SUPPLEMENTAL 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.:	**404-8488**
SRP Section:	**15.06.05 – Loss of Coolant Accidents Resulting From Spectrum of Postulated Pining Breaks Within the Reactor Coolant Pressure Boundary**
Application Section:	**15.6.5.2.2**
Date of RAI Issue:	**02/10/2016**

Question No. 15.06.05-12

Title 10 of the Code of Federal Regulations, Part 50.55a(h), “Protection and Safety Systems,” requires compliance with IEEE Std. 603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generationg Stations,” and the correction sheet dated January 30, 1995. Clause 6.8.1 of IEEE Std. 603-1991 requires that allowances for uncertainties between the anlytical limit and device setpoint be determined using a documented methodology such as ISA S67.040-1987(updated as ISA S67.04-1994). Regulatory Guide 1.105, Rev. 3 describes a method acceptable to the NRC staff for complying with the NRC’s requlations for ensuring that setpoints for safety-related instrumentation are initally within and remain within the technical specification limits, and endorses ISA-S67.04-1994 Part I. DCD Tier 2 Sections 7.1, 7.2 and 7.3 cite compliance with Regulatory Guide 1.105, Rev.3. Technical Report APR1400-Z-J-NR-14004-P, Rev.0, “Uncertainty Methodology and Application for Instrumentation” cites ISA-RP67.04-1994 PartII, which is not endoresed or approved by Regulatory Guide 1.105 Rev.3, as the standard that “provides the systematic method to identify the difinition, classification, sources, and calculation method of uncertainties.” Part II of ISA-RP67.04-1994 provides recommended practies and guidance for implementing Part of ISA-S67.04-1994.

Response

Technical Report APR1400-Z-J-NR-14004-P, Rev. 0, “Uncertainty Methodology and Application for Instrumentation” provides the method to identify the definition, classification, sources, and calculation method of instrument uncertainties in accordance with the requirements of Part I of ISA-S67.04-1994 and Part II of ISA-RP67.04-1994 is only used for guidance to implement the methods of instrument uncertainties

But, Part I of ISA-S67.04-1994 is not referenced in the Technical Report APR1400-Z-J-NR-14004-P, Rev. 0 and the report will be revised to incorporate Part I of ISA-S67.04-1994 as a reference document.

Supplemental Response

The terminology "analytical limit" stated in DCD Tier 2 Section 15.0.0.3 is identical to "analysis setpoint" described in Table 15.0-2. Therefore, in order to clarify the relationship between "analytical limit" and "analysis setpoint", Section 15.0.0.3 will be revised.

Impact on DCD

DCD Tier 2 Section 15.0.0.3 and Table 15.0-2 will be revised as indicated in the attachment.

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 Environmental Report.

15.0.0.2.5 Residual Decay Heat

Total Residual Decay Heat

The ANS 5.1-1979 standard decay heat model is used to calculate the decay heat generation in large-break loss-of-coolant accident (LBLOCA) analyses, and the ANS 5.1-1971 decay heat model is applied to small-break loss-of-coolant accident (SBLOCA) and the post-loss-of-coolant accident (LOCA) long-term cooling analyses. The non-LOCA analyses use the ANS 5.1-1973 decay heat curve with uncertainties.

Distribution of Decay Heat Following a Loss-of-Coolant Accident

Neutron, gamma, and beta energy fission products are generated during normal operation. In a LOCA, there are no neutron-induced chain reactions because the reactor is tripped either by void formation or by CEA insertion. Only gamma and beta radiation is created during a LOCA. During a LOCA, some gamma radiation is released from the fuel rod into another fuel rod, the reactor coolant, or the core structure while the beta radiation remains stored in the fuel rods, resulting in a redistribution of the core heat after the LOCA.

15.0.0.3 Reactor Trip System and Engineered Safety Feature Systems Analytical Limit and Delay Times

During any event, various systems operate in response to the event. The sequence of events and systems operations include information about systems operations. The systems that may operate in an event are (1) electrical, instrumentation, and control systems that are designed to perform a safety function, which are systems that operate during an event to mitigate the consequences and (2) systems that are not required to perform a safety function. Refer to Sections 7.2 through 7.6 and Section 7.7, respectively.

The RPS is described in Section 7.2. Table 15.0-2 lists the RPS trips for which credit is taken in the analyses that are described in Chapter 15, ~~including the setpoint and response times associated with each trip.~~ The analyses take into consideration the response times of actuated devices after the value of the monitored parameter at the sensor has equaled or exceeded the trip setpoint. The relevant reactor trip functions and engineered safety feature (ESF) functions for each event are shown in Table 15.0-7, and the specified reactor

including the analytical limit (or analysis setpoint) and response times associated with each trip.

Table 15.0-2

Analytical Limit

Reactor Protection System Trips Used in the Safety Analysis

Event	RPS	~~Analysis Setpoint~~ (1)	Sensor Response Time	Reactor Trip Delay Time (2)
Events not Mentioned Below	High Logarithmic Power Level	0.05 %	0 ms	550 ms
	Variable Overpower	116.5 %	0 ms	550ms
	CPC Variable Overpower	115 %	0 ms	650 ms
	High Pressurizer Pressure	169.7 kg/cm^2A (2,414 psia)	300 ms	550 ms
	Low Pressurizer Pressure	122.0 kg/cm^2A (1,735 psia)	600 ms	550 ms
	Low SG Pressure	57.1 kg/cm^2A (812 psia)	600 ms	550 ms
	Low SG Water Level	40.7 % wide range (3)	650 ms	600 ms
	High SG Water Level	95 % narrow range (4)	600 ms	550 ms
	Low Reactor Coolant Flow	80 % (5)	0 ms	1200 ms (7)
	CPC Low RCP Shaft Speed	94.83 %	0 ms	450 ms
	CPC Coincident Low Pressure/DNBR	140.6 kg/cm^2A (2,000 psia) /1.45 (6)	300 ms	650 ms
Feedwater and Steam Line Breaks	High Pressurizer Pressure	173.17 kg/cm^2A (2,463 psia)	300 ms	550 ms
	Low Pressurizer Pressure	109.3 kg/cm^2A (1,555 psia)	600 ms	550 ms
	Low SG Pressure	52.7 kg/cm^2A (750 psia)	600 ms	550 ms
	Low SG Water Level	28.4 % wide range (3)	650 ms	600 ms
	High SG Water Level	95 % narrow range (4)	600 ms	550 ms
	Low Reactor Coolant Flow	60 % (5)	0 ms	850 ms (7)
	CPC Low RCP Shaft Speed	94.83 %	0 ms	450 ms
	CPC Variable Overpower	121 % (8)	0 ms	650 ms
	High Containment Pressure	0.28 kg/cm^2G (4 psig)	600 ms	550 ms

(1) Some Chapter 15 analyses assumed more conservative setpoints for specific events.

(2) Reactor protection system response time testing is described in Section 7.2.

(3) Percent of distance between the wide-range instrument taps; the setpoint is valid at full power only (i.e., 100 − 102 % power).

(4) Percent of distance between the narrow-range instrument taps

(5) Percent of hot leg flow

(6) Trip credited for 15.6.3 events

(7) The total response time is the sum of sensor response time and reactor trip delay time. For a shaft break event, a reactor trip is required 1.2 seconds after the flow in the hot leg reaches its analysis setpoint. For a steam line break (SLB) with a LOOP up to 30 minutes into the event, a reactor trip is required 0.85 second after the core flow reaches its analysis setpoint.

(8) For SLB outside the containment, an additional 6 percent is considered conservative.