RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**415-8503**
SRP Section:	**15.06.05 – Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary**
Application Section:	**15.6.5**
Date of RAI Issue:	**02/22/2016**

Question No. 15.06.05-15

Please explain how the two-phase water level in the core was determined. The applicant needs to provide a clear definition of the two-phase level, and elaborate on any criteria or thresholds involved. Document the assumptions involved in calculating the two-phase water level in the core, and justify combining CEFLASH-4AS and COMPERC-II reflood two-phase levels as a conservative approach. Provide a plot of CEFLASH-4AS and COMPERC-II two-phase levels for a cold leg break which requires an extended period of interpolation between CEFLASH-4AS and COMPERC-II. Identify the time period when interpolation between the CEFLASH-4AS and COMPERC-II 2-phase levels is occurring, and explain what are the two endpoints of the interpolation that are being used to get the final two-phase level

Response

Definition of Two-phase level

In CENPD-137(page 30~42) the two-phase mixture level is explained in detail. This response is written to summarize the explanation provided in CENPD-137. The total node mixture level is determined using the phase separation model (see Figure 15.06.05-15-1) and the bubble rise model as described below. The bubble rise model in CEFLASH-4AS has been modified to more realistically describe the effects of non-uniform bubble distribution within a node. CEFLASH-4AS computes the dynamic bubble mass from the following expression:

$$\frac{\Delta M_{GB}}{\Delta t} = \frac{\Delta M_G}{\Delta t} - \sum_{ij}^{paths} C'_{ij}\, W_{ij}\, X_{ij} - \frac{dM_{GB}}{dt}\Big|_{release} \qquad (1.1.1-1)$$

where the disengagement rate is given by

$$\left.\frac{dM_{GB}}{dt}\right|_{release} = V_B\, \alpha A \rho_V \qquad (1.1.1-2)$$

$$V_B = \frac{V_D}{1-\alpha} \qquad (1.1.1-3)$$

The equation 1.1.1-3, V_D is a constant input drift velocity. A value of 2 ft/sec is used in the small break LOCA analysis. This value was compared with the Westinghouse experiments (see Figure 15.06.05-15-2). This method yields unrealistically low values for the bubble release rate in nodes wherein there is a high rate of bubble production due to flashing and/or boiloff. A high bubble production rate produces a vertical gradient of the bubble population with the void fraction being largest at the surface of the two-phase fluid. Based on this the surface void fraction should be used to determine the rate of bubble release.

The true release rate in a node of low uniform bubble production can be represented using a drift velocity twice that of the true drift velocity if the bubble release is computed using the average nodal void fraction($\bar{\alpha}$). This method is employed in the annulus nodes, where a drift velocity of 4 ft/sec is used in conjunction with the disengagement rate based on $\bar{\alpha}$.

The following simplifying assumptions are made in defining the bubble density in the inner vessel.

1) The flashing steam production rate per unit volume is uniform throughout the two-phase region.
2) There is no bubble production due to boiloff in the upper plenum.
3) A uniform linear heat rate is assumed in the core for the purpose of calculating the boiloff rate only.
4) A sub-cooled region, containing no bubbles, resulting from sub-cooled flow into the lower plenum exists in the inner vessel.
5) A quasi-steady state distribution of bubbles exists.

The bubble population distribution using these assumptions is shown in the Figure 15.06.05-15-1.

Typical calculation equations related to the core level are as follows: The total linear bubble production rate is given by

$$\dot{P}_i = \begin{cases} \frac{\dot{q}_{LP}}{h_{lV}} + \dot{W}_{fs,LP} & \text{lower plenum} \\ \frac{\dot{q}_{core}}{h_{lV}} + \dot{W}_{fs,core} & \text{core} \\ \dot{W}_{fs,up} & \text{upper plenum} \end{cases} \qquad (1.1.1\text{-}14)$$

The void fraction as a function of the axial position is as follows

$$\left.\begin{array}{ll} \alpha(Z) = \dfrac{\dot{P}_i(Z - Z_i) + W_{i-1}}{\dot{P}_i(Z - Z_i) + W_{i-1} + V_D \rho_V A_i} & Z > Z_{SC} \\ \alpha(Z) = 0.0 & Z \le Z_{SC} \end{array}\right\} \qquad (1.1.1\text{-}18)$$

The surface void fraction calculated using Equation 1.1.1-19 represents a steady state value. In order to obtain a surface void fraction for use in Equation 1.1.1-2 to calculate the bubble release rates, the steady state surface void fraction is multiplied by the ratio of the dynamic inner vessel bubble population to the steady state inner vessel bubble population.

$$\alpha_{surf} = \frac{\dot{P}_{ii}(Z_{2\emptyset} - Z_{ii}) + W_{ii-1}}{\dot{P}_{ii}(Z_{2\emptyset} - Z_{ii}) + W_{ii-1} + V_D \rho_V A_{ii}} \qquad (1.1.1 - 19)$$

where the disengagement rate is given by

$$\left.\frac{dM_{GB}}{dt}\right|_{release} = V_B\, \alpha A \rho_V \qquad (1.1.1 - 2)$$

The inner vessel bubble mass is obtained for each region

$$M_{GB,i,ss} = \rho_V A_i \int_{Z_i}^{Z_i'} \alpha(Z) dz \qquad (1.1.1 - 20)$$

The bubble disengagement rate is then computed as:

$$\left.\frac{dM_{GB}}{dt}\right|_{release} = \frac{V_D}{1 - f\alpha_{surf}}\, f\alpha_{surf}\, A_{ii} \rho_V \qquad (1.1.1 - 23)$$

Where $f = M_{GB,DYN}/M_{GB,SS}$ and α_{surf} is determined from equation 1.1.1-19.
The height of the two-phase fluid region must be known. This height is determined by comparison of the total two-phase fluid volume to the geometry of the variable area inner vessel node. The two-phase fluid volume is obtained using the inner vessel liquid mass computed from the conservation of mass, the dynamic inner vessel bubble mass computed from equation 1.1.1-1, and the respective phase specific volumes.
The phase separation model described above allows calculation of the bubble population gradient and hence the enthalpy gradient within the two-phase fluid in the inner vessel. The steady state bubble masses within each region are calculated using equation 1.1.1-20. The dynamic bubble mass is assumed to be distributed in proportion to these steady state concentrations. The bubble concentrations are used to define the enthalpy of two-phase fluid leaving the inner vessel. In the lower plenum and upper plenum the two-phase enthalpy is given as:

$$h_{2\emptyset,i} = \frac{M_{li}h_l + M_{GB,i}h_v}{M_{li} + M_{GB,i}} \quad (1.1.2-25)$$

The two-phase volume is obtained from

$$M_{l_i} = \rho_l\,(V_{2\emptyset,i} - \frac{M_{GB,i}}{\rho_v}) \quad (1.1.2-26)$$

where $V_{2\emptyset,i}$ = two − phase volume in region i.

The total node mixture height is given by

$$ZM_i = \sum_{k=1}^{N} \frac{VM_k}{A_k} \quad (1.1.3-1)$$

where

N = total number of subregions
A_k = cross sectional area of k^{th} subregion
VM_k = two-phase mixture volume in k^{th} subregion
ZM_i = two-phase mixture height of node

Tests conducted in the Westinghouse ECCS Verification Test Facility to measure the height of two-phase fluid as a function of water inventory. Comparison of the test results with predictions obtained using the CEFLASH-4AS inner vessel bubble rise model are shown in Figure 15.06.05-15-2. The good agreement between data sets verifies the applicability of the model.

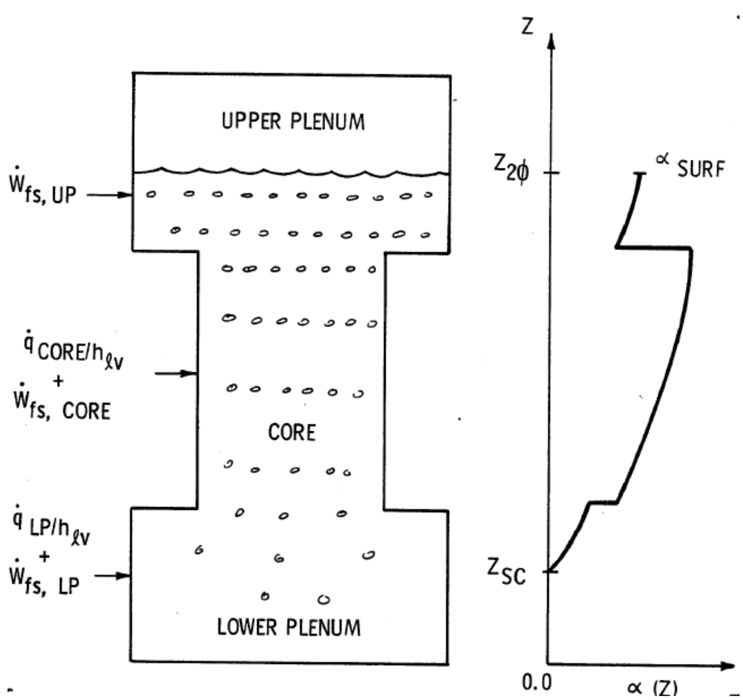

Figure 15.06.05-15-1 CEFLASH-4AS Inner Vessel Phase Separation Model.

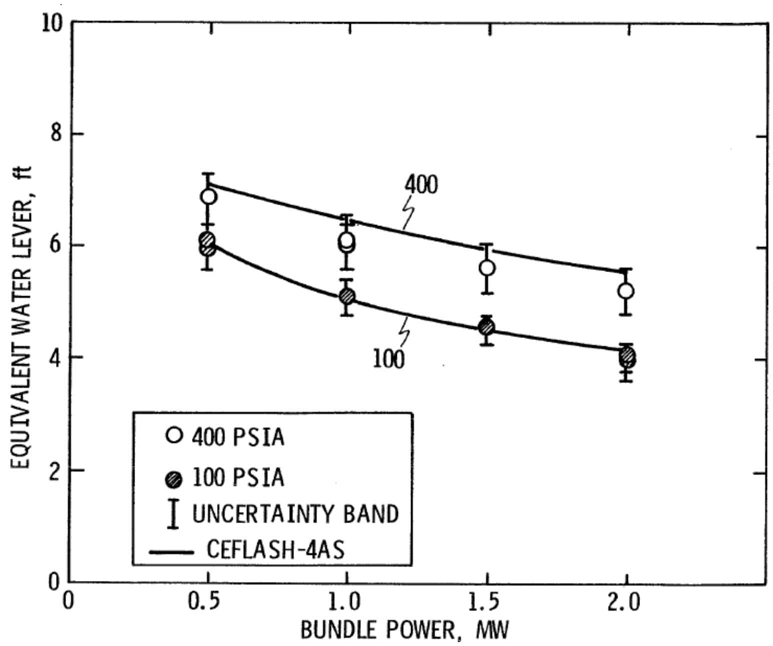

Figure 15.06.05-15-2 Comparison of CEFLASH-4AS bubble rise model with Westinghouse boil-off tests Two-phase level

<u>Plot of CEFLASH-4AS and COMPERC-II two-phase level for cold leg break:</u>
Figure 15.06.05-15-3 shows the CEFLASH-4AS code two-phase core level results for 0.35 ft^2 cold leg break case. Figure 15.06.05-15-4 shows the COMPERC code reslt for 0.35 ft^2 cold leg break. The level above 12.5 ft, which is the same as the top height of the core is omitted from the COMPERC level input.

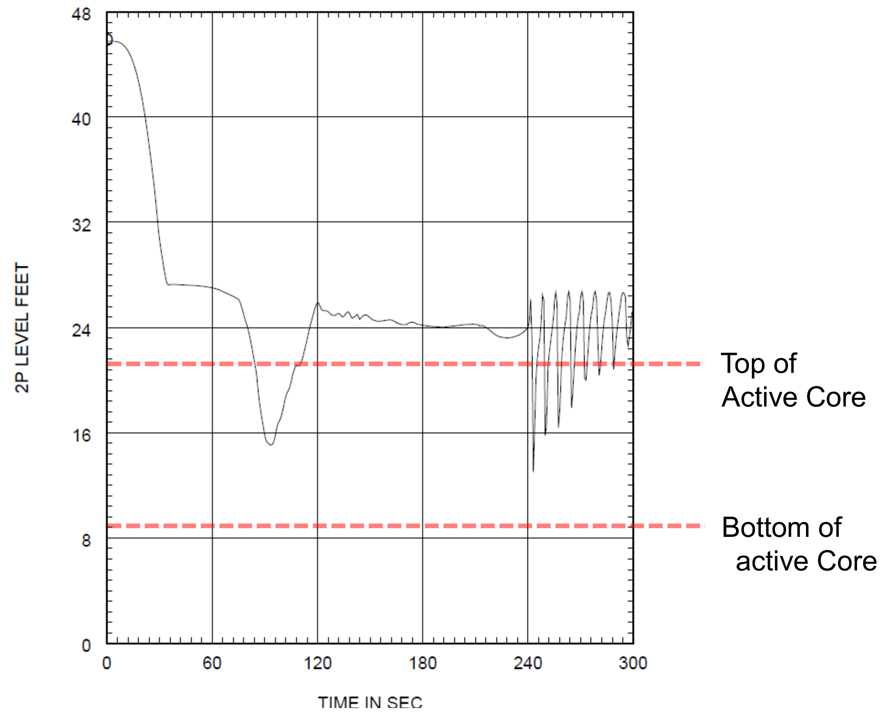

Figure 15.06.05-15-3 CEFLASH-4AS core two-phase level for cold leg break size 0.35 ft^2

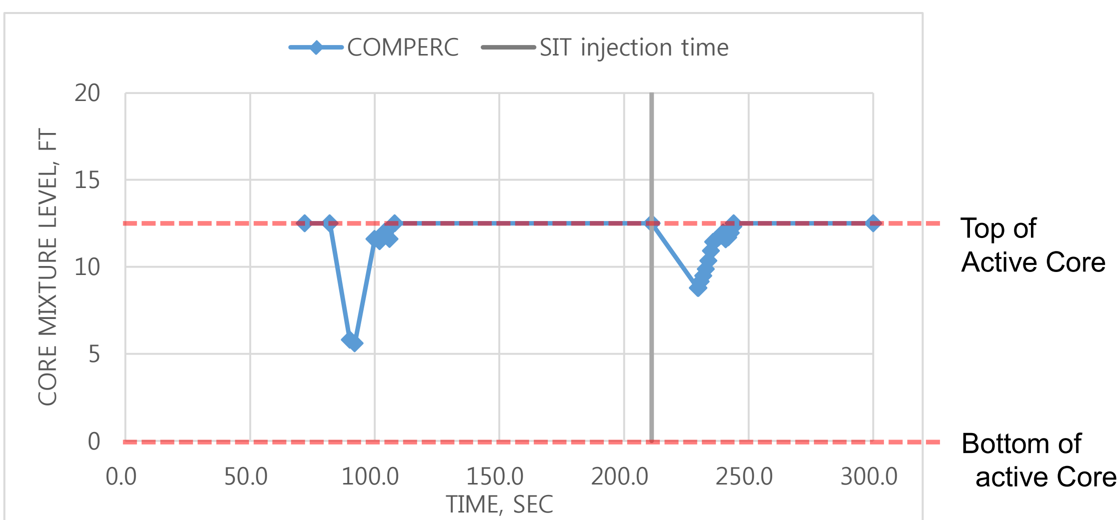

Figure 15.06.05-15-4 COMPERC core two-phase level for cold leg break size 0.35 ft^2

Final level:

The final level is a combination of the two-phase level from the CEFLASH-4AS code and the collapsed level after SIT injection is calculated from COMPERC-II input. The CE SBLOCA methodology has adopted a method to predict the PCT conservatively. That is by combining the CEFLASH-4AS and COMPERC-II reflood two-phase levels. When the safety injection tank flow is taken into account, the COMPERC-II code calculates the two-phase levels instead of CEFLASH-4AS, which can be seen in Figure 15.06.05-15-5. The COMPERC-II code calculates the collapsed two-phase level and the results are transferred to the PARCH-II code. For this reason, the most conservative PCT occurs mainly for the large break size when SIT flow is injected. The combined two-phase level is a conservative approach because the level is shrunk forced.

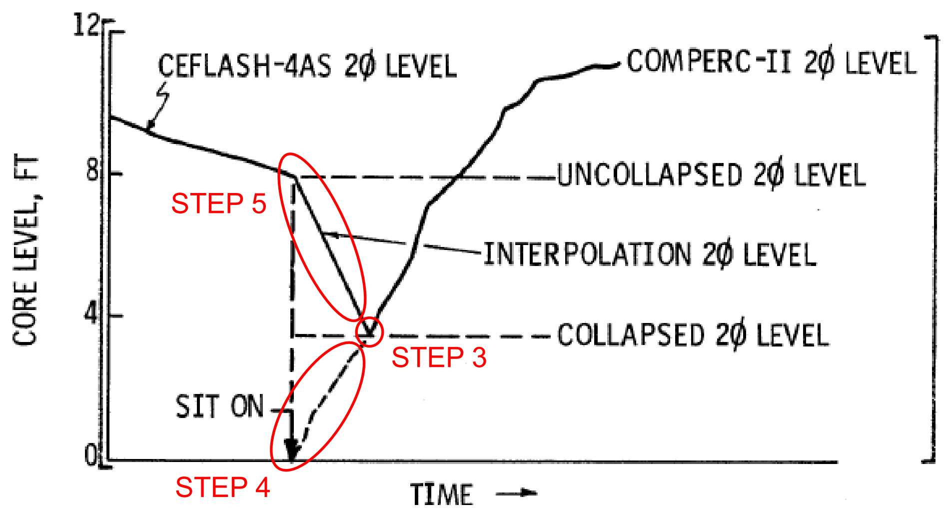

Figure 15.06.05-15-5. Combining CEFLASH-4AS and COMPERC-II reflood two-phase levels

Interpolation time:

The interpolation time begins at SIT injection. This time is calculated from the CEFLASH-4AS code. After this time, the two-phase level is obtained from the manner described below. The final two-phase level is a combination of the CEFLASH-4AS two-phase level, the interpolated level, and the COMPERC level. The method of how to interpolate CEFLASH-4AS level and COMPERC code level is described in below steps. The interpolated time is also described in the below steps.

STEP 1: Obtain CEFLASH-4AS two-phase level.

STEP 2: Remove level data above 12.5 ft. PARCH-II code does not use level data above active core region.

STEP 3: Calculate the collapsed level. The collapsed level is calculated as follows : Collapsed level = excess volume/ (core + downcomer cross section area + bypass area).

STEP 4: Refill SIT water from bottom of core level until the SIT level equal to the collapsed level. The Refill water level by SIT is calculated in the COMPERC code.

STEP 5: Draw a straight line from end of CEFLASH-4AS two-phase level at SIT injection time to the collapsed level.

STEP 6: After STEP 5, COMPERC-II code produces two-phase level

Therefore, the final level is same as the level shown in Figure 15.06.05-15-4.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environment Report.

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Application Section:	**15.6.5**
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Question No. 15.06.05-16

Provide the axial power shapes being used in the SBLOCA runs, both for the average core (CEFLASH-4AS input) and for the hot rod (STRIKIN and PARTCH inputs). Provide the information showing how the hot spot is peaked to an Fq of 2.6. This information is needed to demonstrate to the staff that conservative power shapes have been used in the SBLOCA analysis.

Response

TS

Figure 1. CENPD-254 Power Shape

Figure 1 provides the axial power shapes being used in the SBLOCA runs, both for the average core (CEFLASH-4AS input) and for the hot rod (STRIKIN and PARTCH inputs) and hot spot is peaked to an Fq of 2.625.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

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Question No. 15.06.05-17

CEFLASH-4AS node diagram in Figure 3.1-1 of the SBLOCA Technical Report (APR1400-F-ANR-14001-P) indicates that the safety injection tank (SIT) and safety injection pump (SIP) flows are injected into the downcomer below the node where the cold legs are connected to the downcomer. The TeR description of Figure 3.1-1 also mentions that the emergency core cooling system (ECCS) is modeled by flow paths connected to the lower annulus. However, the direct vessel injection (DVI) nozzles are actually located in the upper annulus of the APR1400 design, 2.1 m above the cold leg nozzles. Therefore, this simulated arrangement may lead to a non-conservative retention of water in the reactor pressure vessel (RPV) by reducing the amount of ECC bypass. The approved CE-ABB SBLOCA methodology specifies that CEFLASH-4AS is to be used until the SITs are activated. Thus, for smaller breaks in the cold leg, there is ample opportunity for the pumped safety injection to be bypassed to the break, an effect which is precluded with the nodalization being employed in the CEFLASH-4AS model. It appears that the applicant's depicted methodology provides no mechanism for the ECCS injection water to be ejected out of the break, and thus, may be non-conservative in this regard. Please demonstrate that placing the DVI nozzles in the lower annulus instead of the upper annulus, where they are physically located, results in a conservative or realistic treatment of ECCS injection in the CEFLASH-4AS simulations for both the DVI line and cold leg breaks. Also describe where the DVI line break and cold leg break nodes are located in the CEFLASH-4AS model.

Response

According to the CENPD-137(page 119), a sensitivity study for the three injection locations(upper annulus, lower annulus, and cold leg) was performed in order to determine the influence of injection location in CEFLASH-4AS on the calculated annulus pressure

transient. The sensitivity study shows that the pressure transient used for reflood calculations is insensitive to the CEFLASH-4AS ECC injection location (See figure 15.06.05-17-1).

When it comes to the CEFLASH-4AS code, if the ECC injection is connected to the upper annulus, a code calculation instability occurs after the ECC is injected into the steam region. The DVI line break node located in the CEFLASH-4AS model is shown Figure 15.06.05-17-2.

During SBLOCA, the effect of ECC bypass is very small because the thermal hydraulic transient is very slow in comparison with LBLOCA. For the case of the DVI line break, the effect of ECC bypass is negligible, because the break height is the same as ECC injection location. For a cold leg break the ECC bypass may have an effect; however the impact is expected to be minimal.

In order to evaluate the effect of ECC bypass, the sensitivity studies were performed for the three cases of HPSI flowrate. As you can see in Figures 15.06.05-17-3 thru 7, even if the HPSI flowrate reduces to 25% considering the effect of ECCS bypass, there was no impact on the analysis result. Therefore, it is determined that the CEFLASH-4AS model has sufficient conservatism for the ECC injection location.

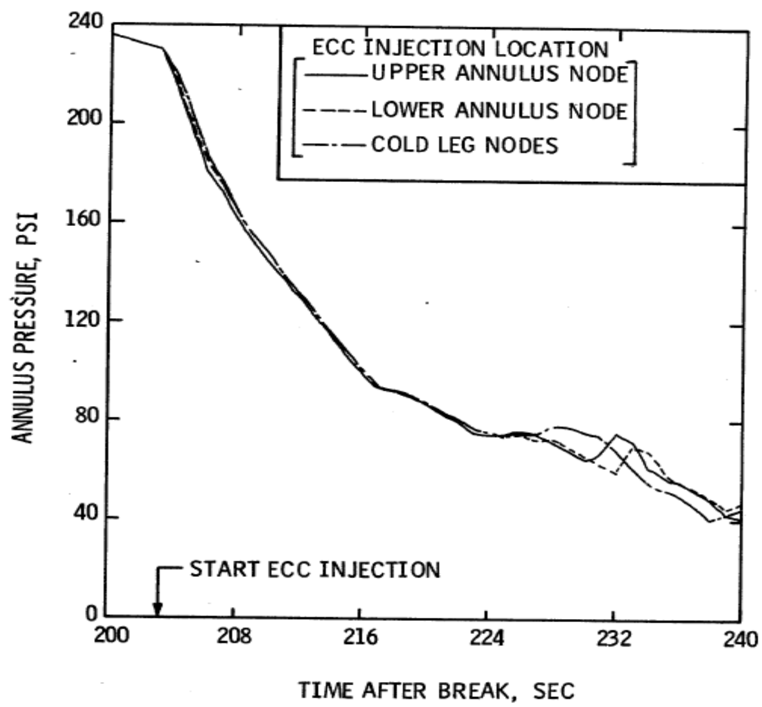

Figure 15.06.05-17-1. Influence of ECC Injection Model on Annulus Pressure Transient

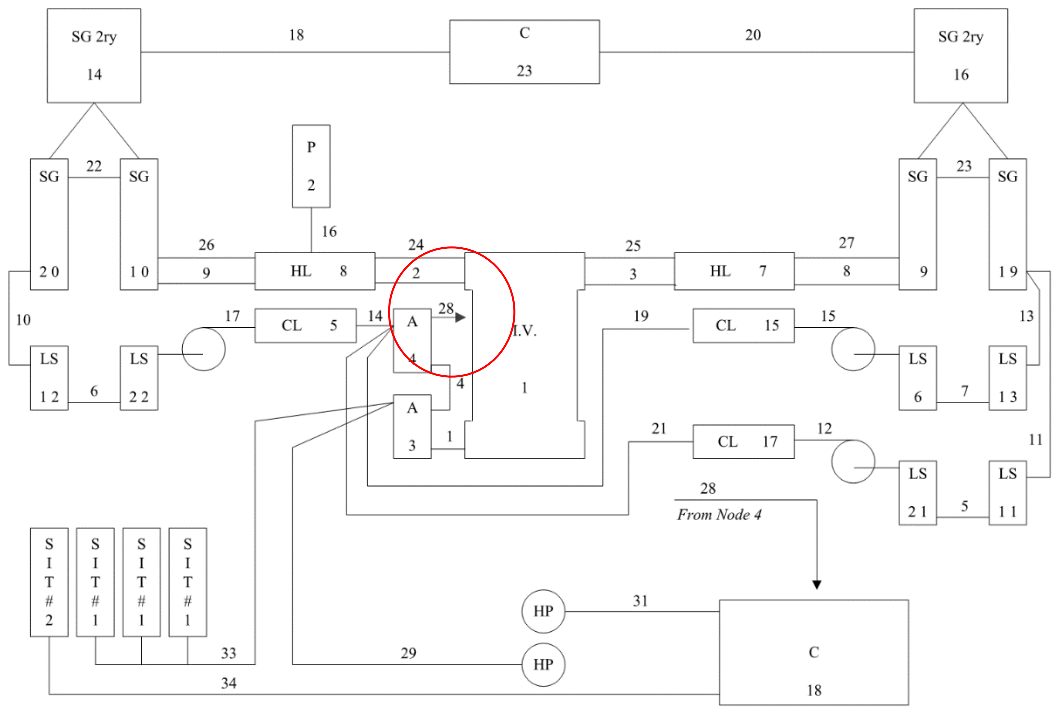

Figure 15.06.05-17-2. Nodalization of DVI line break for CEFLASH-4AS

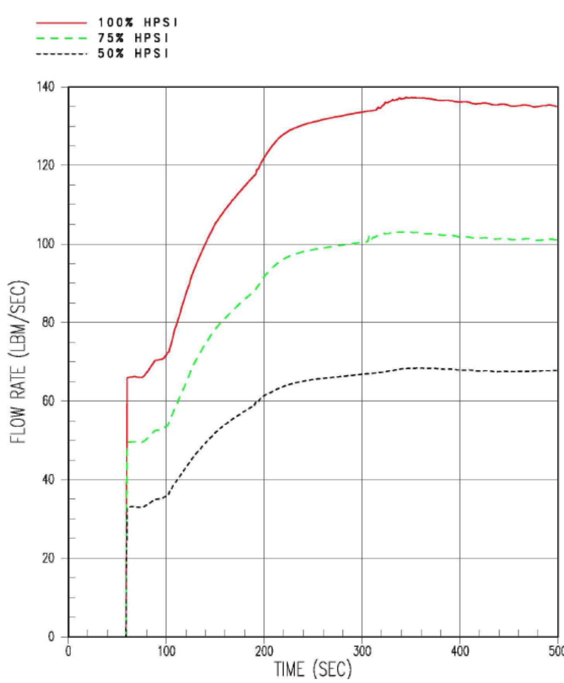

Figure 15.06.05-17-3. HPSI flowrate

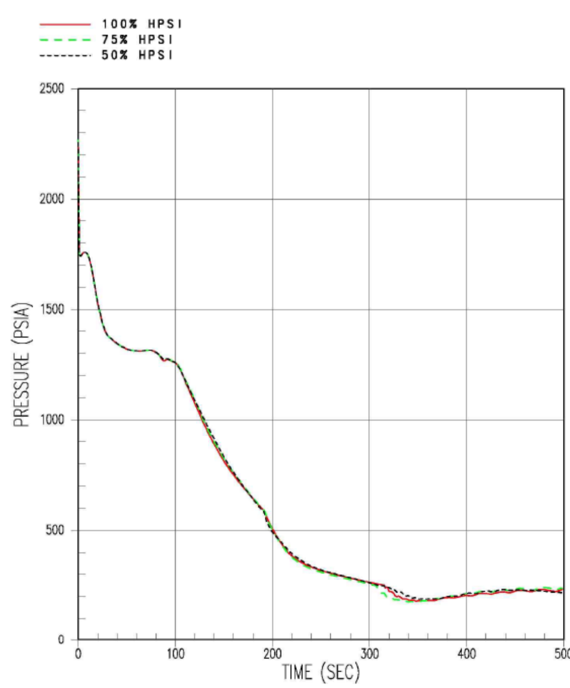

Figure 15.06.05-17-4. Core Pressure

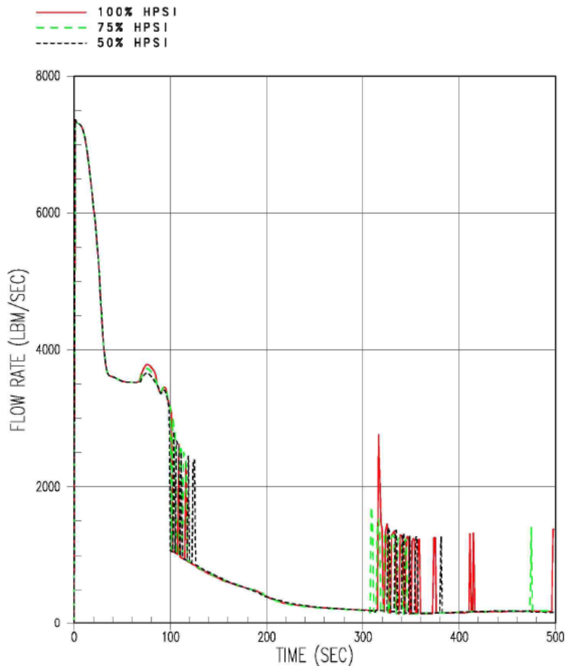

Figure 15.06.05-17-5. Break flowrate

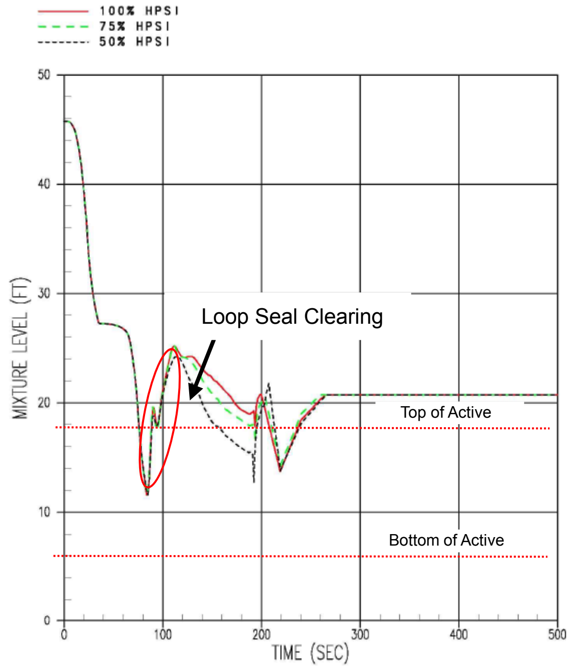

Figure 15.06.05-17-6. Core Mixture Level

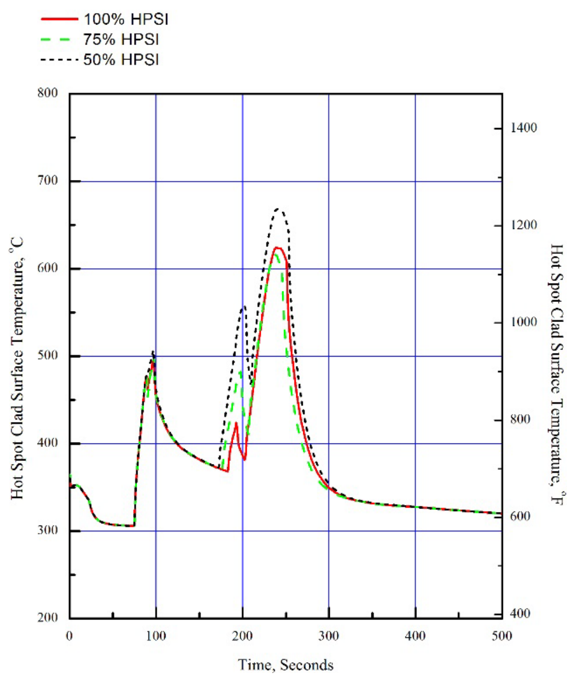

Figure 15.06.05-17-7. Peak Cladding Temperature

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

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Question No. 15.06.05-18

In Section 4, “SBLOCA Analysis”, of the TeR, it is stated that it is conservatively assumed that the offsite power is lost upon reactor trip. The staff would like to understand the basis as to why this is a conservative assumption. If offsite power is not lost, the reactor coolant pumps (RCPs) would continue to run for an extended period resulting in more liquid lost out the break. The applicant is requested to justify that the loss of offsite power upon reactor trip is a conservative assumption vis-à-vis an SBLOCA and provide a supporting analysis.

Response

The result of RCP On/Off sensitivity studies by Combustion Engineering and Westinghouse is as follows.

According to the Combustion Engineering’s CEN-268, Rev.1(Justification of Trip Two/Leave Two RCP Trip Strategy during transient), the RCP trip strategy for the case of 0.1 ft^2 SBLOCA results in no significant core uncovering and virtually no clad temperature heat-up.

In the case of Westinghouse’s WCAP-10054-P-A, the scenarios included are continued pump operation throughout the entire transient and tripping the pump at 10 minutes. The continued pump operation case showed that pump operation results in greater mass depletion than if the pumps tripped case, while maintaining a cooled core. For the 10-minute trip case, NOTRUMP showed that a delayed pump trip could result in a deeper core uncovering than an FSAR trip case. However, the delayed pump trip also caused the accumulator injection setpoint to be reached earlier.

The representative case of the RCP on/off experiment is the LOFT L3-5/6 test series. According to the results of the L3-5/6 experiment and evaluation, the results of RCP-on(L3-6)

show greater mass depletion than RCP-off(L3-5). However, the operation of RCP has a role to provide coolant continuously and keep the core maintained in a saturation state even if the core level is lower than in the RCP-off case and also there is no clad temperature heat-up.

There are two sensitivity cases supporting analyses using RELAP5. One is the case of RCP-off upon reactor trip and the other one is tripping the pump at 5 minutes. The comparison results of the two cases are shown in Figures 15.06.05-18-1 thru 7.

The case of RCP-on(No LOOP Applied) shows the high core pressure (Figure 15.06.05-18-1) and more break flow rate (Figure 15.06.05-18-2) than the case of RCP-off(LOOP Applied). And also, RCP-on case shows the lower core water level (Figure 15.06.05-18-3) and earlier loop seal clearing (Figure 15.06.05-18-6) than RCP-off case. In addition, the core void fraction (Figure 15.06.05-18-4) for the case of RCP-on case shows no core uncovery other than RCP-off case. Therefore, peak cladding temperature occurred in case of RCP-off case. The reason for this is that operation of the RCP has a role to provide coolant continuously and keep the core maintained in a saturation state even if the core level is lower than RCP-off case.

In conclusion, the assumption of LOOP upon reactor trip is conservative during SBLOCA analysis.

TS

Figure 15.06.05-18-1. Core Pressure

Figure 15.06.05-18-2. Break Flow Rate

TS

Figure 15.06.05-18-3. Core Collapsed Level

Figure 15.06.05-18-4. Core Void Fraction

TS

Figure 15.06.05-18-5. Loop Seal Clearing – Loop Applied

Figure 15.06.05-18-6. Loop Seal Clearing – No Loop Applied

TS

Figure 15.06.05-18-7. Peak Cladding Temperature

Impact on DCD

There is no impact on the DCD.

Impact on PRA

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Question No. 15.06.05-20

According to Table 15.6.5-7, the low pressurizer pressure reactor trip setpoint used in the SBLOCA analyses is 109.3 kg/cm^{2}A (1555 psia). A comparison of Figures 15.6.5-27A and 15.6.5-27B suggests that the reactor tripped at a pressure of about 129 kg/cm^{2}A (1820 psia). Please explain the apparent discrepancy between the low pressure reactor trip value given in Table 15.6.5-7 and the value indicated from the normalized power and inner vessel pressure plots.

Response

The reason that reactor pressure and normal power decreased at a pressure of about 129 kg/cm^{2}A (1820 psia) is that the inner vessel two-phase mixture level in Figures 15.6.5-27E reached the top of the hot leg elevation(about 8.4m). So steam in the reactor could be released through the hot leg and then, the reactor tripped at 109.3 kg/cm^{2}A (1555 psia).

TS

Figure 15.6.5-27E 46.5 cm^2 (0.05 ft^2) Break in Pump Discharge Leg: Inner Vessel Two-Phase Mixture Level

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There is no impact on the DCD.

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Question No. 15.06.05-21

SRP Section 6.3, "Emergency Core Cooling System," Review Procedure 22.A states that the lower limit of break size for which emergency core cooling system (ECCS) operation is required needs to be established, which is also the maximum break size for which normal reactor coolant makeup systems (i.e. chemical and volume control system (CVCS)) can maintain reactor pressure and coolant level.

The capability of the ECCS to actuate and perform at this lower limit of break size needs to be confirmed. Currently, the applicant has analyzed small break LOCAs down to and including a break size of 18.6 cm^2 (0.02 ft^2). Breaks smaller than 18.6 cm^2 have not been presented to the staff for review. It is unclear to the staff if 18.6 cm^2 is the smallest small break for which the ECCS actuates and performs satisfactorily, i.e., it is unclear if a break smaller than 18.6 cm2 can be handled by the CVCS system.

The staff needs to confirm that the ECCS actuates and performs satisfactorily at the lower limit of SBLOCAs. Please provide information regarding:
1. The smallest break size for which the APR1400 ECCS is used;
2. The calculation which shows the largest break size that is handled by normal CVCS;
3. The analysis results for the smallest break size event in which the ECCS is used for mitigation;
4. If 18.6 cm^2 is indeed the smallest break size for which the APR1400 ECCS is used, the staff only needs to see a calculation showing that the CVCS can handle break sizes up to 18.6 cm^2.

Response

1. The smallest break size used is the 0.02 ft^2 DVI line break. On the other hand, the only hand calculation without CENPD codes is applied for the evaluation of instrument tube rupture. The break size of instrument tube rupture is 0.003 ft^2. Therefore, the smallest break size for which the APR1400 ECCS is used is 0.003 ft^2.

2. The largest break size that is handled by normal CVCS is 0.00032 ft^2 (0.0461 in^2) for the APR1400. During normal operation, only one charging pump is available. The largest break size was determined by equivalent break area with charging pump flow rate at normal operating condition. The result of calculation is as follow.

Amin = Wcharging pump / Gc
where, Wcharging pump = 1 (pump) × 70.4 = 70.4 [gpm]
Gc = critical mass flux from Henry Fauske-Moody model

Thermal properties at cold leg condition are
Temperature = 555.0 ^oF
Pressure = 2250.0 psia
Density = 46.533 lbm/ft^3
Enthalpy = 553.40 Btu/lbm

From Henry Fauske-Moody critical flow model, use h = 553.40 [Btu/lbm] as stagnation enthalpy.

Pressure(psia)	Enthalpy(Btu/lbm)	Mass flux(lbm/ft^2sec)
2200	487.511	26883.875
	643.400	16218.000
2400	464.778	29708.444
	674.000	15746.000
then, 2200	553.40	22375.771
2400	553.40	23794.249

Therefore, at 2250 psia, Gc = 22730.391 [lbm/ft^2sec]

Amin = W × Density / Gc / 448.83 = 70.4 × 46.533 / 22730.391 / 448.83
= 0.00032[ft^2]
= 0.0461 [in2]

where, 1.0 [gpm] × 2.228E-03 = 1 [ft^3/sec]

3. The smallest break size event in which the APR1400 ECCS is used for mitigation is 0.02 ft^2 DVI line break and the results are provided in the Section 15.6.5 in APR1400 DCD.

4. During SBLOCA, the major phenomena are break flowrate and loop seal clearing and the major parameters are core level and peak cladding temperature. For the cases of break size under the 18.6 cm^2 (0.02 ft^2), the core uncovery and the loop seal clearing do not occur during SBLOCA. So, the fuel rod heat-up caused by core uncovery(boil-off) is no longer generated below this break size. And also, the peak cladding temperature caused by the effect of departure nucleate boiling (DNB) occurring right after reactor trip does not increase any more under the break size of 18.6 cm^2 (0.02 ft^2). Therefore, the smallest break size during SBLOCA of APR1400 is determined as 18.6 cm^2 (0.02 ft^2).

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environment Report.