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.:	**240-8318**
SRP Section:	**15.01.01-15.01.04 - Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve**
Application Section:	**15.01.01-15.01.04**
Date of RAI Issue:	**10/13/2015**

Question No. 15.01.01-1

GDC 10 requires that the reactor coolant system (RCS) is designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences (AOOs). Additionally, sections 15.1.1-15.1.4 of the Standard Review Plan (NUREG-0800) state that the parameters used in the analytical model should be suitably conservative.

DCD Section 15.1.1 "Decrease in Feedwater Temperature" states that the maximum decrease in feedwater temperature due to a failure in the main feedwater system is less than 37.78 ^{o}C (100 ^{o}F), but it is not described how this value is obtained. This caused NRC staff to question whether this value is a bounding input for the analysis. NRC staff is requesting the applicant to explain why a decrease in feedwater temperature of 37.78 ^{o}C (100 ^{o}F) represents a bounding case.

Response

APR1400 DC high pressure feedwater heater system consists of 6 heaters with three stages two parallel trains. With loss of one train of heaters, the other heater train can pass 75% of full power feedwater. The feedwater temperature decrease (ΔT) is evaluated to be less than 55. ^{o}C (100 ^{o}F) according to the system design criteria.
Therefore, the use of 55.6 ^{o}C (100 ^{o}F) as a decrease in feedwater temperature represents a bounding case, because this is the event that could result in an increase in the rate of heat removal by the secondary system, which could lead to a temperature decrease in the reactor coolant system (RCS).
In addition, an editorial error will be corrected in Section 15.1.1.1.

Impact on DCD

DCD Chapter 15.1.1.1 will be revised as indicated on the attached markup.

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.

15.1 Increase in Heat Removal by the Secondary System

This section describes analyses that have been performed for events that could result in an increase in the rate of heat removal by the secondary system, which could lead to a temperature decrease in the reactor coolant system (RCS).

Several anticipated operational occurrences (AOO) and one postulated accident (PA) result in an unplanned increase in heat removal by the secondary system. In these events, a decrease in reactor coolant temperature causes an increase in core reactivity that leads to an increase in core power. Detailed analyses of these RCS cooldown events are presented in this section. The events are:

a. Subsection 15.1.1 – Decrease in feedwater temperature

b. Subsection 15.1.2 – Increase in feedwater flow

c. Subsection 15.1.3 – Increase in steam flow

d. Subsection 15.1.4 – Inadvertent opening of a steam generator relief or safety valve

e. Subsection 15.1.5 – Steam system piping failure inside and outside the containment

15.1.1 Decrease in Feedwater Temperature

15.1.1.1 Identification of Causes and Frequency Classification

A decrease in feedwater temperature may result from a loss of feedwater heaters. The feedwater heaters may be lost due to isolation of one of two high-pressure feedwater heater trains. The maximum decrease in feedwater temperature due to a failure in the main feedwater system is less than ~~37.78~~ °C (100 °F). A LOOP concurrent with a turbine trip is considered a basic assumption.

55.6

A decrease in feedwater temperature event is classified as an AOO. Event frequency conditions are described in Subsection 15.0.0.1 and Table 15.0-5.

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Application Section:	**15.01.01-15.01.04**
Date of RAI Issue:	**10/13/2015**

Question No. 15.01.01-2

GDC 10 requires that the reactor coolant system (RCS) is designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences (AOOs). Additionally, sections 15.1.1-15.1.4 of the Standard Review Plan (NUREG-0800) state that the parameters used in the analytical model should be suitably conservative.

Design control document (DCD) Sections 15.1.1.3.2, 15.1.2.3.2, and 15.1.3.3.2 state that the input parameters and initial conditions for the Decrease in Feedwater Temperature, Increase in Feedwater Flow, and Increase in Steam Flow events are bounded by the input for the Inadvertent Opening of a Steam Generator Safety Relief or Safety Valve event. Although the results of the analyses may demonstrate that the consequences of one event are limiting (i.e. bound the consequences of the other events), the input parameters and initial conditions for each event need to be chosen to be suitably conservative. NRC staff requests that DCD Sections 15.1, 15.2, and 15.3 be updated to reflect that suitably conservative input parameters and initial conditions were used in the evaluation of the Decrease in Feedwater Temperature, Increase in Feedwater Flow, and Increase in Steam Flow events.

Response

Table 15.0-2 of DCD shows the range of the parameters used for Chapter 15 events, and the combined initial conditions are used to get more adverse results for the relevant events. The events corresponding to Sections 15.1.1, 15.1.2, 15.1.3 and 15.1.4 are those that could result in an increase in the rate of heat removal by the secondary system, and it means that the input parameters and initial conditions would have a same tendency with respect to the analysis results. Although the input parameters and initial conditions for each event need to be chosen to be suitably conservative, the results of Section 15.1.4 would demonstrate that the consequences of Section 15.1.4 are limiting, because the evaluation model for IOSGADV would be limiting.

Therefore, the sentence of DCD Sections 15.1.1.3.2, 15.1.2.3.2, and 15.1.3.3.2 would be enough to represent the relevant conservatism.

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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Question No. 15.01.01-3

GDC 10 requires that the reactor coolant system (RCS) is designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences (AOOs). Additionally, sections 15.1.1-15.1.4 of the Standard Review Plan (NUREG-0800) state that the parameters used in the analytical model should be suitably conservative.

DCD Section 15.1.2 "Increase in Feedwater Flow" states:

1. The maximum increase in main feedwater flow at full power is less nominal flow for the main feedwater system (i.e. the limiting increase in main feedwater flow is 100%), and
2. The maximum auxiliary feedwater flow is 950 gpm

However, there is no discussion regarding how these values are obtained. This caused NRC staff to question whether these values are bounding inputs for the analysis. NRC staff requests the applicant to explain why the values provided in the DCD are suitably conservative.

Response

1. The sentence “the maximum increase at full power is less than nominal flow for the main feedwater system” means the total feedwater flow is less than 200% nominal flow.

2. As described in DCD Section 10.4.9.2.2.4, a cavitating venturi is located in the common AFW supply line to each steam generator. The cavitating venturi limits the maximum AFW flow by 900 gpm as described in Table 10.4.9-1. Therefore, the maximum auxiliary feedwater flow of

950 gpm considered in the DCD Section 15.1.2 is conservative, because this is the event that could result in an increase in the rate of heat removal by the secondary system, which could lead to a temperature decrease in the reactor coolant system (RCS).

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:	**15.01.01-15.01.04**
Date of RAI Issue:	**10/13/2015**

Question No. 15.01.01-4

GDC 10 requires that the reactor coolant system (RCS) is designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences (AOOs). Additionally, sections 15.1.1-15.1.4 of the Standard Review Plan (NUREG-0800) state that the parameters used in the analytical model should be suitably conservative.

DCD Section 15.1.3 "Increase in Steam Flow" states that the inadvertent opening of a turbine admission valve can result in an increase in no more that 11 percent of the nominal full-power steam flow rate, but it does not described how this value is obtained. This caused NRC staff to question whether this value is a bounding input for the analysis. NRC staff is requesting the applicant to explain why 11 percent is a bounding increase in steam flow for the inadvertent opening of a turbine admission valve.

Response

All TBN admission valves are sized to admit total 110% of full power steam when the valve is wide open. Also each main steam system valve in the steam line is sized not to exceed 11% of full power steam. Therefore, increase in steam flow in DCD section 15.1.3 is limited by 11% of full power steam.

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.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

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Application Section:	**15.01.01-15.01.04**
Date of RAI Issue:	**10/13/2015**

Question No. 15.01.01-5

GDC 10 requires that the reactor coolant system (RCS) is designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences (AOOs). Additionally, sections 15.1.1-15.1.4 of the Standard Review Plan (NUREG-0800) state that the parameters used in the analytical model should be suitably conservative.

DCD Section 15.1.4 "Inadvertent Opening of a Steam Generator Relief or Safety Valve" presents the bounding analysis for AOOs that result in an increase in heat removal by the secondary system. The analysis provided in DCD Section 15.1.4 lacks sufficient detail for NRC staff to understand some of the input assumptions. This caused NRC staff to question whether the input values are suitably conservative. NRC staff requests the following information:

1. DCD Section 15.1.4.3 states that initial conditions for the principle process variables are varied to determine the set of initial conditions that would produce the greatest overpower condition caused by the increase in steam flow. However, it is not necessarily true that the conditions that produce the greatest overpower condition result in the limiting case for the figures of merit that reflect the specified acceptance criteria (i.e. DNBR, peak pressure in the reactor coolant and main steam systems). NRC staff requests that DCD Section 15.1.4 be updated to reflect the analysis that produces the limiting case in terms of the specified acceptance criteria.
2. DCD Section 15.1.4.2 states the limiting single failure is the failure of the feedwater control system to receive the reactor trip override (RTO) signal to cut back the feedwater flow. However, there is no explanation regarding RTO logic in either DCD Sections 15.1.4 or 7.7, and no reference is cited. NRC staff requests that the DCD be updated with a description of the RTO logic and an explanation regarding how it was determined to be the limiting single failure.

Response

1. The initial conditions for the major parameters are varied to determine the set of initial conditions that would produce the greatest overpower condition caused by the increase in steam flow, which results in a conservative minimum DNBR.

   DCD Section 15.1 events are categorized as those events that would cause an increase in heat removal by the secondary system. The results of the core and system evaluation case demonstrate that the RCS pressure and main steam system pressure remain well below 110% of the relevant system design pressure because these events have depressurization characteristics. Each DCD Section 15.1.1.3.1, 15.1.2.3.1, and 15.1.3.3.1 says that the evaluation model for an IOAGADV is applicable to the relevant event, therefore, there is no need to revise the relevant DCD, because DCD Section 15.1.4 provides the analysis that produces the limiting case in terms of the specified acceptance criteria.

2. Even though Table 15.0-2 shows available reactor trip functions for the relevant events, the relevant sections of the DCD Chapter 15 do not provide a description of the logic considered for the event. Subsection 7.7.1.1.c and Figure 7.7-5 provides the major functions of FWCS except RTO and HLO, and the schematic diagram, respectively. The reason for not discussing in Section 7.7 is that RTO and HLO are the subsidiary functions, not main control functions of the FWCS and it is very complicated to show all functions in the Fig 7.7-5. Therefore, it is also not suitable that the DCD Section 15.1.4.2 should be updated with a description of the RTO logic. However, the relevant sentence will be modified to clarify the meaning.

   The single failure which yields the minimum transient hot channel DNBR is the limiting single failure which combines the greatest decrease in DNBR after initiation of a reactor trip signal, with the lowest possible pre-trip DNBR. The failure of RTO could result in over cooling the RCS by providing excessive FW to the steam generator at the time of reactor trip. Therefore, the failure of RTO is determined as the limiting single failure with respect to minimum DNBR.

Impact on DCD

DCD Section 15.1.4.2 will be revised as indicated on the attached markup.

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.

natural circulation and decreases the coolant temperature. The main steam isolation signal (MSIS) results in the isolation of the unaffected steam generator from the flow path through the ADV, which is stuck open.

After tripping the reactor, the operator manually closes the inadvertently opened ADV, terminating the steam release to the atmosphere from the affected steam generator. The analysis conservatively assumes that the action to close the ADV is delayed 20 minutes beyond the operator's initial action to trip the reactor or a total of 50 minutes after event initiation. The operator is assumed to initiate plant cooldown 30 minutes after the manual reactor trip. RCS heat removal for plant stabilization and cooldown is accomplished by using the ADVs on the unaffected steam generator.

The reactor trip generates the reactor trip override (RTO) signal, which normally causes the feedwater control system to reduce the feedwater flow. The reduced feedwater flow prevents the RCS from overcooling after the reactor trip. However, because

Case 2: IOSGADV with a loss of fe
LOOP

Until the assumed reactor trip occurs, the transient due to the IOSGADV is identical with or without a single failure. For the IOSGADV+SF event, the reactor is manually tripped 30 minutes following the first indication of the event. A LOOP is assumed to occur, concurrent with the turbine trip following the reactor trip. ~~Because~~ of the single failure, it is assumed that the feedwater control system does not receive the reactor trip override (RTO) signal to cut back the feedwater flow. Therefore, primary and secondary pressures continue to decrease, and the main steam safety valves (MSSVs) fail to open. Primary pressure and temperatures decrease more rapidly after a reactor trip with a single failure.

The operator recognizes the incident based on a variety of indications and manually closes the ADV that had been inadvertently opened, terminating steam release to the atmosphere from the affected steam generator. The indications include the initial large power mismatch between the reactor and turbine, the steady decrease in steam generator pressure and water levels after reactor trip, the continued decrease in pressure in the affected steam generator after the MSIS, the low steam generator pressure alarms, and the audible indication of steam blowdown. The analysis assumes that the initial operator action to close the open ADV is delayed until 20 minutes after the operator's initial action to trip the reactor or a total of 50 minutes after event initiation. The operator is assumed to initiate plant cooldown 30 minutes after a manual reactor trip. RCS heat removal for plant stabilization and cooldown is accomplished by manual control of the ADVs on the unaffected SG.

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Date of RAI Issue:	**10/13/2015**

Question No. 15.01.01-6

GDC 10 requires that the reactor coolant system (RCS) is designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences (AOOs). Additionally, sections 15.1.1-15.1.4 of the Standard Review Plan (NUREG-0800) state that the parameters used in the analytical model should be suitably conservative.

DCD Section 15.1.4 "Inadvertent Opening of a Steam Generator Relief or Safety Valve" presents the bounding analysis for AOOs that result in an increase in heat removal by the secondary system. The analysis provided in DCD Section 15.1.4 lacks sufficient detail for NRC staff to verify the results of the calculation. This caused NRC staff to question whether a suitably conservative analysis was performed. NRC staff requests the following information:

1. Figures 15.1.4-1.1 through 15.1.4-1.15 and Figures 15.1.4-2.1 through 15.1.4-2.15 of the DCD present the salient NSSS parameters for the Inadvertent Opening of a Steam Generator Atmospheric Dump Valve (IOSGADV) event. The abscissas in the figures begin at 0 seconds, which coincides with the time that the atmospheric dump valve is opened. Therefore, it is not possible to view the magnitude of the perturbation to the system resulting from the opening of the atmospheric dump valve (this is most clearly demonstrated in Figures 15.1.4-1.9 and 15.1.4-2.9). NRC staff requests the figures in DCD Section 15.1.4 be updated to present a brief null transient before the initiation of the event.

2. Tables 15.1.4-1 and 15.1.4-2 of the DCD provide the sequence of events for the analyses of the IOSGADV event. These tables, however, are missing significant phenomena that occur during the transient. Furthermore, this phenomena is not discussed DCD Section 15.1.4.2 "Sequence of Events and System Operation" or DCD Section 15.1.4.3.3 "Results". NRC staff requests that the tables be updated, at a minimum, with the following phenomena:
- Steam generator water level reaches auxiliary feedwater actuation analysis setpoint, %WR

(this is missing from Table 15.1.4-2)
- Explain what is causing the significant reduction in RCS pressure that occurs at approximately 2600 seconds
- Explain what is causing the significant increase in RCS pressure that occurs at approximately 2800 seconds

3. Figure 15.1.4-1.11 and 15.1.4-2.11 of the DCD show that the main feedwater enthalpy is not impacted as a result of the IOSGADV event. However, as the feedwater mass flow rate is increased through the feedwater heaters, but the steam flow through the feedwater heaters does not increase proportionally, NRC staff expects the feedwater enthalpy to be reduced. NRC staff requests an explanation of the physical behavior of the feedwater enthalpy during the IOSGADV event.

Response

1. Figure 1 corresponding to Figure 15.1.4-1.9 shows a brief null transient before the initiation of the event. Also Figure 2 corresponding to Figure 15.1.4-2.9 shows a brief null transient before the initiation of the event. However, it is not practical to show a brief null transient before the initiation of the event, because all events are assumed to begin at 0 seconds. In addition, sections describing the sequence of events and system operation, and results provide sufficient information on the magnitude of the perturbation to the system resulting from the opening of the atmospheric dump valve.

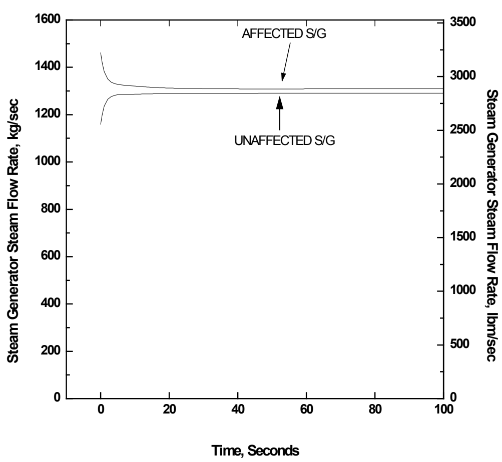

Figure 1. SG steam flow rate vs. Time (IOSGADV+LOOP)

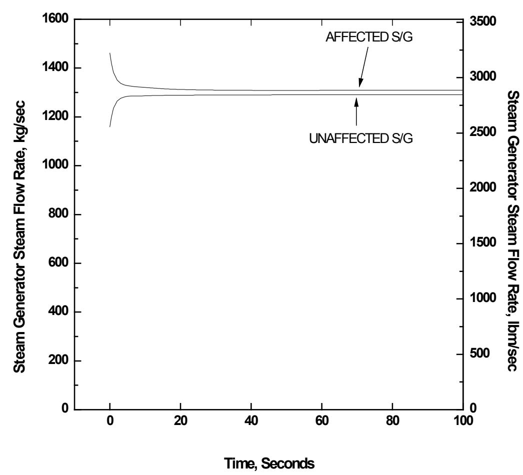

Figure 2. SG steam flow rate vs. Time (IOSGADV+LOOP+SF)

2. DCD Table 15.1.4-2 would be revised to show auxiliary feedwater actuation analysis setpoint. The significant reduction in RCS pressure at approximately 2600 seconds is caused by the collapse of the reactor vessel upper head void and the increased heat removal by the inadvertent of opening atmospheric dump valve. The significant increase

in RCS pressure at approximately 2800 seconds is caused by the the increase of safety injection flow rate with a decreasing RCS pressure.

3. For conservatism of the analysis, the higher feedwater enthaply is used until reactor trip, although the feedwater enthalpy would be reduced during IOSGADV event.

Impact on DCD

DCD Table 15.1.4-2 will be revised as indicated on the attached markup.

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.

Table 15.1.4-2

Sequence of Events of Full-Power Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with a Single Failure (IOSGADV+SF) and with a Loss of Offsite Power

Time (sec)	Event	Setpoint or Value
0.0	One atmospheric dump valve opens fully	-
1,800	Operator initiates manual trip	-
1,800.10	Reactor trip breakers open/turbine trip/ loss of offsite power/RCPs begin to coast down	-
1,801.70	Minimum transient DNBR	1.336
1,887.90	Pressurizer pressure reaches safety injection actuation signal analysis setpoint, kg/cm^{2}A (psia)	121.98 (1,735)
1,888.70	Void begins to form in RV upper head	-
1,927.90	Safety injection flow begins	-
1,955.65	Steam generator pressure reaches main steam isolation signal analysis setpoint, kg/cm^{2}A (psia)	57.09 (812)
1,962.0	MSIVs close completely	-
1,967.0	MFIVs close completely	-
2,722.40	Steam generator water level reaches auxiliary feedwater actuation analysis setpoint, %WR	19.9
3,000	Operator manually closes ADV	-
3,600	Operator initiates plant cooldown	-