 전 영역에 걸쳐 비교적 안전하다고 결론지었다. 하지만, 900MWe 나 1200MWe 를 가진 원자로에 있어서, 어떤 깨진 크기에 대해서는 이 사고가 주의깊게 고려되어야 한다. 자연 순환에서 pool boiling 으로 또는 pool boiling 에서 자연 순환으로 천이할때, 특별히 운전자와 안전 분석에 문제점을 남긴다. Primary pump shutoff, HPI pump shutoff, break isolation, opening relief valve의 운전자 간섭에 대해서도 논의 되었다. Shutoff 후 HPI pump의 연속적인 운전은 primary system의 건전성을 위협하지 않는다는 것이 증명되었다.

1. Introduction

After TMI accident considerable attention has been put to very small break LOCA($\leq$0.05 ft^2 break size). It is inevitable that the operator is not part of the analysis but part of system; approach being taken by NRC in small break analysis is similar to that used for large break and does not consider the various way of operator's intervention.

Due to small energy release through a break steam generators play a very important role in removing decay heat during very small break LOCA. If the steam generators are not available to remove decay heat due to malfunction of auxiliary feedwater system, reactor coolant system repressurization is caused by too small break to expell all decay heat and may seriously limit liquid makeup available from the high pressure injection system (HPI).

To prevent core damage in a small LOCA three essential steps must be accomplished:

1) Insert the control rods
2) Keep the core covered
3) Remove the decay heat from the primary coolant system

The first step, which is automatically initiated, was met in TMI accident. However, the second two, which require operator's intervention for a considerable time, were not met and extensive core damage occurred in the accident. Since there are various ways in which the operator can be involved in this type of accident, it is very important that he should understand what happens in the reactor.

As a result, proper operator response and adequate mitigation can be assured.

2. Sample Calculations

A simple calculation procedure can be helpful in developing better understanding of what happens after small LOCA. The calculations were done for KORI-1. Of interest are the mass flow rate through break, decay heat removal through break, time after break before all decay heat can be removed through break, reactor vessel top plenum drain time, and steam generator drain time.

The blowdown after small break may be divided into three phases.

1) The subcooled blowdown is characterized by a rapid pressure drop to the saturation pressure, which is 1325 psi corresponding to the average coolant temperature 580°F. In practice the subcooled blowdown lasts only a relatively short time.

2) The two-phase blowdown results in either no pressure change or a more gradual pressure decrease. The low break flow rates causes the two-phase blowdown to last a considerably long period of time.

3) The superheated blowdown appears if the temperature of the fluid rises above the saturation temperature after core-uncovery. It is characterized by a more rapid decrease of pressure due to effective energy release through break. If the location of break is at the steam side like the steam dome of the pressurizer, the steam-side blowdown occurs.

For simple calculation the water-side blowdown and steam-side blowdown are considered; the fluid upstream of the break is assumed to be saturated water or steam at 1325 psi, respectively.

Based on Moody's with Moody discharge coefficient (actual flow/Moody calculated flow) of 1.0, the mass flow rate through break is calculated (See Fig. 1.). Considering a single failure criterion, it is assumed that one HPI pump is operated. For a water-side break, if the break area is larger than 0.006 ft^2, fluid lost through break exceed the makeup and then

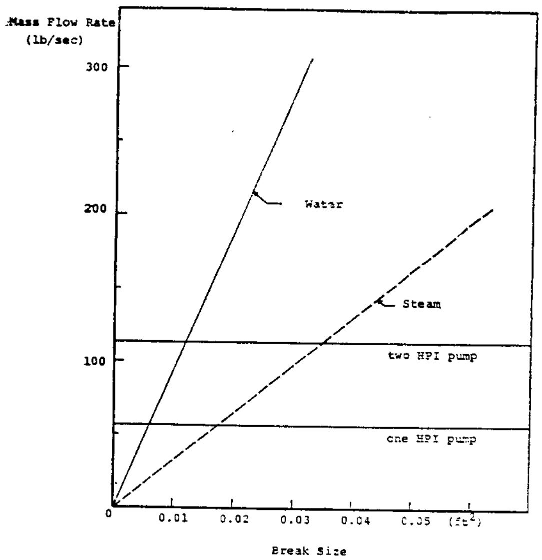

Fig. 1: Mass Flow Rate Through Break with Saturated Upstream Condition at 1325 psia for KORI-1

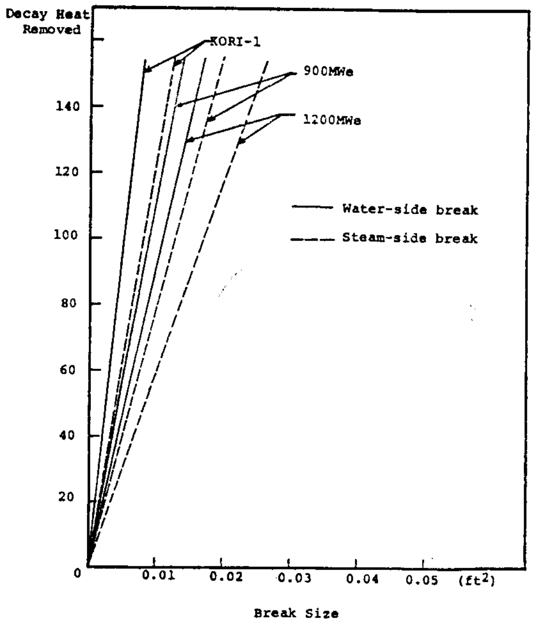

Fig. 2: Decay Heat Removal (2% Rated Power) through Break with Saturated Upstream Condition at 1325 psia

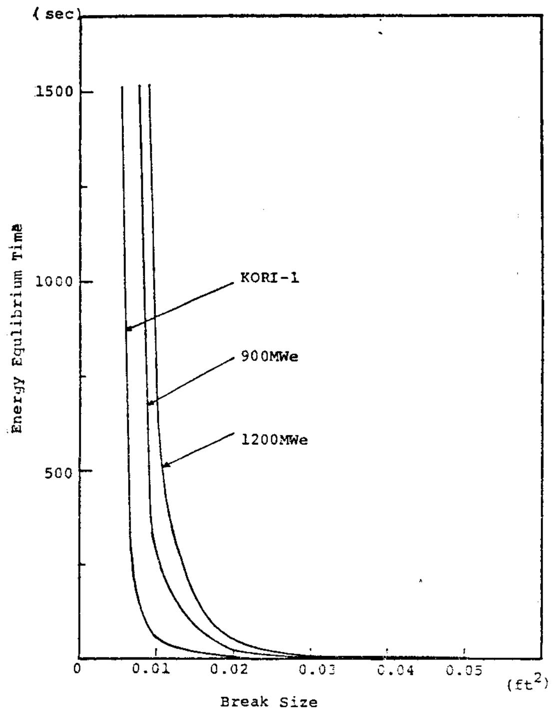

Fig. 3: Time after Break before All Decay Heat Can Be Removed through Break with Saturated Upstream Condition at 1325 psia

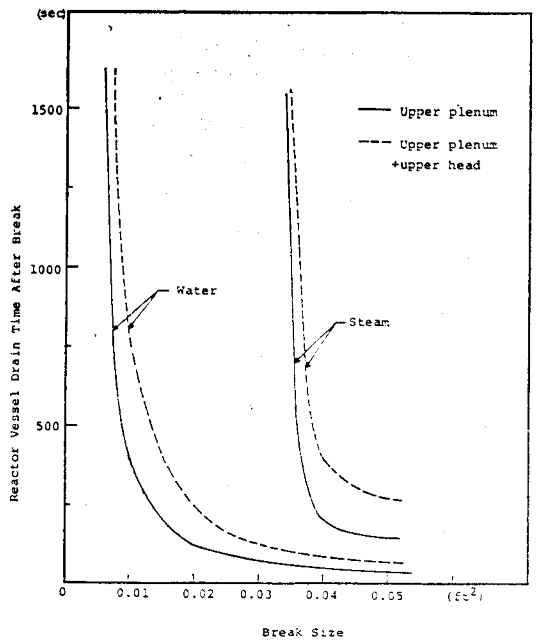

Fig. 4: Reactor Vessel Top Plenum Drain Time with Saturated Upstream Condition at 1325 psia for KORI-1

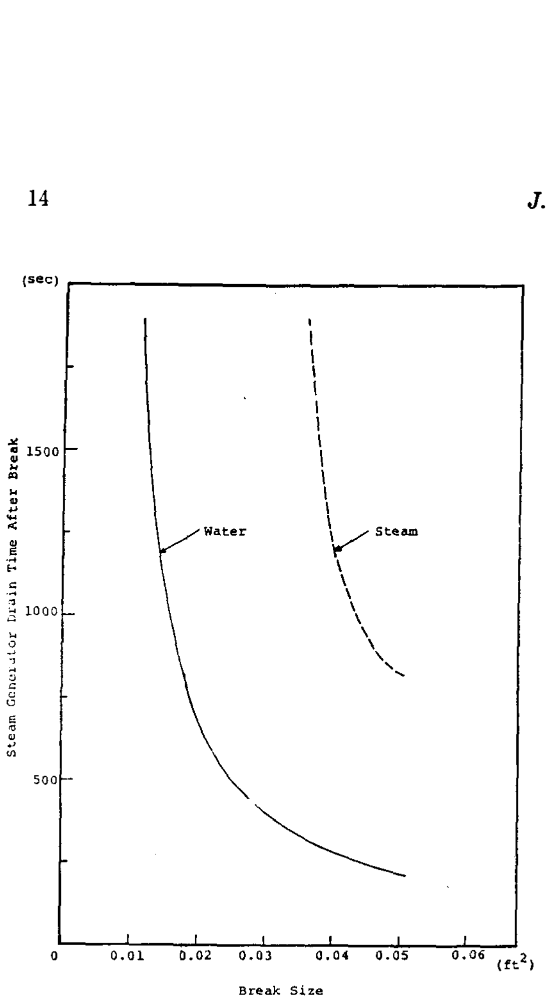

Fig. 5: Steam Generator Drain Time after Reactor Vessel Top Upper Plenum Drain for KORI-1

coolant inventory decreases.

Note that the steam-side blowdown, after the relief valve is stuck open, becomes the water-side blowdown when the steam in the steam dome of the pressurizer is completely expelled. Figure 2 shows the percent of decay heat removed through break. For KORI-1, if the break size is larger than 0.008 ft^2, the decay heat, which is assumed to be 2% of rated thermal power, can be completely removed through break, which is a passive mode to remove decay heat. Based on Figs. 1 and 2, it is evident that KORI-1 is fairly safe for the whole spectrum of sizes in very small LOCA($\leq$0.05 ft^2).

If one HPI pump remains to be operated, three essential steps in Introduction are met. However, if electric power will be increased up to 900 MWe or 1200 MWe, the certain spectrum of sizes in very small LOCA should be carefully considered as shown in Fig. 2.

The time after break before all decay heat can be removed through break is showed in Fig. 3. As the larger the break size is, the shorter the energy equilibrium time is.

The time required to drain the reactor vessel top plenum down to the top of the hot leg pipes is shown in Fig. 4. The pressurizer level is assumed to remain at the level before accident. Figure 4 also shows the drain time,if all water in the upper head falls down into the upper plenum before the level of the upper plenum reaches the top of the hot leg pipes. During this time it is expected that natural circulation may be possible.

Figure 5 shows the time required to uncover the top of the core. If level turnaround in the reactor vessel will not occur up to this time, the superheat blowdown occurs.

3. Accident Sequence

Considering a peak heat flux, 6000 But/hr-ft^2 corresponding to 2% of rated thermal power, it is expected that at this heat transfer rate there is no critical heat flux possible. If any liquid is present around the fuel surface, it can easily cool the surface.

After the loss of offsite power and reactor coolant pump coastdown, two or more of the following phases of operation may be experienced; natural circulation, transition mode, and pool boiling.

(1) Natural Circulation

After reactor coolant pump coastdown natural circulation starts in the primary system. The steam bubble in the pressurizer expands to fill the void created by fluid-loss through break exceeding the makeup, if it is not the steam-side blowdown. Then, the pressurizer level decreases. A steam bubble forms at the highest temperature/lowest pressure location in the primary system. If the secondary side in the S/G is dryed due to malfunction of the auxiliary feedwater system, the initial steam bubble is formed in the U-tube region and it can interrupt natural circulation. Then loss of circulation flow causes to boil aggressively water in the core and to increase reactor pressure. As a result, the steam bubble in the U-tube region is compressed and condensed, and natural circulation is restored. The restoration of natural circulation reduces system pressure until a new steam bubble is formed in the U-tube region.

Once formed, a steam bubble in the reactor vessel grows larger to accomodate the net blowdown of fluid from the system.

For a water-side break the steam bubble is in equilibrium with a reactor vessel pressure.

For a steam-side break like a stuck-open relief valve the entire steam bubble in the pressurizer is removed quickly through break. Then the pressurizer becomes solid and changes to the water-side break.

One of the causes of the extensive damage done at TMI was the fact that the operator turned down the HPI system to maintain a steam bubble in the pressurizer. At full load power the pressurizer safety valves are not large enough to keep the system pressure from increasing to the design pressure of the primary system, if the pressurizer is solid. For a transient, in which the reactor is tripped, it is certain that the safety valves are able to keep the system pressure below the set point of the safety valves in a worst case, even if the pressurizer becomes solid. Appendix I shows sample calculation for KORI-1 that for a decay heat level of 6-7% of rate power immediately after shutdown, the relief valves are large enough to prevent the system from overpressurizing.

(2) Transition from Natural Circulation to pool Boiling

After the level in the reactor vessel reaches the top of the hot leg pipes, steam through the pipes accumulates in the U-tube region and natural circulation ceases, until level turnaround occurs. Water in the core starts to boil agressively due to loss of circulating flow. Repressurization occurs until energy removed through break exceeds decay heat. In this mode the steam generators lose their capabilities to remove decay heat until the refluxing mode, which is one of effective ways to remove decay heat, is established in the steam generators. In the refluxing mode, steam through tubes is condensed and condensed water is falling down to the hot leg pipes.

The transition from natural circulation to pool boiling may be troublesome to the operator for the steam drain time shown in Fig. 5. The drain time represents the minimum time during which system repressurization will occur if all decayheat is not being removed through break.

Since the transition mode is usually neither identified nor implemented in the analysis of small LOCA, the proper operator response has not been investigated. Also this will result in a lower ultimate core level and a higher peak clad temperature if the core is uncovered, than the present ECCS type analysis.

(3) Pool Boiling

Decay heat removal is accomplished, partly by condensation inside the steam generator tubes if the primary side water level is sufficiently below the secondary side water level.

It is well-known that the small amount of non-condensible gases in steam substantially reduces heat transfer coefficients.It appears that for any plant even under normal operating conditions, there is enough non-condensible gas present so that this mode of cooling is uncertain.

Putting vents on the primary system would provide a way of removing non-condensible gas from the locations where it is most likely to accumulate.

If all water above hot leg pipes is drained, water-line break becomes steam-line break and the makeup can exceed fluid-loss through break. The level turnaround in the reactor vessel occurs.

If fluid-loss through break still exceeds the makeup, the reactor vessel level continues to drop until the upper portion of the core has been uncovered. Then the superheated blowdown starts and a rapid decrease of pressure occurs as superheated steam is expelled through break. The reactor vessel level turnaround may occur due to the decrease of pressure. We have to be sure if the maximum cladding temperature should not exeed 2200°F when the level turnaround starts.

(4) Transition from Pool Boiling to Natural Circulation

After the level turnaround hot leg pipes and steam generator tubes are refilled with water. If sufficient non-condensible gases are present, it will be impossible to refill the U-tube region and establish natural circulation.

If there is no venting system in the primary system, the only way to establish natural circulation is to turn on pumps. Then the non-condensibles accumulated in the U-tube region would be trapped in bubbles in the circulating liquid and carried around.

4. Operator's Intervention

The actions, which the operators can take to control and mitigate initial thermal-hydraulic transients, are primary pump shutoff, HPI pump shutoff, break isolation, and opening relief valve.

As said before, continuous operation of HPI pumps after shutdown will not threaten the integrity of the primary system, even if the pressurizer becomes solid. Therefore, the operator should not turn off HPI pumps before he assures that break isolation is completed. If, after the transition from pool boiling to natural circulation, he is sure that there is no controlling steam bubble in the reactor vessel with level detectors, HPI pumps can be shutoff for the establishment of the safe shutdown mode.

It may be a requirement of the emergency operating procedure to isolate a letdown line automatically or manually and a break if it can be located and valved out. Sometimes it takes time for break to be located. In the transition mode in which there is no effective way to decay heat removal except break, it is not wise to isolate break. However, the operator should remember that the prevention of core-uncovery threatening cladding integrity takes priority.

Primary coolant pump operation strongly affects the mass inventory in the primary coolant system. For the pumps being off or left running, the mass depletion in the primary coolant system was found to be significantly higher for the case where the pumps were left running. This result supports an early pump trip for some small LOCAs.

Many non-LOCA events initially manifest the same symptoms as a small break LOCA, i.e., system depressurization, reactor trip, and SI initiation. Tripping the primary pumps for non-LOCA transients can aggravate the consequences of these transients. Therefore, the signals and criteria designated to initiate the pump trip should be carefully chosen.

However, it is advisable that primary pumps are operated intermittently after initial primary pump shutoff, if natural circulation is interrupted by the accumulation of the non-condensibles and repressurization occurs in the transition from pool boiling to natural circulation, and if the thermocouples indicate the cladding temperature up to 2200°F in a worst case. The operator should check whether the loop seal pumps are operated.

Operator action may be invoked to open the pressurizer relief valve to assure continuation of the more stable mode of pool boiling for decay heat removal or provide sufficient depressurization to go on shutdown cooling. However, this valve has not been qualified to perform an essential mitigating function and can be inactivated by a postulated single failure.

5. Conclusions

The followings have been identified as special items of concern during a very small LOCA.

1) The transition mode may be troublesome to the operator. In this mode the steam generators lose their capabilities to remove decay heat.

2) The transition mode is usually neither identified nor implemented in the analysis of small LOCA. Also the proper operator response in this mode has not been investigated.

3) Existence of non-condensible gas deteriorates the heat transfer characteristic of condensation and interrupts natural circulation.

The proposed proper operator actions to mitigate or control accidents are as follows:

1) Primary pump are operated intermittently after initial primary pump shutoff, if natural circulation is interrupted by the accumulation of the non-condensibles and if the thermocouples indicate the cladding temperature up to 2200°F in a worst case.

2) The operator should not turn off HPI pumps before he assures that break isolation is completed.

3) In the transition mode it is not wise to isolate break.

4) The operator should remember that the pressurizer relief valve may not be qualified to perform an essential mitigating function.

Appendix I

Let us compare the volume generation rate by decay heat and two HPI pumps with the volume removal rate through the relief valves and safety valves, in KORI-1. Vapor volume generation rate, $\dot{V}_v$ in the core at x fraction of rated power (Q_0) and system pressure 2300 psia

$$\dot{V}_v=\frac{xQ_0 v_{fg}}{h_{fg}}=502.21x\text{ft}^3/\text{sec} \qquad (\text{A. }1)$$

Liquid volume generation rate, $\dot{V}_l$ by two HPI pumps

$$\dot{V}_l=1.808\text{ft}^3/\text{sce} \qquad (\text{A. }2)$$

Using the capacities of the relief valve (179, 000 lb/hr) and safety valve (380, 000 lb/hr) in KORI-1, we can obtain the total volume removal rate ($\dot{V}_r$) through them

$$\dot{V}_r=51.46\text{ft}^3/\text{sec} \qquad (\text{A. }3)$$

From Eqs. (A. 1), (A. 2), and (A. 3) we get the critical fraction (x_c) of rated power for KORI-1 at which the total volume generation rate by decay heat and two HPI pumps is equal to the total volume removal rate:

$x_c=0.1$

Since this fraction of rated power can be reached immediately after reactor trip, it can be said that the continuous operation of two HPI pumps in the small LOCA of KORI-1 will not threaten the integrity of the primary system, even though the pressurizer becomes solid. For 1200 MWe-Reactor with the same capacities of the relief valves, safety valves, and HPI pumps as KORI-1,

$x_c=0.047$

It can be achieved within 15sec after reactor trip. Therefore, the operator should operate continuously HPI pumps after he assures that 15sec after reactor trip elapses when the pressurizer becomes solid.