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. 06.02.01.03-5

Conservatisms in the Limiting LOCA M&E Release Calculations from the Containment Perspective

General Design Criterion (GDC) 50, "Containment design basis," and Appendix K to 10 CFR Part 50, "ECCS Evaluation Models" require, in part, analyzing the most severe consequences for the spectrum of postulated pipe breaks sizes, locations, and single failures. NUREG-0800, SRP Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)" lays out several acceptance criteria to ensure that that containment mass and energy (M&E) release calculations are performed for the worst design basis accident (DBA). In this regard, the staff seeks the following information to address concerns about the conservative treatment of M&E release calculations for the limiting LOCA analysis from the containment perspective. The applicant is also requested to update the APR1400 DCD or the KHNP Technical Report (TeR) APR1400-Z-A-NR-14007-P/NP (LOCA Mass and Energy Release Methodology) to document the explanations. (The regulatory bases identified in the above are applicable to all subsequent questions in this RAI.)

SRP Section 6.2.1.3 Acceptance Criterion No. 1C(ii) suggests that mass release rates should be calculated using a model that has been demonstrated to be conservative by comparison to experimental data. Even though the DCD and TeR mention that the LOCA M&E release is analyzed using the computer codes CEFLASH-4A and FLOOD3 for the blowdown and reflood/post-reflood periods respectively, they do not comment on whether these codes have been validated against experimental data. The applicant is requested to document this information in the DCD.

Response

The mass release calculation model in the CEFLASH-4A code during blowdown period has been demonstrated to be conservative by comparison to experimental data in the reference, Section III.C.1.b.(4) of CENPD-132P (Calculative Methods for the C-E Large Break LOCA Evaluation Model, Aug. 1974). The comparison was performed between the CEFLASH-4A analytical results and the experimental results of LOFT semiscale test 850. The three separate critical flow formulations are considered for the mass release calculation model in the CEFLASH-4A, the Moody, the modified Henry/Fauske and a combination of the modified Henry/Fauske and Moody. The detailed descriptions are provided in Attachment 1.

During reflood/post-reflood periods, no critical break flow is predicted and the mass release calculation in the FLOOD3 code is performed using the flow resistances in the hydraulic network presented in Figure 1 of Reference 10 in DCD Section 6.2.9. Since the flow resistances in the hydraulic network were conservatively considered in the calculation of the break flow, i.e. minimized, though the experimental data of reflood break flow is not available, the mass release calculation in the FLOOD3 code is conservative.

The DCD and TeR will be revised for this information as Attachment 2 and 3.

Impact on DCD

DCD Tier 2, Section 6.2.1.3.3 will be revised, as indicated in Attachment 2 to 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, Re.0, “LOCA Mass and Energy Release Methodology,”) Section 3.5 will be revised, as indicated in Attachment 3 to this response.

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(4) C-E Comparison Studies

As part of the evaluation of the appropriate discharge model, Combustion has conducted a study using C-E's CEFLASH-4A code to predict the result of the LOFT semiscale test 850. The purpose of this study was to gain some insight into the influence of various critical flow formulations and to compare these analytical predictions to experimental results.

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(5) Internal Path Critical Flow

During the blowdown of a pressurized, high temperature system, critical flow may occur in the internal passages as well as at the break location. The critical flow rates in these passages are generally less than the break flow rate because of the geometrical differences between these internal passages and a nozzle, for which the break critical flow formulations were developed. It is appropriate, therefore, that the internal path critical flow rates in CEFLASH-4A be chosen to account for these differences.

The flow rates in the internal paths in CEFLASH-4A are determined from the conservation of momentum equation. The upper limits on these flow rates are taken from the same formulations as are employed for the break critical flow rate (see above). This procedure has been retained in the new LOCA Evaluation Model following discussions with the Division of Licensing Staff.

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Figure III.C.1-7

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Figure III.C.1-8

sources and amounts of stored energy is given in Table 6.2.1-38. The items of energy sources are broken down into the detail specific sources.

6.2.1.3.3 Description of Blowdown Model

Blowdown M&E release rates are calculated using the CEFLASH-4A computer code. A description of the CEFLASH-4A code, including the conservatisms in modeling, is given below. This section includes the justification of the heat transfer correlations. The following assumptions are used in selecting input data for the code. Detailed descriptions of the assumptions of blowdown model are provided in Reference 3.

a. The CEFLASH-4A code model of the heat transfer in a node allows only one wall per node.

The calculation model of the code has been demonstrated to be conservative by comparison to experimental data in the CENPD-132P, Section III.C.1.b.(4).

b. … t capacitance of all the RCS walls for effective heat transfer.

c. The conservative conductivity of carbon steel rather than that of stainless steel is used for the entire wall.

d. Wall surface heat transfer coefficients are assumed to be infinite.

e. All primary water volumes are conservatively increased by including water level instrumentation error, pressure and temperature expansion of volume.

f. Two-phase heat transfer correlation (Jens Lottes) is used for the core-to-coolant heat transfer whenever the flow through the core is not pure steam.

g. Heat transfer across the steam generator tubes is modeled with the same heat transfer coefficient in both the forward and reverse directions.

h. The turbine stop valves are assumed to close at 0.01 second. This is conservative because it keeps energy within the NSSS, which is a source of energy for containment pressurization.

For the core decay heat curve as a fraction of the initial power level following the accident; a [ ]TS conservatism factor is used for the first [ ]TS, followed by a [ ]TS factor thereafter. The normalized decay heat curve is shown in Figure 4-1.

Initial conditions in the reactor coolant system are given in Table 4-3. It shows various stored energies in the RCS at the initial time.

3.5 Description of Blowdown Model

Blowdown mass and energy release rates are calculated using the CEFLASH-4A computer code (Reference 2). The node diagram of this code is presented in Figure 3-1 for the blowdown analysis of LOCA discharge leg break. A description of the CEFLASH-4A code including the conservatisms in modeling is given below. This section includes justification of the heat transfer correlations. The following assumptions are made in selecting input data for the code.

The calculation model of the code has been demonstrated to be conservative by comparison to experimental data in the CENPD-132P, Section III.C.1.b.(4).

a. The CEFLASH-4A code model of the heat transfer in a node allows only one wall per node. Accordingly, the thickness used for the "U" factor for each node wall is selected so that the energy released from the system is conservatively modeled.

b. The CEFLASH-4A ... reactor coolant sy... some of the walls ... example, the geometry of the actual flow path allows some components to partially shield others from the flow. This effect is conservatively omitted from the modeling.

c. Although much of the steel facing the coolant in the reactor coolant system is stainless cladding [ ]TS, a conservative carbon steel conductivity [ ]TS, is used for the entire wall. This conservatively overpredicts the energy released from all such walls.

d. Wall surface heat transfer coefficients are assumed to be infinite.

e. All primary water volumes are conservatively increased from their nominal design values in order to obtain an upper bound for the available mass and energy in the system prior to LOCA. The pressurizer water volume includes an allowance for level instrumentation error. Pressure and temperature expansion of the reactor coolant system and steam generator to the normal operating condition is included.

f. An accepted (Jens Lottes) two-phase heat transfer correlation is used for the core to coolant heat transfer whenever the flow through the core is not pure steam.

g. Heat transfer across the steam generator tubes is modeled with the same heat transfer coefficient in both the forward and reverse directions. This is conservative since it maintains a nucleate boiling heat transfer coefficient on the secondary side during the LOCA blowdown. In reality, the reactor trip following the LOCA would result in a turbine trip that would close the turbine stop valves and then the heat transfer in the secondary side would be through natural convection, in which the heat transfer coefficient has a small value. However, in this analysis, it is conservative to assume the overall heat transfer coefficient in the initial steady state at full-power operation since it maximizes the reverse heat transfer.

h. The turbine stop valves are assumed to close at [ ]TS. This is conservative since it keeps energy within the NSSS which in turn is a source of energy for containment pressurization.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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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. 06.02.01.03-10

SRP Section 6.2.1.3 Acceptance Criterion No. 1C(iv) asks for a description of the long-term cooling (or post-reflood) model. However, the DCD or TeR do not provide any discussion or justification of the methods used to calculate the core inlet and exit flow rates and removal of the sensible heat from primary system metal surfaces and the steam generators (SGs). Liquid entrainment correlations for fluid leaving the core and entering the SGs are neither described nor justified by comparison with experimental data. No statements are made about steam quenching by ECCS water or the applicable experimental data, or whether and how all the remaining stored energy in the primary and secondary systems would be removed during the post-reflood phase. No references are made to compare the results of post-reflood analytical models with the applicable experimental data. The applicant is requested to add these descriptions in the DCD, or appropriately reference them.

Response

The post-reflood model is identical to the reflood model except that the CRF is changed from 0.8 to 1.0. The post-reflood transient is a continuation of the reflood model. Therefore, any assumption and method of the reflood phase is applicable to the post-reflood phase as well.

The liquid entrainment (CRF) is assumed to be 1.0 during the post-reflood period, while the applicable value in the NRC SRP Section 6.2.1.3 Acceptance Criterion No. 1C(iii) is 0.05. This assumed CRF of 1.0 is a conservative value which increases the system flow rate and eventually maximizes the break flow during the post-reflood period.

The assumption of steam quenching is also identical to that of the reflood period which is expressed as “the steam condensation”. As mentioned in the item i of Section 3.6 of the TeR (APR1400-Z-A-NR-14007, Rev.0), credit is not taken when the SI flow is too low to thermodynamically condense all of the steam in the annulus. Thus, credit is not taken for the condensation after the turndown to low SIT flow by the fluidic device during post-reflood period.

For the removal of all the remaining stored energy in the primary and secondary systems, the detailed descriptions are provided in the section of the reflood model description, Section 3.6 of the TeR.

During the decay heat phase after the EOPR, which is relatively stable period characterized by decay heat release, the M/E release through the break is calculated using the RCS model based on the GOTHIC lumped-parameter volume approach.

The RCS model has three lumped-parameter volumes that represent a RCS core, a downcomer and a piping (hotleg or coldleg). The RCS core volume contains a thermal conductor that models the RCS metal sensible energy release and three GOTHIC heater components that model heat releases from the core decay heat, metal and coolant energies stored in SGs secondary side. All energies stored in metal and coolant in SGs secondary side are modeled to be directly released to the core coolant and exhausted within 24 hours after accident initiation.

This modeling approach is to demonstrate that the containment pressure at 24 hours of the accident is compliant with the requirement (SRP 6.2.1.1.A) describing that the containment pressure should be reduced to less than 50 % of the calculated peak pressure within 24 hours after the accident initiation.

The RCS model and the analysis method for the decay heat phase are described in detail in the Appendix A, Section A.2.3 and Section A.2.4 of the Technical report APR1400-Z-A-NR-14007-P, Rev. 0 “LOCA Mass and Energy Release Methodology”.

Impact on DCD

DCD Tier 2, Section 6.2.1.3.5 will be revised, as indicated in Attachment 1 to 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, Re.0) Section 3.7 will be revised, as indicated in Attachments 2 to this response.

e. The thermal resistance corresponding to the steam generator tubes is 0.000077 $(kcal/m^2\text{-}hr\text{-}°C)^{-1}$ (0.000376 $(Btu/hr\text{-}ft^2\text{-}°F)^{-1}$). This value is also used in calculating secondary-to-primary heat transfer.

f. The carryout rate fraction (CRF) used during the reflood is based on the guideline in NUREG-0800, Section 6.2.1.3 (Reference 14).

g. Reflood is assumed to terminate when the 3.2 m (10.5 ft) quench level in the core is reached.

h. As conservatism for the available energy sources, 120 percent of the standard decay heat curve in Figure 6.2.1-32 is used up to 1,000 seconds, and 110 percent is used after 1,000 seconds.

i. For the suction leg and discharge leg cases, credit is taken for the condensation of approximately 42 percent of the total steam flow when the annulus is full and the high safety injection tank (SIT) flow is injected. No credit is taken for the condensation after the SITs empty or the turndown to low SIT flow by the fluidic device.

6.2.1.3.5 Description of Post Reflood Model

The post-reflood model is identical to the reflood model except that at the end of the reflood, the CRF is changed from 0.8 to 1.0. The change conservatively increases the system flow rates due to the increased CRF. The flow rates are further enhanced by the fact that the core liquid height is now constrained at the 3.2 m (10.5 ft) level, which maximizes the available driving head between the annulus level and the core in the flooding equation. All of the heat transfer coefficients are kept at the values used for the reflood analysis. Condensation is analyzed as previously described; however, there is insufficient ~~spillage~~ SIT flow for complete thermodynamic condensation of the steam so that credit for condensation is not taken.

active core. Other variables, such as core inlet temperature, pressure, flow rate, linear heat rate, or other experimental data are not used to determine the CRF.

g. Reflood is assumed to terminate when the 3.2 m (10.5 ft) quench level in the core is reached.

h. [ ]TS of the standard decay heat (Figure 4-1) curve is used as a conservatism for the available energy sources.

i. During reflood, credit is taken for the condensation of steam in the annulus by the cold SIS water. As a conservatism, credit is not taken unless the reactor vessel annulus is full since the SI flow is injected directly into the annulus. Also, as an additional conservatism, credit is not taken when the SI flow rate is too low to thermodynamically condense all of the steam in the annulus. Thus, credit is not taken for the condensation after the SITs empty or the turndown to low SIT flow by the fluidic device. The percentage of the total steam flow condensed varies slightly with time for each case. For suction leg and discharge leg cases, credit is taken for the condensation of approximately [ ]TS of the total steam flow when the annulus is full and the thermodynamic criteria are simultaneously met.

3.7 Description of Post-Reflood Model

The post-reflood model is identical to the reflood model except that, at the end of reflood, the CRF is changed from [ ]TS to [ ]TS. This conservatively increases the system flow rates due to the increased CRF. The flow rates are further enhanced by the fact that the core liquid height is now constrained at the [ ]TS level, which maximizes the available driving head between the annulus level and the core in the flooding equation. All heat transfer coefficients are kept at the values used for the reflood analysis. Condensation is analyzed as previously described; however, there is insufficient ~~spillage~~ SIT flow for complete thermodynamic condensation of the steam so that credit for condensation is not taken.

3.8 Description of Decay Heat Phase Model

The final phase of the large break LOCA is a relatively stable period characterized by decay heat release and it extends from the end of the post-reflood phase. The analysis method used to determine the mass and energy released during this period is described in Appendix A.2.4, "Decay Heat Phase M/E Analysis Model."

3.9 Single Active Failure Analysis

Two possible failures are considered as single failure in LOCA mass and energy analysis, the failure of one SI pump and the failure of one emergency diesel generator (EDG). Both failures would degrade SI flow and eventually degrade emergency core cooling system (ECCS) performance to cool down the core. In LOCA mass and energy analysis, the single failure is assumed for minimum SI flow and no failure is assumed for maximum SI flow.

Another failure in the containment system is considered as a single failure, the failure of one train of containment spray. The failure reduces the capability to suppress containment pressure, which results in higher containment pressure during the LOCA transient. In the case with maximum SI, the failure of one train of containment spray system (CSS) is assumed. In the case with minimum SI, the failure of one train of containment spray system is assumed also, due to the failure of one emergency diesel generator.

The limiting case is determined by the case analyses with the maximum and minimum safety injection flow. This case analysis with the maximum and minimum safety injection flow is performed at the three break locations.

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Application Section:	**6.2.1.3**
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Question No. 06.02.01.03-11

According to SRP Section 6.2.1.3 Acceptance Criterion No. 1C(v), the fission product decay energy model is acceptable if it is equal to or more conservative than the decay energy model given in SRP Section 9.2.5. However, the DCD or TeR provides no information to ascertain how conservative the decay energy model is compared to the one given in SRP Section 9.2.5. SRP Section 6.2.1.3

Acceptance Criterion No. 1C(v) also suggests that steam from decay heat boiling in the core should be assumed to flow to the containment by the path which produces the minimum amount of mixing with ECCS injection water. No such description is found in the DCD or the TeR. The applicant is request to clarify these two aspects in the DCD.

Response

The decay energy model in the LOCA M/E calculation uses two different decay heat curves both of which are based on the ANSI/ANS 5.1 standard decay curve. One is ANS 5-1971 decay heat curve which is chosen for decay energy releases during the earlier phases of a LOCA (blowdown, reflood and post-reflood) and the other is ANS 5.1-1979 decay heat curve used for the remaining decay heat phase.

For the earlier phases of a LOCA until the EOPR, the ANS standard decay curve (Proposed ANS Standard, “Decay Energy Release Rate following Shutdown of Uranium-Fueled Thermal Reactors,” October 1971.) corrected for decay of the heavy elements U^{239} and Np^{239} has been incorporated into the CEFLASH-4A. Base on Section III.A.4 of CENPD-132, it is found that the fission product decay energy of the ANS standard decay curve does not differ substantially from the ANS 5.1-1979 which is recommended in the SRP Section 9.2.5. The DCD and TeR will be revised for this information as Attachment 1 and 2.

The decay heat curve used during the decay heat phase is based on the ANSI/ANS 5.1-1979 with 2σ uncertainties. The decay heat contribution from actinides other than U^{239} and Np^{239} is additionally taken into account to the decay heat curve for conservatism. Thus, the decay heat model used in the analysis is more conservative compared to that provided in the SRP Section 9.2.5. The detailed description of the decay heat curve chosen for the decay heat phase is provided in the Appendix A, Section A.2.4.1 of the Technical report APR1400-Z-A-NR-14007-P, Rev. 0 “LOCA Mass and Energy Release Methodology”.

During the decay heat phase, the steam coming from the intact SG may be condensed when it contacts subcooled SI water in the downcomer which would decrease coolant evaporating. In the APR1400 containment model, however, it is conservatively assumed that there is no mixing of steam and SI water in the RCS.

In the RCS model for the LOCA M/E release during decay heat phase, energy release from coolant and metal of the SGs’ secondary side are modeled using the GOTHIC heater components which are submerged in the RCS core volume. The downcomer of the RCS model receives the IRWST water through the SIP then feeds it to the core volume as needed to make up for steaming and returns the remaining water to the IRWST volume as spillage without temperature increase. This modeling approach basically excludes mixing of the steam with the ECCS injection water, consequently maximizes steaming in the core.

The RCS model and the M/E release analysis for the decay heat phase are described in the Appendix A, Section A.2.3 and A.2.4 of the Technical report APR1400-Z-A-NR-14007-P, Rev. 0 “LOCA Mass and Energy Release Methodology”.

Impact on DCD

DCD Tier 2, Section 6.2.1.3.2 will be revised, as indicated in Attachment 1 to 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, “LOCA Mass and Energy Release Methodology,” Section 3.4 will be revised, as indicated in Attachment 2 to this response.

For the conservative analysis, the assumptions about the energy sources are biased to maximize the stored energy.

For considering the stored energy in the coolant, the initial RCS water volumes are conservatively calculated based on the maximum manufacturing tolerances of the reactor vessel and steam generator tubes. Volume expansion of the loop components from cold to hot operating conditions is also considered for the primary and secondary coolant stored energy. The initial water volume in the pressurizer includes an allowance for level instrumentation error. This makes the maximized pressurizer water volume, which includes the maximized stored energy.

For considering the stored energy in the primary and secondary walls, the large specific heat and heat conductivity of carbon steel are conservatively assumed for all of the walls in the RCS.

The core stored energy may be increased slightly by thermal conductivity degradation (TCD). However, the effect of TCD on the M&E release is negligible. The results are described in Reference 5.

For considering the energy in the safety injection water, the liquid break flow is assumed to be mixed with the water in IRWST. The mixed water is taken and discharged into the direct vessel injection (DVI) by SI pumps, which increases the energy in the safety injection water.

The initial power level ... RCP power. The init... instrumentation error. ... pressure calculations.

The ANS standard decay curve (Proposed ANS standard, "Decay Energy Release Rate following Shutdown of Uranium-Fueled Thermal Reactors," October 1971) corrected for decay of the heavy elements has been incorporated into the CEFLASH-4A. The fission product decay energy of the ANS standard decay curve does not differ substantially from the ANS 5.1-1979 in the Acceptance Criterion of Reference 14.

For the core decay heat curve as a fraction of the initial power level following the accident, a 20 percent conservatism factor is used for the first 1,000 seconds, followed by a 10 percent factor. The normalized decay heat curve is shown in Figure 6.2.1-32.

Initial conditions in the reactor coolant system are given in Table 6.2.1-20. It shows various stored energies in the RCS and containment at the initial time. A tabulation of

For the core decay heat curve as a fraction of the initial power level following the accident; a [ ]TS conservatism factor is used for the first [ ]TS, followed by a [ ]TS factor thereafter. The normalized decay heat curve is shown in Figure 4-1.

Initial conditions in the reactor coolant system are given in Table 4-3. It shows various stored energies in the RCS at the initial time.

The ANS standard decay curve (Proposed ANS standard, "Decay Energy Release Rate following Shutdown of Uranium-Fueled Thermal Reactors," October 1971) corrected for decay of the heavy elements has been incorporated into the CEFLASH-4A. The fission product decay energy of the ANS standard decay curve does not differ substantially from the ANS 5.1-1979 in the Acceptance Criterion of Reference 14.

3.5 Description of Blow

Blowdown mass and energy (Reference 2). The node dia LOCA discharge leg break. modeling is given below. Th assumptions are made in se

a. The CEFLASH-4A code model of the heat transfer in a node allows only one wall per node. Accordingly, the thickness used for the "U" factor for each node wall is selected so that the energy released from the system is conservatively modeled.

b. The CEFLASH-4A wall representation uses the total heat capacitance of all the walls in the reactor coolant system that actually face a given node. This is conservative since, in reality, some of the walls will not participate as effectively as others in the heat transfer process. For example, the geometry of the actual flow path allows some components to partially shield others from the flow. This effect is conservatively omitted from the modeling.

c. Although much of the steel facing the coolant in the reactor coolant system is stainless cladding [ ]TS, a conservative carbon steel conductivity [ ]TS, is used for the entire wall. This conservatively overpredicts the energy released from all such walls.

d. Wall surface heat transfer coefficients are assumed to be infinite.

e. All primary water volumes are conservatively increased from their nominal design values in order to obtain an upper bound for the available mass and energy in the system prior to LOCA. The pressurizer water volume includes an allowance for level instrumentation error. Pressure and temperature expansion of the reactor coolant system and steam generator to the normal operating condition is included.

f. An accepted (Jens Lottes) two-phase heat transfer correlation is used for the core to coolant heat transfer whenever the flow through the core is not pure steam.

g. Heat transfer across the steam generator tubes is modeled with the same heat transfer coefficient in both the forward and reverse directions. This is conservative since it maintains a nucleate boiling heat transfer coefficient on the secondary side during the LOCA blowdown. In reality, the reactor trip following the LOCA would result in a turbine trip that would close the turbine stop valves and then the heat transfer in the secondary side would be through natural convection, in which the heat transfer coefficient has a small value. However, in this analysis, it is conservative to assume the overall heat transfer coefficient in the initial steady state at full-power operation since it maximizes the reverse heat transfer.

h. The turbine stop valves are assumed to close at [ ]TS. This is conservative since it keeps energy within the NSSS which in turn is a source of energy for containment pressurization.