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CONSIDERATIONS FOR REALISTIC ECCS EVALUATION METHODOLOGY FOR LWRs

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ABSTRACT

This paper identifies the various phenomena which govern the course of large and small break LOCAs in LWRs, and affect the key parameters such as Peak Clad Temperature (PCT) and timing of the end of blowdown, beginning of reflood, PCT, and complete quench. A review of the best-estimate models and correlations for these phenomena in the current literature has been presented. Finally, a set of models have been recommended which may be incorporated in a present best-estimate code such as TRAC or RELAP5 in order to develop a realistic ECCS evaluation methodology for future LWRs and have also been compared with the requirements of current ECCS evaluation methodology as outlined in Appendix K of 10CFR50.

INTRODUCTION

Emergency core cooling system (ECCS) provides the means for nuclear power plants to mitigate the consequences of loss of coolant accidents (LOCAs). Currently, two very different approaches are used to evaluate the performance of ECCSs during design base LOCA. In one approach, a set of guidelines established by the U.S. Nuclear Regulatory Commission (USNRC) (stated in Appendix K of 10CFR50) are followed. These guidelines are based on the understanding in the early seventies of fuel behavior and two-phase thermal hydraulics. These guidelines have built-in conservatisms such as, high decay power, subtraction of the injected coolant during the blowdown phase from the primary system inventory, permitting the return to nucleate boiling only in the reflood phase, etc. This approach is used for licensing purposes. Various recent studies have shown that this approach is very conservative and many of these guidelines are not physically realistic.

The second approach is based on the latest understanding of fuel behavior and two-phase thermal hydraulics. It is in the form of best estimate codes such as, TRAC-PF1/MOD1, TRAC-BD1/MOD1, RELAP5/MOD2 and RETRAN-03. These codes provide realistic simulation of plant conditions during the transient. A realistic ECCS methodology impacts the overall design of a nuclear plant in terms of capital cost, operating cost and plant complexity. Using a realistic ECCS methodology can potentially save $50-100 million per future LWR plant. This methodology can also be used by existing plant owners to gain significant operational flexibility.

This paper describes the various phenomena which govern the course of LOCAs. A review is presented of the current best-estimate models and correlations in the NRC funded codes (TRAC-PF1/MOD1 (1), TRAC-BD1/MOD1 (2), COBRA/TRAC (3) and RELAP5/MOD2 (4)). Finally, a set of models is recommended which can be incorporated in a best-estimate code to create a realistic ECCS evaluation methodology.

PHENOMENA GOVERNING LARGE BREAK LOCA

Large Break LOCA in PWRs

Various simulations of large break LOCA in Pressurized Water Reactors (PWRs), both through experiments in scaled facilities such as LOFT (Loss of Fluid Test) and Semiscale and through code calculations, have shown that this is a continuous transient consisting of three distinct phases. These are: (a) blowdown, (b) refill, and (c) reflood. Each of these periods is governed by different dominant phenomena. The key parameters which characterize these stages of accidents are the timing of the end of blowdown, the beginning of reflood and quench, and the timing and the magnitude of the peaks during clad temperature history in the blowdown and reflood.

Blowdown Phase

It is assumed that the accident is initiated by a double-ended guillotine (200% area) pipe break in the cold leg of a PWR at full power condition. Initially, the break flow rate is large due to subcooled water in the system. The break flow rate even exceeds the flow in the intact cold legs leading to flow reversal at the core inlet. This leads to flow stagnation and critical heat flux (CHF) conditions inside the core. The clad heats up as the clad-to-fluid heat transfer decreases. However, this situation lasts for only 2 to 3 seconds as the system pressure decreases to saturation pressure and the break flow decreases. The intact cold leg flow now exceeds the break flow rate and the core inlet flow is restored, which may result in partial or core-wide quenching. This sequence of events causes the first peak in the clad temperature history. The core inlet flow oscillations due to two-phase loop conditions lead to the second heat-up of the clad. This time a quench may occur due to the flow reversal at the core outlet. The system pressure continues to decrease, and when it reaches the set points for various components of the ECCS, injection of cold water into the primary system begins. This may cause significant oscillations due to steam condensation on cold water. However, there is no net gain in the vessel inventory for some period as most of the injected liquid is bypassed through

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the break due to reverse steam flow in the core. The blowdown phase ends when the system pressure is close to the containment pressure and the break flow is negligible.

The important parameters of this part of the transient are the heights of the two peaks in the clad temperature history and their timings.These parameters depend upon the initial stored energy in the fuel, fuel and clad thermal properties, fuel-to-clad gap size and conductance, the heat transfer coefficient between the clad and the fluid, core power, and radial and axial peaking factors which depend on the life of the fuel in the fuel cycle. Other phenomena which indirectly influence the system response are the break flow rate, single- and two-phase pump performance, and the interfacial heat, mass and momentum transfer. The flooding or CCFL (countercurrent flow limitation) phenomenon is also important as it controls the ECC bypass and water penetration into the reactor vessel downcomer.

In a recent study (5) performed at Brookhaven National Laboratory (BNL) using TRAC-PD2/MOD1, it was concluded that the peak clad temperature in the blowdown phase was not very sensitive to the break flow rate but it was significantly influenced by the initial fuel condition and power distribution in the core. This confirmed the conclusion of a previous study [6] conducted using the RELAP4/MOD6 code.

Refill Phase

The Refill phase begins when there is very little loss of fluid through the break or there is a net downward flow of water in the downcomer. In this period of the accident, which generally lasts for approximately 20 seconds (5), the ECC water starts to reach the lower plenum and there is usually a counter-current flow in the downcomer. The steam upflow in the downcomer is relatively small due to lower reactor power and high steam condensation in the downcomer. This facilitates the downward flow of the injected liquid in the downcomer. The refill stage ends when the lower plenum is reasonably full with water (e.g., liquid volume fraction exceeding 0.95, Rohatgi (5)). The clad continues to heat up during this period, although at a lower rate because the power generated in the core is decaying. The energy added by metal-water reaction and pump power is relatively insignificant, during this period. However, the metal water reaction is important in current ECCS evaluation methodology. The important phenomena in this stage of accident are the steam condensation in the cold legs and downcomer, the downcomer wall heat transfer to the fluid, the interfacial momentum transfer or slip, the entrainment and countercurrent flow limitation (CCFL) or flooding. This is the most difficult part of the accident to calculate, not only due to involved multi-dimensional thermal-hydraulic phenomena, but also due to the geometry of the downcomer annulus, multiple cold and hot leg connections, etc. No study has been performed, to our knowledge, to rank these aspects in terms of their impact on the duration of refill and the peak clad temperature in the reflood phase.

Reflood Phase

The reflood phase is the last phase of the accident in which final core-wide quench is achieved. The liquid rises into the core region from the lower plenum, and the region below the quench front enters into the nucleate boiling regime. This results in rapid cooldown of the core and a large axial temperature gradient in the clad across the quench front. The core region above the quench front will still be either in the inverted annular flow regime (for high flooding rate) or the dispersed droplet regime (for low flooding rate).These regimes have poor heat transfer characteristics, but may contribute to some clad cooldown ahead of the quench front. The peak clad temperature in this phase generally occurs in the middle section of the core where axial power peaking is normally expected. In the best-estimate calculations performed so far, the clad temperature in the blowdown phase has been calculated to be higher than the clad temperature in the reflood phase (5). The void fraction and temperature distribution were quite asymmetric. A multidimensional analysis, is preferable to account for the interaction of core power distribution with appropriate fluid conditions.

The phenomena which govern the reflood phase of the transient are the condensation in cold legs, heat transfer regimes in the core specially the minimum stable film boiling temperature, the flooding rate at the core inlet, the entrainment, the transition between the inverted annular regime and the dispersed droplet regime, and finally the axial conduction in the clad. There has been no study, to our knowledge, to rank these parameters in terms of their influence on the peak clad temperature and quench time.

In PWR plants equipped with upper head injection (UHI), the initial quenching occurs due to flow from the upper head to the top of the core through the guide tubes. However, this additional coolant vaporizes and maintains higher pressure in the core region, and thus prevents the liquid from entering the core and delays the start of reflood. However, as the clad is generally cooler due to early quenching, the reflooding occurs rapidly. A study by Guidotti and Thurgood (7) has shown the various events and phenomena which can be expected during a large break LOCA for PWRs with UHI. The important phenomena for large break LOCA in PWRs with and without UHI are quite similar.

Large Break LOCA in BWRs

A LBLOCA transient in a Boiling Water Reactor (BWR) can also be partitioned into three stages: (1) blowdown and window period, (2) lower plenum flashing and (3) refill/reflood.The following description of these events is supported by the observations in the TLTA (two loop test apparatus) (8) and SSTF (steam sector test facility) (9) test facilities, and the TRAC-BD1 BWR/6 LBLOCA calculation (10).

Blowdown/Window Period

Initially the reactor (a BWR/6) is assumed to be in steady-state operation with the water level in the upper downcomer and separator regions. A double-ended break is postulated to occur in one of the recirculation loops at the pump suction. The reactor system starts losing coolant through the break.Subsequently there is flow reversal in the broken loop jet pumps. The feedwater and recirculation pumps are also tripped. The water level in the downcomer drops and flow through the jet pumps begins to decrease. This leads to reduction in the flow at the core entrance, and possible dryout in the core channels. However, there will also be reduction in core power due to additional voids which will reduce the clad heat-up rate. Initially, the syst . pressure decreases slowly as the volumetric loss through the break is small. This lasts until the downcomer water level uncovers both the jet pump and the recirculation line suctions. This can be considered as the end of the blowdown phase. The next phase is characterized by a rapid decrease in the system pressure due to the large volumetric flow through the break.

The lower plenum pressure approaches the saturation pressure, and the core inlet flow rate is still small. This time period in the accident sequence is called the "Window" period. It lasts only 2-3 seconds.

The important phenomena in this first phase of the accident are the break flow rate, pump coast-down, fuel stored energy, fuel properties, initial core power, and the wall heat transfer.

Lower Plenum Flashing Period

As the lower plenum pressure decreases below the saturation pressure, the liquid in the lower plenum flashes. This results in a rapid expansion of the lower plenum fluid, and a two-phase mixture is pushed into the jet pumps, core channels and also to the bypass region through the guide tubes. This increases the flow at the core inlet and probably causes some early rewet.

As the system continues to lose mass through the break, the water levels in the core channel and bypass region also decrease. However, when the mass loss from the lower plenum exceeds the mass gained from the core channel and bypass region, the side entry orifices (SEOs) are uncovered. The water level in the lower plenum is still above the jet pump discharge so that the steam has a low resistance path through the core channels instead of through the jet pump. This results in a large steam flow and subsequent CCFL at SEOs. Furthermore, there will also be CCFL at the upper tie plate due to steam flow from the lower plenum and additional steam generation in the core channel. Therefore, only small amounts of liquid will be flowing down to the lower plenum and guide tubes. During this stage of the accident, the HPCS will start injecting ECC water into the upper plenum. As the system pressure drops below ~250 psi, the LPCS will also start injection. These events mark the end of the lower plenum flashing period.

The important phenomena during the lower plenum flashing period are the lower plenum flashing rate, core inlet flow rate, core heat transfer including quenching, level tracking in the lower and upper plena, and CCFL at the upper tie plate and SEOs.

Refill/Reflood Phase

This phase can be assumed to begin when the system pressure is around 150 psia. At this time all the ECCS should be operational and delivering cold water into the upper plenum, which might have been almost empty as the upper plenum water had drained into the bypass and some of the core channels. These highly subcooled safety injection fluids mix with the existing steam or two-phase mixture resulting in steam condensation and further decrease in the system pressure. The two-phase mixture generated in the upper plenum flows down to the bypass region without any restriction. A CCFL condition has never been observed between the upper plenum and the bypass region in any multichannel test facility such as SSTF. The LPCI also contributes subcooled liquid to the bypass region inventory which not only condenses vapor but also assists in cooling the channels. The bypass region quickly fills up, and continues to inject liquid into the core channels through the leakage paths. Some of the core channels may also receive some liquid from the upper plenum in spite of CCFL at the upper tie plate, while some liquid may drain into the lower plenum even though a CCFL condition exists at the SEOs.

The upper plenum also accumulates some high density two-phase mixture, after the bypass is filled up. This upper plenum inventory has a significant influence on the transient. As the water level increases, it covers the sparger, and creates two conditions. First, the condensation decreases. Second, subcooled liquid accumulates at the periphery which cools the two-phase mixture near the upper tie plate and causes a breakdown of CCFL through the peripheral channels.The liquid begins to flow down in these channels to the lower plenum. This causes the upper plenum water level to recede below the sparger, and the spray again mixes with the bulk vapor in the upper plenum and causes higher condensation. The temperature distribution is more uniform and the core-wide CCFL conditions are again restored. As the water level increases and covers the sparger, the cycle repeats itself.

During this period of the accident, strong multidimensional effects along with parallel channel effects have been observed. Initially, when the upper plenum has low water level, all channels are in a counter-current flow regime.They are receiving liquid from the bypass region through the leakage path and from the upper plenum with CCFL at the upper tie plate. All the channels are subjected to the same pressure drop. However, this situation does not last, as there is a breakdown of CCFL in the peripheral channels due to the changing upper plenum conditions resulting in a downward co-current flow. The other effect is on the high power (or central) channels, which are generating more vapor than the average channel which controls the pressure drop. Both of these types of channels are receiving liquid from the bypass region and the upper plenum, and are filling up. However, the central channels require a higher elevation of the mixture level to maintain the same core pressure drop.They fill up quickly and are no longer able to maintain the same pressure drop resulting in a transition to high void, co-current upward flow regime. These variations in fluid conditions in different channels are called parallel channel affect and details are given elsewhere [11]. The peak clad temperature occurs in this phase of the transient and clads are eventually quenched between 130 and 150 seconds when all channels are filled up with high density mixture.

The important phenomena in this phase are the CCFL at upper tie plate (UTP) and side entry orifices (SEO), parallel channel effects, vapor generation, multi-dimensional effects in the upper plenum and bypass region, and the wall heat transfer including quenching.

There has been no detailed and systematic study, to our knowledge, to rank the various phenomena in terms of their influence on the peak clad temperature during a BWR LBLOCA. A limited sensitivity study has been performed at INEL [10] for a BWR/6 LBLOCA using the TRAC-BD1 code.

LBLOCA with Additional Failure

The current ECCS methodology for licensing requires successful cooling despite some additional equipment failure and under conservative requirements. These include a lower containment pressure, availability of only one train of ECCS, lower initial inventory of accumulators and loss of power to the reactor coolant pumps (PWR) and recirculation pump (BWR). These additional requirements adversely affect the amount of ECC water delivered into the reactor system. Consequently, they affect the timings of refill, reflood, quenching, and peak clad temperature in the reflood phase. Since these considerations delay the quenching and reduce the ECC liquid, the clad temperatures are generally higher. Thus, other phenomena such as metal water reaction and clad deformation may become important. The possible flow blockage due to clad ballooning must be accounted for in the licensing calculation. The new ECCS evaluation methodology should account for these additional phenomena which were not important in the best-estimate or optimistic sequence of events assuming no equipment failure.

REALISTIC MODELS AND CORRELATIONS FOR LARGE BREAK LOCA

Simulation of a large break LOCA requires modeling of at least the important phenomena described in the previous section. There are many best-estimate models and correlations available for these phenomena in the present literature, particularly in the thermal-hydraulics area. Some of these models are already incorporated in the NRC best-estimate codes, such as TRAC-PD2, TRAC-PF1/MOD1, TRAC/BD1, TRAC-BD1/MOD1, COBRA/TRAC, and RELAP5/MOD2. A comparison of pertinent models used in the TRAC-BD1/MOD1, TRAC-PF1/MOD1, COBRA/TRAC, and RELAP5/MOD2 codes is presented in (11).

Based on a review of the present literature (including the advanced codes), it is now possible to outline realistic models for most of the phenomena important during a large break LOCA. In some cases, several models should be assessed before the final selection is made.

Initial Fuel Stored Energy

The initial stored energy in the fuel rods depends upon the temperature distribution and thermal conductivity of fuel and gap. The gap conductance depends upon the fuel-to-clad gap. Analysis of various cycles of fuel life should be undertaken to establish realistic stored energy, using codes such as FRAPCON.

Gap Conductance

The initial stored energy and subsequent temperature distribution in the fuel rod during the transient depends upon the fuel-to-clad gap width and the corresponding gap conductance. The codes should have a dynamic gap conductance model, which takes into account the deformation of the fuel and clad due to thermal expansion and also due to the difference between gap pressure and fluid pressure outside the clad. Among the current advanced codes, TRAC-PF1/MOD1, COBRA/TRAC and RELAP5/MOD2 account for some or all of these effects.

Peaking Factor

Axial and radial peaking factors are very important as they describe the power density distribution in the core and also define the hot spots, where the peak clad temperature is likely to occur. These peaking factors depend upon the time of life of the core and should be established by some auxiliary calculation with codes like NODE-P2, PDQ, SIMULATE, etc.

Decay Heat

Decay heat from fission products influences the peak clad temperature for both PWR and BWR large break LOCA. The recommendation for decay heat is to use the ANS Standard 5.1 of 1979, along with the actual fuel history which will provide the correct steady state distribution of fission products. The advanced codes such as TRAC-PF1/MOD1, TRAC-BD1/MOD1 and RELAP5/MOD2 have an option to compute the decay heat from neutronic calculations.

Metal-Water Reaction

The metal-water reaction becomes significant when the clad temperature exceeds 1600°F and it generates more energy in addition to the decay power. The recommended model for this reaction is Cathcart-Pawel (12). This model is more realistic than the Baker-Just equation used in the current ECCS evaluation methodology. It should be noted that the PCT sensitivity to the model for metal-water reaction increases at higher temperatures and it has been shown recently (13) that replacement of the Baker-Just model with the Cathcart-Pawel model (while keeping everything else fixed in a licensing calculation), results in a drop in PCT of 63°F. This reduction will be less at lower temperatures. There is another model available (14) which is based on internal heating data and should be considered.

Flow Regime Maps

In two-phase flow modeling, some idea about the distribution of each phase and the shape of the interface are needed to compute the interfacial heat, mass and momentum transfers. Some of this information is obtained from flow regime maps. The type of regimes and their boundaries are functions of the geometry and orientation of channel, void fraction, and either the mass flux or superficial phasic velocities. Among various flow regime descriptions available in the literature, the description used in the RELAP5/MOD2 code is the simplest; yet it accounts for channel orientation, viz, horizontal or vertical, pre-CHF or post-CHF fluid conditions, and well mixed region such as pump discharge.However, there is still a question about the applicability of these flow regimes in the heated rod bundle region. COBRA/TRAC, on the other hand, uses special flow regime maps for the heated bundle region which can be used as a guide for other codes. At lower mass fluxes, stratified flow may occur in the horizontal pipes of a PWR system and is modeled by the Taitel and Dukler (15) correlation in TRAC-PF1/MOD1 and RELAP5/MOD2 codes. There is no need to include such models in the BWR LOCA codes.

Liquid Entrainment

In some situations such as high vapor velocity or high void fraction, the flow regime changes from bubbly or bubbly-slug to annular-mist flow. Modeling of this regime requires estimating the distribution of liquid in the film and droplet forms. The recommendation here is to use the modified Ishii entrainment model as suggested by Popov, et al (16). The COBRA/TRAC code has very detailed entrainment and de-entrainment models, and thus merits consideration in the new realistic ECCS methodology.

Interfacial Momentum Transfer

TRAC and RELAP5 codes have detailed flow regime dependent models for the interfacial momentum transfer, and may be considered as state-of-the-art at this time. However, the TRAC-PF1/MOD1 code lacks the virtual mass effect except in the choked flow model. For annular flow, the recommendation is to use the Bharathan-Wallis correlation (17) for the counter-current flows and the Wallis correlation (18) for co-current flows. This is important as it controls the refill and reflood rates during a LBLOCA. Another approach (19) in which the vapor drift velocity can be calculated as a continuous function of void fraction and superficial liquid velocity is also available. The interfacial shear required in the two-fluid formulation can then be obtained from this drift velocity.

Interfacial Mass and Energy Transfer

The peak clad temperature depends upon the heat transfer coefficients which are functions of the vapor fraction, vapor and liquid temperatures, flow rates, and the heat transfer regime. All these parameters depend upon the interfacial heat, mass and momentum transfer. The effect of noncondensables is also important as it tends to impede the condensation rate. All advanced codes, namely TRAC, COBRA/TRAC and RELAP5/MOD2, have detailed models for vapor generation which attempt to account for flow regimes and interfacial area and shapes. A thorough review of these models is needed. A limited assessment of the TRAC-BD2 code (20) shows room for significant improvement in this area. In a recent survey (21) of existing correlations and data in direct contact condensation, it was concluded that there is a lack of data in the area of high pressure and also at high liquid subcooling.

Critical Flow

There are two approaches to the computation of critical flow: (a) using rigorous formulation such as the method of characteristics, and (b) using critical flow correlations such as those developed by Moody (22) or Henry-Fauske (23). The first approach is very time consuming from the computational viewpoint, and most of the codes are, therefore, using the second approach. The critical flow model also depends upon the location and size of the break and the fluid conditions upstream of the break. For subcooled liquid, it is recommended to apply Jones' correlation (24) of flashing delay for computing the choke plane pressure and the Bernoulli equation up to choke plane for calculating the flow rate.The use of the homogeneous equilibrium (HEM) model is recommended for two-phase conditions.

Flooding or CCFL

This phenomenon controls the amount of liquid which can flow down for a specified vapor upflow rate and it is very sensitive to interfacial momentum transfer. Therefore, either interfacial momentum transfer for each geometry, where flooding is expected, has to be accurately modeled, or empirical flooding curves for each geometry have to be used. Flooding curves or correlations due to Wallis or Kutateladze are available for that purpose. Only TRAC-BD1/MOD1 has incorporated the flooding or CCFL correlations for the upper core support plate and the side entry orifices (SEOs). Use of flooding correlations can be avoided only if accurate models for interfacial shear and entrainment can be developed.

Level Tracking

In BWRs, it is important to calculate the location of downcomer water level since it controls the initiation of the safety injection system. A level tracking model is, therefore, essential for the BWR LOCA codes. TRAC-BD1/MOD1 has a model for this purpose; however, it has not yet been extensively assessed.

Parallel Channel Effects

It has been observed in multi-channel test facilities such as SSTF that different core channels behave differently during the refill/reflood phase of a BWR LOCA. The flow configuration in these channels varies from co-current or counter-current two-phase flow to single phase flow. These variations in channel flows are caused by the multi-dimensional fluid conditions in the upper plenum due to the sparger location and the variation in the power levels in these channels. A code with a one-dimensional model for the VESSEL component cannot model these effects. However, this impact of the phenomenon on PCT has not yet been established. TRAC-BD1/MOD1 has a capability to model the vessel with either one- or three-dimensional formulations, and can be used to assess the sensitivity of important LOCA parameters to these effects.

Wall Friction

There are many correlations available in the literature for calculating the momentum transfer between the wall and a two-phase mixture. However, based on our experience, the HTFS correlation (25) is the most accurate. Columbia University (26) has also developed a correlation, which has a wide data base covering possible BWR and PWR conditions. These correlations were, however, developed for round tubes while the flow regimes and pressure drops in the rod bundle will be different. In the absence of any correlation based on rod bundle data, the HTFS and Columbia University correlations with appropriate hydraulic diameter are recommended.

Additional Friction Losses

Besides the friction loss at the wetted surfaces, there are additional losses at area changes, bends and grids, etc. These losses should be provided for as additional loss coefficients in a code. Some codes such as TRAC-PF1/MOD1 and COBRA/TRAC have built-in area change models along with the option for user supplied additional friction. However, TRAC-BD1/MOD1 requires user supplied additional losses at all locations.

Wall Heat Transfer

Single-Phase Liquid

The Rohsenow-Choi correlation (27) is recommended for Re $\leq$ 2000 (laminar flow).

The Dittus-Boelter correlation with correction for rod bundle due to Weisman (28) is recommended for Re > 2000 (turbulent flow).

Nucleate Boiling

The Chen correlation (29) is recommended up to the CHF or DNB (departure from nucleate boiling) point. An alternative could be that used in the EPRI void model (30).

Post-CHF Heat Transfer Package

There are two best-estimate packages, one by Groeneveld and Rousseau (31) and the other by Hsu and Young (32). Both packages consist of a CHF correlation, a correlation for minimum stable film boiling temperature, and the heat transfer correlations for transition boiling, film boiling and superheated steam. Because of space limitation, a discussion of these packages is not possible here, but it can be found in (11).

In addition to these packages, there are other critical heat flux models recommended by various organizations, e.g., University of Waterloo (33), Westinghouse (34), Columbia University (35), and Argonne National Laboratory (36).

Columbia University (35) reviewed a wider range of data and developed their own CHF correlation (EPRI-1) which can also be used as an option in a LOCA code. Leung (36), on the other hand, suggested the modified Zuber correlation for the low mass flux (< 100 kg/m^{2}s), and CISE (37) and Biasi correlations (38) for high mass flux (> 200 kg/m^{2}s). These correlations have been used in the COBRA/TRAC code, and are available as options in the TRAC-BD1/MOD1 code.

Pump Model

The pump performance under two phase conditions expected during LOCA, affects the core inlet flow and PCT. In the case of a small break LOCA, the vapor concentration is smaller and the pump influence will be even more significant. Currently all the advanced codes have the same pump model based on Semiscale two-phase homologous curves and degradation function. These models have not yet been verified with different scale pumps. There are other models available through EPRI programs (39 -41) which should be considered.

CONSIDERATIONS FOR SMALL BREAK LOCA

The previous two sections have emphasized the phenomena and best-estimate models for large break LOCA in both PWRs and BWRs. However, an ECCS must also meet the challenge of a small break LOCA. In this section, we will discuss the differences (if any) between the large and small break LOCA (SBLOCA) in both PWRs and BWRs, and examine if any special models are required to predict the reactor behavior during a SBLOCA.

SBLOCA in a PWR

The transient scenario during a small break LOCA in a PWR is quite different from that during a large break LOCA as described in the section on "Large Break LOCA in PWRs". First, the mass discharge rate during a SBLOCA is, of course, much lower than that during a LBLOCA. This results in a slower depressurization of the primary system and a longer, less violent transient. However, the break flow rate model is just as important in SBLOCA as it is in LBLOCA. Moreover, the break flow model should take into account the location and orientation of the break, i.e., at the bottom, top or side of a pipe [42].

Second, the scenario is significantly affected by the assumption regarding the reactor coolant pump (RCP) operation. If the pumps are assumed to be running, the pump performance for two-phase operation becomes very important. On the other hand, if the pumps are tripped, either manually or because of loss of A.C. power, the primary system settles in a natural circulation mode resulting in phase separation in the pressurizer, reactor vessel and reactor pipings. Thus, accurate models are needed to predict stratification in horizontal pipes and two-phase mixture levels in the pressurizer, upper head, upper plenum and core. This is an additional requirement if a large break LOCA code has to be modified for application to SBLOCA.

Third, the steam generator heat transfer is more important during a SBLOCA than in a LBLOCA since the energy loss through the break is significantly less in the former case. Thus, the primary heat loss through the steam generator becomes a significant part in the overall balance between the energy input and output of the primary system. On the other hand, the ECC bypass stage of the LBLOCA does not occur during a PWR SBLOCA. This is a considerable modeling relief for a SBLOCA code. However, the wall heat transfer package as described in the "Wall Heat Transfer" section should hold good for both large and small break LOCAs.

In summary, a code developed for best-estimate analysis of LBLOCA in a PWR can also be used for a SBLOCA analysis if the following modeling additions are made:

1. Expand the break flow model to include the effect of break location and orientation.
2. Stratified flow model in horizontal hot and cold legs.
3. Phase separation or two-phase mixture level calculation in the vertical components, i.e., pressurizer, core, etc.
4. Single and two-phase natural circulation including reflux mode in the hot leg.
5. Accurate two-phase pump model.
6. Accurate heat transfer models in steam generator. This includes condensation in the primary side and two-phase mixture level in the secondary side.

Alternatively, a specialized fast running code for SBLOCA can also be developed.

SBLOCA in a BWR

Unlike in PWRs, there is no significant difference between the transient scenarios during a large and a small break LOCA in a BWR. This is because of the actuation of the automatic depressurization system (ADS) during a SBLOCA. This system essentially transforms a SBLOCA into a LBLOCA. Thus, a best-estimate code suitable for analyzing a BWR LBLOCA should also be adequate for analysis of a BWR SBLOCA.

RECOMMENDATIONS FOR REALISTIC ECCS EVALUATION

The parameters which characterize a large or small break LOCA in an LWR are the fuel cladding temperature peaks and their timings in the blowdown and reflood phases, and the timing of the quench of hot spots. Prediction of these parameters is affected by many thermal-hydraulic and fuel behavior models as discussed. For a realistic ECCS evaluation methodology, these models must be physically reasonable and have a broad data base. One approach of developing a new ECCS evaluation procedure would be to examine the present best-estimate LOCA codes, select the best available code, and improve the code by incorporating more accurate and physically realistic models.

There are three NRC-sponsored codes, namely, TRAC-PF1/MOD1, RELAP5/MOD2 and COBRA/TRAC that should be considered for the PWR LOCA analysis. based on the overall technical capabilities and user convenience, we recommend that the latest version of TRAC-PWR, i.e., TRAC-PF1/MOD1, and RELAP5/MOD2 be considered for the final selection as the base code for developing the new ECCS evaluation methodology for PWRs.

TRAC-PF1/MOD1 is an improved version of the large break LOCA code, TRAC-PD2, and is applicable to both large and small break LOCA analyses for PWRs. The VESSEL module of TRAC-PF1/MOD1 has the option of using either the three-dimensional or the one-dimensional thermal-hydraulic formulation for modeling the reactor vessel. Both formulations are based on a two-fluid model. The loop components, i.e., the components outside the reactor vessel, also employ the one-dimensional two-fluid model. For the one-dimensional formulation, a two-step numerics has been used to reduce the computer running time. Finally, the previous versions of TRAC-PF1/MOD1, i.e., TRAC-PF1 and TRAC-PD2, have been assessed at several organizations including Brookhaven National Laboratory (BNL) with test data pertinent to both large and small break LOCAs in PWRs, and reasonable agreement between the code predictions and the test data has been found in most cases.

RELAP5/MOD2 on the other hand, is more flexible and user oriented code than TRAC. It also uses a one-dimensional two-fluid formulation. The constitutive packages are quite comparable in TRAC-PF1/MOD1 and RELAP5/MOD2. However, RELAP5/MOD2 lacks a multi-dimensional capability for modeling the vessel and has a very limited history of assessment. Therefore, if multi-dimensional effects can be accounted for by correlations with a wide data base, and if the code is extensively assessed, then RELAP5/MOD2 will be a serious contender for realistic LOCA analysis. It has an added advantage that it could be used for both PWR and BWR applications.

For BWR LOCA analysis, the selection should be made between the TRAC-BD1/MOD1 and RELAP5/MOD2 codes. TRAC-BD1 is essentially the BWR version of TRAC-PF1. Thus, the thermal-hydraulics in the reactor vessel is calculated using a three-dimensional, two-fluid model. As discussed in "Large Break LOCA in BWRs", multi-dimensional phenomena, particularly in the upper plenum, play a major role during the BWR LOCA. Besides, the RELAP5 development efforts during the past few years had concentrated mostly on PWRs. Therefore, based on overall technical merits, we recommend that TRAC-BD1/MOD1 be selected as the base code for developing the new ECCS evaluation methodology for BWRs. However, if RELAP5/MOD2 or a similar code is chosen for PWR LOCA analysis, it might be advantageous to modify or improve the same code for the BWR LOCA analysis as well.

Table 1 shows a comparison among the current ECCS, TRAC-PF1/MOD1, TRAC-BD1/MOD1, and the BNL-suggested models for LOCA analysis based on the current literature survey. The BNL-suggested models may be incorporated either in the TRAC-PF1/MOD1, TRAC-BD1/MOD1 or RELAP5/MOD2 code and the results should be compared with a broad range of experimental data pertinent to the LOCA. It should be kept in mind that these models are usually based on separate-effects tests, which may be limited by scale and fluid conditions such as pressure, temperature, etc., and should be assessed with integral and larger scale tests. The effort should, of course, be concentrated on the phenomena which have maximum impact on the peak clad temperature (PCT). The relative importance of these models,on the basis of their effect on PCT in the blowdown phase of a PWR LBLOCA, were established in a previous study (6) using RELAP4/MOD6. The core power, gap conductance, peaking factor, etc. were in the top seven positions, while surprisingly, critical or break flow models were ranked tenth. Similar conclusions were also reached in the recent BNL study using the TRAC-PD2/MOD1 code (5).

Besides the thermal-hydraulic models, there are also uncertainties regarding the initial conditions such as core power, peaking factor and fuel and clad properties for both PWR and BWR LOCA analyses. It is recommended that realistic values be used for these parameters. Some of these parameters do depend upon the time of life of fuel in the fuel cycle.

Other conservatisms are in the transient scenario specified in the current ECCS evaluation method. These are: single failure in ECC system which may result in loss of one of the two trains for ECC, and reactor coolant pump trip. For PWR LBLOCA, these scenarios shift the PCT from the blowdown to the reflood phase, and now the important parameters are the decay power, flooding in downcomer, condensation, metal-water reaction, etc.
Finally, the current ECCS methodology does not specify any constraints for vapor generation or condensation model. However, nonequilibrium phase change is an accepted important phenomenon during the transient, and it will affect the break flow rate, the heat transfer regimes in the core, ECC bypass in a PWR downcomer, BWR upper plenum mixing, etc. The vapor generation models in the TRAC-PF1/MOD1, TRAC-BD1/MOD1 and RELAP5/MOD2 codes are acceptable in most cases, but direct contact condensation models are still not established with the same degree of confidence. Moreover, there is still a need for ranking the various phenomena and models which are relevant during various stages of LOCA when additional failures are postulated.

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TABLE 1 - COMPARISON AMONG THE CURRENT ECCS EVALUATION MODELS, TRAC-PF1/MOD1 MODELS, TRAC-BD1/MOD1 MODELS, AND BNL SUGGESTIONS FOR DEVELOPING A NEW REALISTIC ECCS METHODOLOGY FOR PWRS.

MODEL NO.	CURRENT ECCS EVALUATION MODELS	TRAC-PF1/MOD1 MODELS	TRAC-BD1/MOD1 MODELS	BNL-SUGGESTED MODELS
A. Source of Heat During LOCA				
A4	Fission product decay heat will be 1.2 times 1971 ANS	Code allows decay power as input.	Code allows decay power as input.	Decay power based on ANSI-ANS/1979 Standard and with actual operating history
A5	Metal-water reaction. Baker-Just (64). Reaction not limited by steam and also applicable to inner side of clad.	Cathcart (43)	Cathcart (43)	Cathcart-Pawel (13) or Ocken (15).
B. Swelling and Rupture of the Cladding				
	Amount of fuel swelling due to increase in gap pressure.	The code accounts for clad deformation due to temperature but not due to gap pressure.	The code does not account for clad deformation.	Code should account for fuel swell due to gap pressure and clad temperature similar to FRAP, GAPCON or COBRA/TRAC.
C. Blowdown Phenomena				
C1b	Break flow rate based on Moody model; C_D=0.6 - 1.0	Model close to HEM for two phase flow and for subcooled liquid, the choked velocity is obtained from the Bernoulli equation with choked plane pressure from Jones (24) with HEM critical velocity as lower limit.	Model based on RELAP5 for two phase conditions and modified Burnell (44) for subcooled choking.	Nonequilibrium model for subcooled blowdonw,HEM for saturated blowdown. (See Section 3.10)

MODEL NO.	CURRENT ECCS EVALUATION MODELS	TRAC-PF1/MOD1 MODELS	TRAC-BD1/MOD1 MODELS	BNL-SUGGESTED MODELS
C1c	Subtract the total amount of liquid injected during the bypass period from the liquid inventory of the vessel.	The code is based on phenomenological model of interfacial mass and momentum transfer. Entrainment model is based on Kataoka and Ishii correlation (45) interfacial shear in annular flow - Wallis correlation (8).		Code should properly calculate ECC bypass during blowdown phase. The models can be developed on the basis of Creare, BCL, tests for ECC bypass. In the interim, either the interfacial shear should be increased or empirical flooding curves be imposed on the downcomer flow as done in TRAC-BD1/MOD1 at core support plate and side entry orifice.
C2	Frictional pressure drop will be based on experimentally verified two-phase multiplier.	Homogeneous and annular flow multipliers. Each of them have different single phase correlation.	Hancox and Nicoll (46) two phase multiplier with single phase friction factor from Pfann (47) correlation.	HTFS (25) or Baroczy or Columbia University (26) correlation.
C4	Critical heat flux models. Ones based on steady state data are permitted for LOCA transients.	Biasi correlation (28).	Biasi critical quality or GE/CISE critical quality correlation for G > 200 kg/m^{2}s and modified Zuber for low flows.	The following correlations are suggested (1) Bowring WSC-2 (48) (2) Columbia EPRI-1 (35) (3) Groeneveld (49) (4) a For PWR, Biasi (38) high flow. Modified Zuber (50) low flows (PWR) b For BWR, CISE-GE (51) high flows, Modified Zuber (50) low flows.
C4b	Critical heat flux models (11) from steady state data permitted but not restricted to: W3, B&W-2, Hench-Levy, Macbeth, Barnett, Hughes.	Same as C4	Same as C4	Same as C4
C4d	Critical heat flux models (11) from transient data are acceptable but not restricted to Slifer-Hench.	Same as C4	Same as C4	Same as C4
C4e	No rewetting or return to nucleate boiling permitted in blowdown phase.	Code will compute heat transfer rate as permitted by fluid and rod surface	Code will compute heat transfer rate as permitted by fluid and rod condi-	Rewet or return to nucleate boiling should depend upon fluid and clad conditions,
C5	Post-CHF heat transfer. Any model which is supported with data.			
C5b	Some of the correlations which are acceptable are: Groeneveld correlation for film boiling away from low pressure singularity. Dougall-Rohsenow film boiling correlction.	Liquid HTC based on Bromley (52), radiation and Forslund-Rohsenow for dispersed droplets. Vapor HTC in film boiling.	Bromley (52) for liquid phase and maximum of Dougall-Rohsenow (54), laminar flow and natural circulation correlation (53).	Chen (55) for vapor phase. Tien's model for radiation heat transfer to droplets. Forslund-Rohsenow for contact droplet heat transfer. Or, Berenson (56) with modification for subcooled liquid.
	Westinghouse correlation for transition boiling (11), McDonough, Milich and King correlation for transition boiling.	Transition boiling, total heat transfer is obtained from quadratic interpolation between CHF and correlation at minimum stable film boiling point. The vapor heat transfer is the same as in C5b and liquid difference of mixture or total and vapor phase.	Vapor HTS as max of Dougall-Rohsenow (54) and natural convection. Liquid HTC as combination of Bromley and quadratic interpolation between CHF and T_{min} points.	Tong & Young (57) for liquid phase and Dougall Rohsenow (54) for vapor or vapor droplet with thermal nonequilibrium model. For low flow (mass flux <680 kg/ m^{2}s) Hsu correlation (32) may be used. Or, linear interpolation between CHF and MSFB on log-log graph.

MODEL NO.	CURRENT ECCS EVALUATION MODELS	TRAC-PF1/MOD1 MODELS	TRAC-BD1/MOD1 MODELS	BNL-SUGGESTED MODELS
C5b	The transition boiling correlation shall not be used during blowdown if the temperature difference between the clad and the fluid exceeds 300°F.	Transition boiling regime is assumed whenever α<0.96 and T_{clad} <T_{MSFB}. The minimum stable film boiling temperature is obtained from homogeneous nucleation correlation by Henry (58).	Transition boiling regime will be used if the fluid and surface conditions meet the criterion T_{clad}<T_{min}.	T_{MSFB} can be obtained from homogeneous nucleation correlation by Henry(58). Other models such as (56) should also be considered.
	No return to transition boiling during blowdown, even if temperature difference between clad and fluid is less than 300F.	Transition boiling is allowed whenever fluid and clad conditions permit.	Transition boiling is allowed whenever fluid and clad conditions permit.	Transition boiling is allowed whenever fluid and clad conditions permit. Model should use realistic homologous curves and degradation function obtained for actual pump or properly scaled pump.
C6	Pump modeling based on realistic data. Furthermore, when saturation conditions are detected at the BWR pump suction, the pump heat is assumed to be proportional to inlet quality, going to zero for 1% quality, so long as core flow stops before pump suction quality reaches 1%.	Code has option of specifying single and two-phase homologous curves along with degradation function through the input or using built-in Semiscale curves.	Code has option of specifying single and two-phase homologous curves along with degradation function through the input or using built-in Semiscale curves.	
D. Post-Blowdown Phenomena				
D4	No steam flow in unbroken loops during refill and reflood phase, when accumulator flow in cold leg.	Computed on the basis of phenomenological models.		Computed on the basis of realistic models.
D5	Refill and reflood heat transfer for PWR. For reflood rate of 1 inch/sec or more, the heat transfer rate is based on applicable experimental data including FLECHT data.	Code does not have any special correlations and the calculation is based on mechanistic model, accounting for finer nodalizations around the quench front and axial conduction in the rods.		No distinction needed for 1 inch/sec of reflood rate. A mechanistic model along with axial conduction would be sufficient.
D6	Convective heat transfer coefficient for BWR rods under spray cooling will be based on appropriate experimental data.		This code has multi-dimensional/multi-channel models for the core. The heat transfer coefficients at various locations at the rod surface and channel box surface will be determined on the basis of fluid conditions and surface temperature.	Correlations based on TLTA or ones in TRAC-BD1/MOD1 can be used after sufficient verification.
D6a	Convective heat transfer coefficient of zero be applied between the time period of lower plenum flashing and core spray reaching rated flow.			
D6b	H.T. coefficients during period between core spray and reflooding will be 3.0, 3.5, 1.5 and 1.5 Btu/hr ft^{2}F for rods in outer corner, outer row, next to outer row and interior.			
D6c	After reflooding fluid reaches the required level, a H.T. coefficient of 25 Btu/hr ft^{2}F will be applied to all rods.			

MODEL NO.	CURRENT ECCS EVALUATION MODELS	TRAC-PF1/MOD1 MODELS	TRAC-BD1/MOD1 MODELS	BNL-SUGGESTED MODELS
D7	Requirements for channel box are similar to rods as described in D6.		Same as for D6	Correlations based on TLTA or ones in TRAC-BD1/MOD1 can be used if verified.
D7a	Same as D6a			
D7b	H.T. coefficient of 5 Btu/hr ft^2F instead of given in D6b.			
D7c	Wetting of the channel box shall be assumed to occur 60 seconds after the time determined using the correlation based on Yamanouchi analysis.		Wetting of channel box is predicted on phenomenological basis.	Wetting of channel box should be based on phenomenological basis.

Note: Several correlations have been suggested for the same phenomena in Sections C4 and C5. Sensitivity studies and code assessment are needed to select the best model.