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.:	**73-8025**
SRP Section:	**03.09.01 - Design Transients**
Application Section:	**3.9.1**
Date of RAI Issue:	**07/15/2015**

Question No. 03.09.01-1

DCD Tier 2, Table 3.9-1, presents the APR1400 design basis initiating events and frequencies used in the stress analysis of ASME Boiler and Pressure Vessel Code (BPV Code) Class 1 and Class CS components of the primary system. The staff noticed that the number of event occurrences listed in this table is either higher or lower than the number of event occurrences for similar events listed in the NRC staff's final safety evaluation report for the System 80+ design certification (NUREG-1462, Section 3.9.1). Standard Review Plan (SRP) 3.9.1, Section III.1, states that the list of transients, the number of events estimated for each transient presented in the applicant's SAR, and the method for determining this number are compared to the same information on similar and previously licensed applications and to the acceptance criteria outlined in Subsection II of this SRP section. Any deviations from previous accepted practice are to be noted and the applicant should justify them. Given that APR 1400 DCD, Table 3.9-1 has significantly different values for number of design transient occurrences compared to the design transients of a similar certified design application, in accordance with the SRP 3.9.1, Section III.1, the applicant is requested to provide the bases for these variations (higher/lower occurrences), as compared to previous licensed or certified applications.

Response

Event frequencies of the APR1400 are based on those of the OPR1000, which has adopted the System 80 design and was a reference design for the APR1400 development. Event frequencies of the OPR1000 are similar to those of the System 80+. The design life of the OPR1000 is 40 years. Event frequencies of the APR1400 are determined using the following general guidelines.

- Event frequencies for the 40 year design life of SSCs are kept the same as the OPR1000 plant.
- Event frequencies for 60 years are increased by more than one half (1/2) an occurrence per year .

- Event frequencies for events equal to or less than one half (1/2) occurrence per year for 40 years are kept at that frequency for the 60 year period, except for the test events. Such events are very rare or hypothetical. These frequencies are conservative for the 60 year period.
- Event frequencies for heatup/cooldown were decreased after performing environmental fatigue analysis for ASME class 1 components for the APR1400. Heatup/cooldown events have large effects on the thermal fatigue and some reasonable re-evaluation is required. The increase in fuel cycle length from 12 months to 18 months is one of the factors for re-evaluation.
- Event frequencies for low power operation below 15% power were reduced compared to those of the OPR1000 considering the relatively short period of plant operation at low power compared to higher power operation.

The attachment shows the design basis events and frequencies of APR1400 and System 80+ used for the stress analyses. As shown in this attachment, most APR1400 design transient occurrences are greater than or equal to the given System 80+ events. This is considered conservative in terms of designing plant equipment. The basis for determining the frequencies of design basis initiating events that have lower frequencies than those of the System 80+ is provided in the following paragraphs. These events are bolded in the APR1400 design basis frequency column in attachment. The System 80+ events which are not established as design basis events for the APR1400 design are noted by bold lettering in the specific events column in this same attachment.

1. Low power operation

The number of occurrences for low power operation is determined considering the relatively short period of plant startup and operation at low power below 15% power compared to operation at higher power levels. The total number of occurrences of low power operation is expected to be 1,560 times over the design life (52 weeks/year × 1/2 × 60 years = 1,560 occurrences, in each direction (i.e., power ascension and power descension)). The value is rounded up to 1,600. The number of occurrences of the corresponding System 80+ events is 2,000 times over the design life. The APR1400 events considered in this determination are as follows:

- Turbine power steps of +1 percent (5-15% power)
- Turbine power steps of -1 percent (5-15% power)
- Turbine power ramps of +1 %/min (5-15% power)
- Turbine power ramps of -1 %/min (5-15% power)
- NSSS operations with the control systems in the manual mode of the Control Element Assembly (CEA), turbine bypass valves, pressurizer spray/heaters, pressurizer level control and feedwater flow (0-5% power)

2. Plant heatup and cooldown operation

The number of occurrences of a plant heatup/cooldown related operation is determined considering the interval of refueling and unplanned plant maintenance outages. The total number of occurrences is 250 over the design life. The heatup and cooldown related events for the APR1400 are expected to occur at most twice per year for scheduled refueling outages and unplanned plant maintenance and 30 times during the initial startup test period. The total assumed plant heatup and cooldown frequency for refueling operations is conservatively considered to be 150 cycles over 60 years. However, the design frequency is increased to 250 for conservatism in order to envelop transients which result in reactor trips and the need to be cooled down to cold shutdown conditions and from other Technical Specification driven or plant upset events. The number of corresponding occurrences for System 80+ events is 300 times over the design life. The APR1400 events considered in this determination are as follows:

- Manual operation of the auxiliary spray system
- Startup and shutdown of the shutdown cooling system at Hot Shutodown (HSD)
- Plant heatup
- Plant cooldown

3. Startup and coastdown of a reactor coolant pump at Hot Standby (HSB)

The number of occurrences of a startup and coastdown of a reactor coolant pump at HSB event is determined based on the number of plant heatup/cooldown operations. Each pump is assumed to be started or stopped independently of the other pumps. Therefore, the frequency is determined based upon the number of pumps times the frequency of plant heatup/cooldown events (4 pumps × 250 cycles = 1,000 cycles). This value is increased to 2,000 cycles considering unexpected events that result in startup and coastdown of reactor coolant pumps. The number of corresponding occurrences for System 80+ events is 4,000 times over the design lifetime.

4. Natural circulation cooldown (HSB to HSD)

The number of occurrences of a natural circulation cooldown (HSB to HSD) event is determined based on U.S. nuclear power plant operating experience data (NUREG/CR-5750 and NUREG/CR-6928), which shows that the expected number of occurrences of natural circulation cooldown operation is less than two times over a 60-year design life. The number of occurrences of this event for the APR1400 design is conservatively set to be ten times over the plant lifetime. The corresponding frequency for System 80+ events is 30 times over the design lifetime.

5. Uncontrolled CEA withdrawal at low and high power

The number of occurrences of an uncontrolled CEA withdrawal event at low and high power is determined based on U.S. nuclear power plant operating experience data (NUREG/CR-5750 and NUREG/CR-6928). According to this data, the expected number of occurrences of an uncontrolled CEA withdrawal event is expected to be approximately one time over the plant lifetime. The number of occurrences for this event for the APR1400 design is conservatively set to be five. The corresponding number of occurrences for System 80+ events is 30 times over the design lifetime.

6. CVCS malfunction that increases RCS inventory

The number of occurrences of a CVCS malfunction event that increases RCS inventory is determined based on U.S. nuclear power plant operating experience data (NUREG/CR-5750 and NUREG/CR-6928). According to this data, the expected number of occurrence of a CVCS malfunction event that increases RCS inventory is expected to be less than three times over lifetime. The number of occurrences of this event for the APR1400 design is conservatively set to be ten. The corresponding number of occurrences of System 80+ events is 20 times over the design lifetime.

7. SIS/SCS check valve operability test

The number of occurrences of a SIS/SCS check valve operability test event is determined based on the APR1400 SIS/SCS check valve test plan. According to the test plan, the test is performed using safety injection pumps when the shutdown cooling system is in operation during refueling operation. The APR1400 SIS/SCS check valve operability test is expected to be performed at most once per year considering normally scheduled refueling outage frequency. However, the design frequency is conservatively determined to be 120. The corresponding number of occurrences of System 80+ events is 500 times over the design lifetime.

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 Environmental Reports.

Table 1 Comparison of design basis events and frequencies between APR1400 and System 80+

Event Conditions	Event Description	Specific Events	Occurrences for Design Purpose		Occurrences of System 80+ transients
			60 Years*	40 Years*	60 Years
Level A Service Conditions					
Normal Event-1A	Steady state operation with normal parameter variations in the increasing direction (5-100%)		1,500,000	1,000,000	1,000,000
Normal Event-1B	Steady state operation with normal parameter variations in the decreasing direction (5-100%)		1,500,000	1,000,000	1,000,000
		Frequency control	N/A		1,000,000
Normal Event-2A	Daily Load Follow Operation (From 100 to 50% power)		22,000	15,000	22,000
Normal Event-2B	Daily Load Follow Operation (From 50 to 100% power)		22,000	15,000	22,000
Normal Event-3A	Turbine Step Load Change in the Increasing Direction	Turbine power steps of +10 percent (15-100 % power)	3,200	2,000	2,000
		Turbine power steps of +1 percent (5-15 % power)	**1,600**	1,000	2,000
Normal Event-3B	Turbine Step Load Change in the Decreasing Direction	Turbine power steps of -10 percent (15-100 % power)	3,200	2,000	2,000
		Turbine power steps of -1 percent (5-15 % power)	**1,600**	1,000	2,000
Normal Event-3C	Large Turbine Load Step Decrease	Turbine load rejection up to 50 % (50-100 % power)	60	40	40
		Turbine generator runback to house load	60	40	40
		Reactor trip	150	100	60
		Turbine Trip	150	100	N/A
Normal Event-4A	Turbine Ramp Load Change in the Increasing Direction	Turbine power ramps of +5 %/min (15-100 % power)	3,200	2,000	N/A
		Turbine power ramps of +1 %/min (5-15 % power)	**1,600**	1,000	2,000
Normal Event-4B	Turbine Ramp Load Change in the Decreasing Direction	Turbine power ramps of -5 %/min (15-100 % power)	3,200	2,000	N/A
		Turbine power ramps of -1 %/min (5-15 % power)	**1,600**	1,000	2,000
		Loss of a main feedwater pump without reactor trip	60	40	40
		Tie line thermal backup	N/A		60

Normal Event-5	Non-Load Change Events (Planned)	NSSS operations with the control systems in the manual mode of the CEA, turbine bypass valves, pressurizer spray/heaters, pressurizer level control and feedwater flow (0-5 % power)	**1,600**	1,000	2,000
		Opening or closure of the feedwater economizer valve	500	500	400
		NSSS operations with the control systems in the manual mode (5-100 % power)	3,200	2,000	2,000
		Manual operation of the auxiliary spray system	**250**	170	300
		High steam generator blowdown	3,200	2,000	2,000
		Shift from normal to maximum CVCS flow rate and return	3,200	2,000	2,000
Normal Event-6	Non-Load Change Events (Unplanned)	Low-low VCT level and charging pump diversion to the boric acid storage tank	60	40	N/A
		Spurious actuation of the pressurizer spray	60	40	40
		Spurious actuation of the pressurizer heaters	60	40	40
		Inadvertent closure of one economizer or downcomer feedwater valve	60	40	40
		Inadvertent opening of one economizer or downcomer feedwater valve	60	40	N/A
		Inadvertent isolation of one main feedwater heater	60	40	40
Normal Event-7	Plant Events Below Power Operation	Startup and coastdown of a Reactor Coolant Pump at HSB	**2,000**	1,340	4,000
		Startup and shutdown of the shutdown cooling system at HSD	**250**	170	300
		Spurious startup of a safety injection pump during shutdown conditions	60	40	40
		Spurious actuation of the pressurizer heaters at HSB	60	40	N/A
Normal Event-8	Plant heatup		**250**	170	300
Normal Event-9	Plant cooldown		**250**	170	300
Level B Service Conditions					
Upset Event-1	Increase Heat Removal by the	Decrease in feedwater temperature	20	20	20

	Secondary System	Increase in feedwater flow rate	20	20	20
		Increase in steam flow rate	20	20	20
		Inadvertent opening of a main steam safety valve	10	10	10
Upset Event-2	Decrease Heat Removal by the Secondary System	Loss of external load	20	20	19
		Loss of condenser vacuum	20	20	20
		Loss of non-emergency AC power to the station auxiliaries	20	20	10
		Main steam isolation valve closure	20	20	5
		Loss of normal feedwater flow	20	20	20
		Turbine trip	N/A		20
Upset Event-3	Decrease in Reactor Coolant System Flow Rate	Loss of forced reactor coolant flow	20	20	20
		Natural circulation cooldown (HSB to HSD)	**10**	10	30
Upset Event-4	Reactivity and Power Distribution Anomalies	Uncontrolled CEA withdrawal at low power	**5**	5	10
		Uncontrolled CEA withdrawal at high power	**5**	5	10
		Control rod misoperation, RPCS inadvertent operation or operator error	50	35	20
Upset Event-5	Increase in Reactor Coolant System Inventory	Loss of component cooling water to the letdown heat exchanger	10	10	10
		CVCS mal-function that increases RCS inventory	**10**	10	20
Upset Event-6	Decrease in Reactor Coolant System Inventory	Inadvertent opening of a pilot operated safety & relief valve (POSRV closed as expected)	10	10	N/A
		Failure of small lines carrying coolant outside containment (letdown line break)	20	20	5
Upset Event-7	Reactor coolant pump seal failure		10	10	10
Upset Event-8	Loss of seal injection with loss of cooling water		5	5	5
Level C Service Conditions					
none					

Level D Service Conditions					
Faulted Event-1	Increase Heat Removal by the Secondary System	Steam system piping failure	1	1	1
Faulted Event-2	Decrease Heat Removal by the Secondary System	Feedwater system pipe break (FWLB)	1	1	1
Faulted Event-3	Decrease in Reactor Coolant System Flow Rate	Reactor coolant pump rotor seizure	1	1	1
		Reactor coolant pump shaft break	1	1	1
Faulted Event-4	Reactivity and Power Distribution Anomalies	Rod ejection accident	1	1	1
Faulted Event-5	Decrease in Reactor Coolant System Inventory	Inadvertent opening of a pilot operated safety relief valve (POSRV fails to close)	1	1	1
		Steam generator tube rupture (SGTR)	1	1	1
		Loss of coolant accidents resulting from postulated pipe breaks within the RCS pressure boundary (LOCA)	1	1	1
		Inadvertent opening of a SDS valve	N/A		1
Faulted Event-6		Total loss of feedwater flow	1	1	N/A
Test Conditions					
RCS hydrostatic test			15	10	10
Secondary hydrostatic test			15	10	10
RCS leak test			200	200	200
Secondary leak test			200	200	200
SIS/SCS preoperational and maintenance test			360	240	240
SIS/SCS check valve operability test			**120**	80	500

Note:
(1) * : The design life for RCS main components and Class 1 piping is 60-year, and the design life for Class 2 & 3 piping and other components except RCS main components is 40-year.
(2) N/A : not applicable

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Question No. 03.09.01-2

In DCD Tier 2, Table 3.9-1 note (1), the applicant states that the design life for RCS main components and Class 1 piping is 60 years, and the design life for Class 2 and 3 piping and other components except RCS main components is 40 years. SRP Section 3.9.1, Section III.1, states that the list of transients, the number of events estimated for each transient presented in the applicant's SAR, and the method for determining this number are compared to the same information on similar and previously licensed applications and to the acceptance criteria outlined in Subsection II of this SRP section. Any deviations from previous accepted practice are to be noted and the applicant should justify them. To support the staff's finding associated with SRP Section 3.9.1, Section III.1, the applicant is requested to provide the basis for designing Class 1 piping and components for 60 years and Class 2/3 piping and components for 40 years, as well as to provide the basis for the number of transient cycles in DCD Tier 2, Table 3.9-1 when the 40-year and 60-year columns are identical.

Response

(1) The main components and supports of the RCS primary system, Class 1, have a 60-year design life. The exchangeable components and supports of the RCS secondary systems, Class 2 and 3, have a 40-year design life. A safety evaluation to assess acceptable continued operation of the Class 2 and 3 components is performed after 40 years of plant operation. In addition, their design life can be extended through repair or exchange if required.

(2) As provided in the response to Question 03.09.01-1, event frequencies for the events having equal to or less than half (1/2) an occurrence per year for the 40 year design life are assumed for the full 60 year design life. Such events are very rare or hypothetical, so the frequencies applied are considered conservative for the 60 year life. Other transients reviewed include opening or closing of the economizer feedwater control valve and RCS and secondary leak tests.

Opening of the economizer feedwater control valve is postulated to occur at 20% power during power ascension operation and can result in thermal fatigue in the steam generator economizer feedwater nozzles. For the APR1400, this event is set at 500 occurrences over 60 years and is considered sufficiently conservative because such operation is normally performed only during a reactor startup. The event frequency of 500 occurrences over 40 years is taken to be the same as that of the currently designed OPR1000 40-year design life.

The number of occurrence of RCS and secondary leak tests is determined based on the APR1400 leak test program. The test is performed after every refueling outage on an 18-month fuel cycle and is expected to occur 40 times over the 60-year plant lifetime. The number of occurrences for a 40-year design is based on the frequency of the OPR1000 design having a 12-month fuel cycle and results in 40 times over the plant lifetime. The assumed frequency of 200 times is considered conservative for both the 60-year and 40-year lifetime designs.

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 Environmental Reports.

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Question No. 03.09.01-3

DCD Tier 2, Table 3.9-1 presents the APR1400 design basis initiating events and frequencies used in the stress analysis of ASME Boiler and Pressure Vessel Code (BPV Code) Class 1 and Class CS components of the primary system. The staff noticed that the design transient of steam generator tube rupture (SGTR) is included in Level D Service Conditions. In ANSI/ANS-51.1, SGTR is classified as a plant condition 3 event, which appears to be equivalent to the severity of Level B or C events in the APR1400 design. GDC 15 requires that the reactor coolant system and associated auxiliary, control and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. The applicant is requested to provide the justification for why the SGTR is included as a Level D event in the list of transients in DCD Tier 2, Section 3.9.1 versus a Level B or C event.

Response

Classification of the SGTR event has been a topic of discussion in the nuclear industry since SGTR events have occurred more frequently compared to other postulated accidents. SGTR is classified to Level D event in the System 80+ design. Classification of the APR1400 SGTR event is also based on the OPR1000 design practice in which a SGTR is also classified as a Level D event.

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 Environmental Reports.

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Question No. 03.09.01-4

During an audit related to DCD Tier 2, Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components," the staff observed that one of the audited reports indicated that the computer code DPVIB was used in the design of the nuclear steam supply system. However, DCD Tier 2, Section 3.9.1 has no description of this computer code. SRP Section 3.9.1 indicates that the DCD should include a list of computer programs to be used in dynamic and static analyses to determine the structural and functional integrity of seismic Category I Code and non-Code items and the analyses to determine stresses. To support the staff's finding associated with this SRP section, the applicant is requested to provide the staff with a DCD markup to include the description of computer code DPVIB in DCD Tier 2, Section 3.9.1.

Response

The DPVIB computer program is a short and simple program to obtain pressure fluctuations from the reactor coolant pumps on the core support barrel. The program is not used to analyze stresses in determining the structural or functional integrity of the reactor vessel internals (RVI). As such, the program does not contain physically meaningful information to be listed in the DCD. KHNP also noted that other operating nuclear units that are prototypes for the APR1400 RVIs have not included a description of the DPVIB computer code.

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 Environmental Reports.