Non-Proprietary

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.: **394-8460**

SRP Section: **06.02.01.03 – Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)**

Application Section: **6.2.1.3**

Date of RAI Issue: **02/03/2016**

Question No. 06.02.01.03-8

SRP Section 6.2.1.3 Acceptance Criterion No. 1C(iii) specifies that the calculations of liquid entrainment, i.e., carryout rate fraction (CRF), which is the mass ratio of liquid exiting the core to the liquid entering the core, should be based on the pressurized water reactor (PWR) full length emergency cooling heat transfer experiments. The DCD or TeR do not document the technical basis of the CRF values used in the M&E calculations. The applicant is requested to justify their CRF selection.

Response

The carryout rate fraction (CRF) values used in the reflood M/E analysis are based on the acceptable approach defined in Acceptance Criterion No. 1C(iii) of the NUREG-0800 Section 6.2.1.3. The values of the CRF are provided in Section 3.6.f of the TeR (APR1400-Z-A-NR-14007, Rev. 0)

The TeR (APR1400-Z-A-NR-14007, Rev. 0) will be revised to include a statement referencing the guideline set forth in NUREG-0800 under Section 6.2.1.3 as a technical basis for the CRF values used in the report.

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

Technical Report (APR1400-Z-A-NR-14007, Rev. 0) Section 3.6.f will be revised as indicated in the Attachment to this response.

i. The main feedwater isolation valves(MFIVs) are assumed to close only after the generation of a main steam isolation signal (MSIS) of [ ]TS containment pressure. This signal occurs very rapidly (~1 second). MFIV closure is assumed to take [ ]TS, with an additional allowance of [ ]TS for the MSIS signal delay. Feedwater flow and enthalpy are kept at their normal values due to the short times involved. As an additional conservatism, the feedwater is assumed to be added at the end of blowdown, so that the steam generator secondary temperatures during the blowdown are not lowered by the relatively cold feedwater. Note that the feedwater is hot relative to potential peak LOCA containment temperatures so that feedwater addition at the end of blowdown is conservative both for blowdown and for reflood calculations.

j. Auxiliary feedwater flow is conservatively omitted since it is cold [ ]TS relative to both blowdown and reflood conditions.

3.6 Description of Core Reflood Model

Reflood mass and energy release rates are calculated using the FLOOD3 computer code (Reference 3). The hydraulic network of this code is presented in Figure 3-2 for the reflood analysis of LOCA discharge leg break. Heat transfer is conservatively modeled for core, vessel walls, vessel internals, loop metal, steam generator tubes, steam generator secondaries, and steam generator secondary walls. The FLOOD3 code hydraulics calculates flow rates and pressure. The heat transfer process predicts fluid enthalpies. Fluid densities are calculated as functions of pressures and enthalpies. The conservatisms in the model are as follows:

a. The containment backpressure during reflood is assumed to be [ ]TS and constant. It is given for the input of FLOOD3 code.

b. A one-dimensional heat transfer model is used for all wall heat transfer calculations. This is demonstrated in Reference 4 where comparisons of one-dimensional models and otherwise identical two-dimensional models show that one-dimensional modeling is more conservative.

c. A nucleate boiling heat transfer coefficient of [ ]TS is used to model the heat transfer from the steam generator tubes to the primary coolant. This coefficient represents an upper limit, and is conservatively used at all times throughout the tubes.

d. During reflood, the behavior of steam generator liquid level is calculated. The liquid level is predicted to be decreased due to the reversed heat transfer and a part of tube area is in contact with the secondary steam. The heat transfer coefficient of steam-to-tube area is the Nusselt condensation heat transfer coefficient, and is much higher than that of liquid-to-tube, natural circulation heat transfer coefficient. Therefore, it is conservatively assumed that the whole tube heat transfer area is in contact with the secondary steam. A conservative Nusselt condensation heat transfer coefficient of [ ]TS is used in conjunction with the tube area.

e. The then

The carryout rate fraction (CRF) used during the reflood is based on the guideline in NUREG-0800, Section 6.2.1.3 (Reference 1).

]. This value is also used in calculating secondary to primary heat transfer.

f. The ~~carryout rate fraction (~~CRF~~) used during reflood~~ is constant (0.05) up to the 46 cm (18 in) core level, and linearly increases to 0.8 up to the 61 cm (24 in) core level, and is kept constant at 0.8 until the 3.2 m (10.5 ft) level is reached, which is 0.6096 m (2 ft) below the top of the

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Application Section: 6.2.1.3

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Question No. 06.02.01.03-9

SRP Section 6.2.1.3 Acceptance Criterion No. 1C(iii) asks for a justification for the assumption of steam quenching by comparison with applicable experimental data. No information was provided either to ascertain if the liquid entrainment calculations considered the effect on the CRF of the increased core inlet water temperature caused by steam quenching assumed to occur from mixing with the emergency core cooling system (ECCS) water. The DCD or TeR do not mention whether or not steam quenching was assumed. Acceptance Criterion No. 1C(iii) also suggests to assume the steam leaving the SGs to be superheated to the temperature of the secondary coolant. The DCD does mention the discharge fluid to be superheated steam but does not mention what assumption was made about its temperature. The applicant is requested to include these descriptions into the DCD.

Response

The liquid entrainment (i.e., CRF) used in the reflood analysis is not based on the calculated results from the computer code, FLOOD3, but given as the input based on the acceptable approach defined in Acceptance Criterion No. 1C(iii) of the NUREG-0800 Section 6.2.1.3. The basis and values of the CFR are stated in the Section 3.6.f of TeR (APR1400-Z-A-NR-14007, Rev. 0). Thus, the CRF is not affected by the increase of the core inlet water temperature.

The steam quenching is also conservatively assumed as stated in Section 3.6.i of TeR (APR1400-Z-A-NR-14007, Rev. 0), which is expressed as “the steam condensation.” The assumption in Section 3.6.i of the TeR is summarized as follows:

During reflood, credit is taken for the condensation of steam in the annulus by the cold SI water. For conservatism, credit is not taken when the reactor vessel annulus is not full or when the SI flow rate is too low to thermodynamically condense all of the steam in the

annulus. Thus, no credit is taken for the condensation after the SITs empty or the turndown to low SIT flow by the fluidic device even though the low SIT flow enters into the RCS.

The steam leaving the SGs during the initial time of the reflood phase is predicted to be superheated to the temperature as high as of the steam generator secondary coolant temperature by the FLOOD3 code. The revision of the DCD and TeR pertaining to the temperature of the superheated steam is provided in Attachments 1 and 3. The prediction of the temperature of the superheated steam by the FLOOD3 code is based on the assumptions of the initial condition of the steam generator secondary side and the steam generator secondary-to-primary heat transfer described in items c, d and e of DCD Section 6.2.1.3.4 and TeR Section 3.6. The initial condition of the steam generator secondary side is assumed to have the highest SG secondary pressure of 1020 psia at 102% core power and the saturation condition at the corresponding secondary pressure. Attachments 2 and 4 show the initial steam generator secondary pressure in the DCD and TeR, respectively.

Impact on DCD

DCD Tier 2, Section 6.2.1.3 and Table 6.2.1-20 will be revised as indicated in Attachments 1 and 2 associated with this response.

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

Technical Report (APR1400-Z-A-NR-14007, Rev. 0) Section 3.2 and Table 4-3 will be revised, as indicated in Attachments 3 and 4 associated with this response.

most of the initial primary coolant is released to the containment as a two-phase mixture. Following blowdown, the water for releases is supplied from the safety injection system (SIS).

There is an important distinction between hot leg breaks and cold leg breaks for LOCA post-blowdown analyses. For a hot leg break, the majority of the SIS-supplied water leaving the core can vent directly to the containment without passing through a steam generator. Therefore, there is no mechanism for releasing the steam generator energy to the containment for a hot leg break, and only the blowdown period is considered. For cold leg breaks like discharge leg break and suction leg break, the water passes through a steam generator before reaching the containment so the post blowdown releases to the containment are considered for cold leg breaks.

b. The first post-blowdown period is refill. During refill, the SIS water refills the bottom of the reactor vessel to the bottom of the core. This period is conservatively omitted from the analysis.

c. The second post-blowdown period is the reflood period. During reflood, SIS water floods the core. Reflood is assumed to end when the liquid level in the core is 0.6096 m (2 ft) below the top of the active core. During reflood, a significant amount of the SIS water entering the core is postulated to be carried out of the core by the steaming action of the core-to-coolant heat transfer process. This fluid then passes through a steam generator where reverse (i.e., secondary to primary) heat transfer heats it before it reaches the containment. The residual steam generator secondary energy is sufficient to convert all of this fluid to superheated steam during the initial part of the reflood period. Subsequently, as the steam generators are cooled by this process, there is not enough heat transfer to boil all of the fluid passing through the tubes. This causes the break flow to change from pure steam to two-phase.

heated as high as the secondary coolant temperature

As the entire NSSS cool entually subcooled because the safety injection water is subcooled. The onset of the two-phase release to the containment may or may not occur before the end of reflood; typically, this occurs close to the end of the reflood. The potential

Table 6.2.1-20 (1 of 2)

Initial Conditions for Containment Peak Pressure and Temperature Analysis

Part A. Reactor Coolant Systems (Based on a nominal core power of 3,983 MWt)

Parameter	Value
Reactor Coolant System	
Reactor power level (1), MWt	4,091.86
Average coolant temperature, °C (°F)	312.45 (594.4)
Steam Generator Pressure, kg/cm^2A (psia)	71.713 (1,020)
Mass of reactor coolant system liquid, kg (lbm)	304,767.84 (671,898.0)
Mass of reactor coolant system steam, kg (lbm)	3,025.05 (6,669.1)
Energy in Reactor coolant system liquid plus steam (2), 10^6 kcal (10^6 Btu)	103.04 (408.91)
Energy from feedwater nozzle to MSIV per Steam Generator (2), 10^6 kcal (10^6 Btu)	33.047 (131.14)

c. The second post-blowdown period is the reflood period. During reflood, SIS water floods the core. Reflood is assumed to end when the liquid level in the core is [ ]TS below the top of the active core. During reflood, a significant amount of the SIS water entering the core is postulated to be carried out of the core by the steaming action of the core-to-coolant heat transfer process. This fluid then passes through a steam generator where reverse (i.e., secondary to primary) heat transfer heats it before it reaches the containment. The residual steam generator secondary energy is sufficient to convert all of this fluid to superheated steam during the initial part of the reflood period. Subsequently, as the steam generators are cooled by this process, there is not enough heat transfer to boil all of the fluid passing through the tubes. This causes the break flow to change from pure steam to two-phase.

heated as high as the secondary coolant temperature

As the entire NSSS c… …omes subcooled because the safety injection water … …se to the containment may or may not occur before the end of reflood; typically, it occurs close to the end of reflood. The potential release of subcooled fluid to the containment does not occur during reflood when conservative system parameters are used.

d. The third post-blowdown period is the post-reflood period. During this period, the dominant process is the continued cooling of the steam generators by the SIS water leaving the core. The release to the containment during this period becomes generally two-phase in the earlier stage of this period as the cooling of the steam generators continues. The post-reflood ends when the affected steam generator has essentially reached the containment temperature.

e. The final post-blowdown period is the decay heat period, which begins at the end of post-reflood. During the decay heat period, the dominant mechanisms for release rates are the generation of the decay heat and the cooling of all NSSS metal. The decay heat period ends when the containment pressure and the environmental pressure are essentially equal.

LOCA mass and energy releases are analyzed using the computer codes, CEFLASH-4A and FLOOD3 for the categorized phases. The CEFLASH-4A computer code is used for analysis of the blowdown period and the FLOOD3 computer code is used for analysis of the reflood period. The detailed descriptions of the codes are presented in References 2 and 3, respectively.

The M/E calculated by CEFLASH-4A and FLOOD3 is supplied as input to the GOTHIC computer program (References G-5, G-6 and G-7 in Appendix G) for the containment analysis. Mass and energy release during the decay heat period is calculated directly by GOTHIC and is integrated with the containment analysis. Detailed descriptions of the codes are presented in Appendix A.

3.3 Mass and Energy Release Data

Pipe breaks and locations are assumed to be as follows:

- Double-ended suction leg slot break (DESLSB) in the RCP suction leg.
- Double-ended discharge leg slot break (DEDLSB) in the RCP discharge leg.
- Double-ended hot leg slot break (DEHLSB) in the RCP hot leg.

The break type is assumed to be a slot break that has the break area equivalent to the double-ended break. The largest break area, i.e. the double-ended break area is limiting for a large break LOCA.

Table 4-3 Initial Conditions for Containment Peak Pressure Analysis

(Based on a nominal core power of 3983 MWt)

Parameter	Value
Reactor Coolant System	
- Reactor power level [1)], MWt	4,091.86
- Average coolant temperature, °C (°F)	312.45 (594.4)
- Mass of reactor coolant system liquid, kg (lbm)	304,767.84 (671,898.0)
- Mass of reactor coolant system steam, kg (lbm)	3,025.05 (6,669.1)
- Energy in Reactor coolant system liquid plus steam [2)], 10^6 kcal (10^6 Btu)	103.04 (408.91)
- Energy from feedwater nozzle to MSIV per Steam Generator [2)], 10^6 kcal (10^6 Btu)	33.047 (131.14)

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1) At full power plus 2% uncertainty plus max. RCP power [ ]^{TS}
2) Energy is relative to 0 °C (32°F)

- Steam Generator Pressure, kg/cm^{2}A (psia)	71.713 (1,020)