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.:	**78-8021**
SRP Section:	**14.03 Inspections, Tests, Analyses, and Acceptance Criteria**
Application Section:	**14.03.03**
Date of RAI Issue:	**07/15/2015**

Question No. 14.03.03-1

Section 1.3 of DCD Tier 1 states that “The figure legend and acronym and abbreviation list are provided for information only and are not part of the Tier 1 DCD” A similar statement appears in multiple tables, such as Table 2.4.1-3 (page 2.4-9) where it states “The column ‘Item No.’ is information only (not part of certified design).” This language could create confusion about the process for changing Tier 1 information described in 10 CFR 52.63. In particular, the idea of “information only” material appears to differ from the definitions of Tier 1 information that have appeared in previous design certification rule Appendices to 10 CFR Part 52. Therefore, it should be revised for consistency with these processes and in a manner that retains a unique (though possibly generically phrased) identifier that enables traceability of component information throughout the DCD. A related example can be found in the AP1000 DCD Tier 1 on page 1.2-2, where the “Interpretation of Figures” section provides some discussion on as-built attributes and their relationship to safety functions that has been accepted by the NRC staff; however, this does not extend to item numbers.

Response

A definition of “item number” will be inserted in 1.1 Definition subsection to describe the process similar to that of “Interpretation of Figures” used in Tier 1 Section 1.2.4. All the statements in the multiple tables written “Item number is information only” will be deleted to prevent confusion.

Impact on DCD

DCD Tier-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 Report

There is no impact on any Technical, Topical, or Environment Reports.

Description and to the extent such equipment is located in a harsh environment during or following a design basis accident.

Test means the actuation or operation, or establishment of specified conditions, to evaluate the performance or integrity of as-built structures, systems, or components, unless explicitly stated otherwise.

Type Test means a test on one or more sample components of the same type and manufacturer to qualify other components of that same type and manufacturer. A type test is not necessarily a test of the as-built structures, systems or components.

Item number
Item numbers used in Tier 1 represent the complete item number or a portion of the item number used to identify the actual hardware(or associated software). The Tier 1 item number format conforms to the full equipment number from which the plant designator and safety designator are omitted. Tier 1 item numbers are not part of the Tier 1 material.

Table 2.4.1-3 (2 of 2)

Instrument Name	Item No.~~(1)~~	ASME Section III Class	Seismic Category	Class 1E /Harsh Envir. Qual.	Display / Alarm at MCR	Display / Alarm at RSR
Refueling Water Level (NR)	L-105, 106	-	II	No/No	Yes/Yes	-/-
Refueling Water Level (WR)	L-115, 116	-	II	No/No	Yes/No	-/-
SG Pressure	P-1013A, B, C, D P-1023A, B, C, D	-	I	Yes/Yes	Yes/Yes	Yes/Yes
SG Level (NR)	L-1114A, B, C, D L-1124A, B, C, D	-	I	Yes/Yes	Yes/Yes	-/-
SG Level (WR)	L-1113A, B, C, D L-1123A, B, C, D	-	I	Yes/Yes	Yes/Yes	Yes Yes

~~(1) The column "Item No." is information only (not part of certified design).~~

(2) Dash(-) indicates not applicable.

(3) No alarm.

(4) Trend is displayed instead of indication.

All the statements written below in multiple page will be deleted in whole Tier-1 DCD

\- The column "Item No." is information only (not part of certified design)

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.: **78-8021**

SRP Section: **14.03.03 - Piping Systems and Components – Inspection, Tests, Analyses, and Acceptance Criteria**

Application Section: **14.03.03**

Date of RAI Issue: **07/15/2015**

Question No. 14.03.03-2

DCD Tier 1, Table 2.3-3 is inconsistent with the remainder of Tier 1. Several systems, as named in the table, do not appear in later sections with the appropriate ITAAC. The system naming and ITAAC references should be made consistent across Tier 1. Examples include:

1. Safety Depressurization and Vent System – not found
2. Containment Air Monitoring/Sampling System – not found
3. Auxiliary Feedwater Pump Turbine System – not found
4. Station Heating System – not found
5. Breathing Air System – not found
6. Instrument Air System – not found
7. Service Air System – not found
8. Auxiliary Steam System - This section is stated as “no entry” (Tier 1, Section 2.7.1.9).
9. Safety Injection/Shutdown Cooling System - These two systems should be separate entries.
10. Essential/Normal Chilled Water System - These two systems should be separate entries.

Response

DCD Tier 1 Table 2.3-3 and the corresponding DCD Tier 1 and Tier 2 sections will be revised as indicated below to be consistent.

1. Safety Depressurization and Vent System – Renamed in Table 2.3-3 to Reactor Coolant Gas Vent System.
2. Containment Air Monitoring/Sampling System – Renamed in Table 2.3-3 to Process and Effluent Radiation Monitoring and Sampling System.
3. Auxiliary Feedwater Pump Turbine System – Eliminated from Table 2.3-3 with clarifying note added to the Auxiliary Feedwater System.
4. Station Heating System – Eliminated from Table 2.3-3.
5. Breathing Air System – Eliminated from Table 2.3-3.
6. Instrument Air System – Eliminated from Table 2.3-3.
7. Service Air System – Eliminated from Table 2.3-3.
8. Auxiliary Steam System – This system is not required to perform a safety function. Therefore, the design description does not need to be included in the DCD Tier 1 document and Section 2.7.1.9 will reflect “no entry.”
9. Safety Injection/Shutdown Cooling System – Table 2.3-3 revised to reflect two separate systems.
10. Essential/Normal Chilled Water System – Table 2.3-3 revised to reflect two separate systems.
11. Compressed Air System – Renamed in Table 2.3-3 instead of breathing air system, instrument air system, and service air system.

Impact on DCD

DCD Tier 1 Table 2.3-3, Sections 2.4, 2.6, 2.7 and DCD Tier 2 Tables 3.4-1, 3.6-1, Sections 1, 3.8 and 6.8 will be revised as shown 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 Reports.

Table 2.3-3

High and Moderate Energy Piping Systems

System(1)(2)	High-Energy	Moderate-Energy
Reactor Coolant System	O	
~~Safety Depressurization and Vent System~~(3) Reactor Coolant Gas Vent System	O	
Safety Injection/~~Shutdown Cooling~~ System	O	
Shutdown Cooling System	O	
Chemical and Volume Control System	O	
Steam Generator Blowdown System(4)	O	
Component Cooling System		O
Spent Fuel Pool Cooling and Cleanup System		O
~~Containment Air Monitoring/Sampling System~~ Process and Effluent Radiation Monitoring and Sampling System		O
Containment Spray System		O
Essential Service Water System		O
Main Steam System	O	
~~Auxiliary Feedwater Pump Turbine System~~	~~O~~	
Main Feedwater System	O	
Auxiliary Feedwater System (6)	O	~~O~~
Auxiliary Steam System	O	
Emergency Diesel Generator System(5)	O	
Diesel Fuel Oil Transfer System		O
Fire Protection System		O
~~Station Heating System~~		~~O~~
Drain System		O
Essential/~~Normal~~ Chilled Water System		O
Plant Chilled Water System		O
~~Breathing Air System~~ Compressed Air System		O
~~Instrument Air System~~		~~O~~
~~Service Air System~~		~~O~~

(1) Systems classified as high-energy are either totally or partially high-energy. If portions of system are high-energy, it is classified as high-energy system.

(2) Systems or portions of systems outside the containment building and auxiliary building are excluded from this table.

~~(3) The piping portion from 2-in piping located at downstream of POSRV to the first isolation valve is high-energy piping and the rest is moderate-energy piping.~~

(4) Wet layup recirculation system is classified as moderate-energy system.

(5) Subsystems other than an EDG engine starting air system are classified as moderate-energy systems.

(6) Subsystems other than an auxiliary feedwater pump turbine subsystem are classified as moderate-energy systems.

2.4.2 In-containment Water Storage System

2.4.2.1 Design Description

POSRVs and the reactor coolant gas vent system (RCGVS).

The in-containment water storage system (IWSS) is a safety-related system and includes the in-containment refueling water storage tank (IRWST) which is an integral part of containment building structures, the holdup volume tank (HVT) which is also an integral part of the containment building structures, and the cavity flooding system (CFS).

The IRWST provides borated water for the safety injection system (SIS) and the containment spray system (CSS). It is the primary heat sink for discharges from the ~~safety depressurization and vent system.~~ It is the source of water for the CFS, and for filling the refueling pool via the shutdown cooling system (SCS).

The HVT collects water released in containment during design basis events (DBEs) and returns water to the IRWST through spillways. It receives water discharged from the IRWST and transfers the water to the reactor cavity area by the CFS.

The CFS is used to provide water to flood the reactor cavity in response to beyond DBEs.

The IWSS is located in the containment.

1. The functional arrangement of the IWSS is as described in the Design Description of Subsection 2.4.2.1 and in Table 2.4.2-1 and as shown in Figure 2.4.2-1.

2.a The ASME Code components identified in Table 2.4.2-2 are designed and constructed in accordance with ASME Section III requirements.

2.b The ASME Code piping including supports identified in Table 2.4.2-1 is designed and constructed in accordance with ASME Section III requirements.

3.a Pressure boundary welds in ASME Code components identified in Table 2.4.2-2 meet ASME Section III requirements.

3.b Pressure boundary welds in ASME Code piping identified in Table 2.4.2-1 meet ASME Section III requirements.

8.b All controls required by the design exist in the RSR to open and close SOVs identified in Table 2.4.5-2.

8.c All displays and alarms required by the design exist in the MCR as defined in Tables 2.4.5-2 and 2.4.5-3.

8.d All displays and alarms required by the design exist in the RSR as defined in Tables 2.4.5-2 and 2.4.5-3.

9.The high-energy piping systems, including the protective features are reconciled with pipe rupture hazards analyses report to ensure that the safety-related SSCs are protected against or are qualified to withstand the dynamic effects associated with postulate failures of these piping systems.

2.4.5.2 Inspections, Tests, Analyses, and Acceptance Criteria

The inspections, tests, analyses, and associated acceptance criteria of the RCGVS are specified in Table 2.4.5-4.

Table 2.4.5-4 (5 of 5)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8.a All controls required by the design exist in the MCR to open and close SOVs identified in Table 2.4.5-2	8.a Tests will be performed using the controls in the MCR.	8.a All controls in the as-built MCR open and close SOVs identified in Table 2.4.5-2.
8.b All controls required by the design exist in the RSR to open and close SOVs identified in Table 2.4.5-2.	8.b Tests will be performed using the controls in the RSR.	8.b All controls in the as-built RSR open and close SOVs identified in Table 2.4.5-2.
8.c All displays and alarms required by the design exist in the MCR as defined in Tables 2.4.5-2 and 2.4.5-3.	8.c Inspections will be performed on the displays and alarms in the MCR.	8.c All displays and alarms exist and are retrieved in the as-built MCR as defined in Tables 2.4.5-2 and 2.4.5-3.
8.d All displays and alarms required by the design exist in the RSR as defined in Tables 2.4.5-2 and 2.4.5-3.	8.d Inspections will be performed on the displays and alarms in the RSR.	8.d All displays and alarms exist and are retrieved in the as-built RSR as defined in Tables 2.4.5-2 and 2.4.5-3.

9. The high-energy piping systems, including the protective features are reconciled with pipe rupture hazards analyses report to ensure that the safety-related SSCs are protected against or are qualified to withstand the dynamic effects associated with postulate failures of these piping systems.

9. Inspections and analyses of the as-built high-energy piping including the protective features and safety-related SSCs will be performed.

9. Pipe rupture hazard analysis report exists and concludes that the as-built safety-related SSCs are protected against or are qualified to withstand the effects of postulated pipe failures of the as-built high-energy piping system.

20. When the Class 1E EDG is started by an ESF actuation signal, all Class 1E EDG protection systems, except for overspeed and generator differential current, are automatically bypassed.

high

21. The ~~moderate~~-energy piping systems are reconciled with pipe rupture hazards analyses report to ensure that the safety-related SSCs are protected against or are qualified to withstand the environmental effects associated with postulate failures of these piping systems.

2.6.2.2 Inspection, Test, Analyses, and Acceptance Criteria

Table 2.6.2-3 specifies the inspections, tests, analyses, and associated acceptance criteria for the EDG system.

Table 2.6.2-3 (7 of 7)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
18. When running ina test mode, an EDG is capable of responding to an automatic start signal.	18. Tests will be performed with each EDG in a test mode configuration. An automatic start signal will be simulated.	18. When running in a test mode, each EDG resets to its automatic control mode upon receipt of a simulated automatic start signal.
19. Each Class 1E EDG is designed and sized to supply power to its train's safety-related loads after a LOOP or a LOOP concurrent with LOCA conditions.	19.a Analyses will be performed to verify that each Class 1E EDG is capable of supplying power to its train safety-related loads after a LOOP or a LOOP concurrent with LOCA conditions.	19.a A report exists and concludes that each Class 1E EDG is designed and sized to supply power to its train's safety-related loads after a LOOP or a LOOP concurrent with LOCA conditions.
	19.b Inspections will be performed to verify that the rating of each as-built Class 1E EDG is in accordance with the size requirements of the analysis.	19.b The rating of each Class 1E EDG bounds the size requirements of the analysis.
20. When the Class 1E EDG is started by an ESF actuation signal, all Class 1E EDG protection systems, except for overspeed and generator differential current, are automatically bypassed.	20. Tests will be performed to verify the as-built Class 1E EDG protection systems.	20. A report exists and concludes that the as-built Class 1E EDG protection systems, except for overspeed and generator differential current, are automatically bypassed when the Class 1E EDG is started by an ESF actuation signal.
21. The ~~moderate~~ high-energy piping systems are reconciled with pipe rupture hazards analyses report to ensure that the safety-related SSCs are protected against or are qualified to withstand the environmental effects associated with postulate failures of these piping systems.	21. Inspections and analyses of the as-built ~~moderate~~ high-energy piping and safety-related SSCs will be performed.	21. Pipe rupture hazard analysis report ~~exits~~ exists and concludes that the as-built safety-related SSCs are protected against or are qualified to withstand the effects of postulated pipe failures of the as-built ~~moderate~~ high-energy piping system.

12.a Each AFW pump delivers the minimum flow to its respective steam generator for removal of core decay heat against a steam generator feedwater nozzle pressure.

12.b The cavitating flow-limiting venturis limit maximum flow to each steam generator with both AFW pumps running in the division against a steam generator pressure.

high

13. The ~~moderate~~-energy piping systems are reconciled with pipe rupture hazards analyses report to ensure that the safety-related SSCs are protected against or are qualified to withstand the environmental effects associated with postulate failures of these piping systems.

2.7.1.5.2 Inspection, Tests, Analyses, and Acceptance Criteria

Table 2.7.1.5-4 specifies the inspections, tests, analyses, and associated acceptance criteria for the AFWS.

Table 2.7.1.5-4 (7 of 7)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12.a Each AFW pump delivers the minimum flow to its respective steam generator for removal of core decay heat against a steam generator feedwater nozzle pressure.	12.a A test of each AFW pump will be performed to determine the system flow against steam generator pressure under preoperational condition. Analysis will be performed to convert the test results to the design conditions.	12.a A test report exists and concludes that each AFW pump delivers minimum flow of 2,461 L/min (650 gpm) to its respective steam generator against a steam generator feedwater nozzle pressure of 87.18 kg/cm^{2}A (1,240 psia).
12.b The cavitating flow-limiting venturis limit maximum flow to each steam generator with both AFW pumps running in the division against a steam generator pressure.	12.b A test will be performed with both pumps in a division running under preoperational condition. Analysis will be used to convert the test results to the design conditions.	12.b A test report exists and concludes that the maximum flow to each SG is less than or equal to 3,596 L/min (950 gpm) with both pumps running against a steam generator pressure of 0 kg/cm^{2}G (0 psig).
13. The high ~~moderate~~-energy piping systems are reconciled with pipe rupture hazards analyses report to ensure that the safety-related SSCs are protected against or are qualified to withstand the environmental effects associated with postulate failures of these piping systems.	13. Inspections and analyses of the as-built high ~~moderate~~-energy piping and safety-related SSCs will be performed.	13. Pipe rupture hazard analysis report exists ~~exits~~ and concludes that the as-built safety-related SSCs are protected against or are qualified to withstand the effects of postulated pipe failures of [illegible]uilt high ~~moderate~~-energy piping system.

2.7.2.4 Plant Chilled Water System

2. The moderate-energy piping systems are reconciled with pipe rupture hazards analyses report to ensure that the safety-related SSCs are protected against or are qualified to withstand the environmental effects associated with postulate failures of these piping systems.

2.7.2.4.1 Design Description

The plant chilled water system (PCWS) is a non safety-related closed loop chilled water system that serves non safety-related HVAC cooling loads, with the exception of containment isolation valves (CIVs) and the piping between the CIVs that are safety-related, ASME Section III Class 2 and seismic Category I as described in Subsection 2.11.3. The PCWS performs the containment isolation function for the PCW lines penetrating the containment.

The PCWS consists of central chilled water subsystem and compound building chilled water subsystem. The central chilled water subsystem is located in the auxiliary building and provides chilled water to the reactor containment fan coolers (RCFCs) and reactor cavity air handling unit (AHU) located in the reactor containment building. This subsystem also provides chilled water to AHUs and cubicle coolers located in auxiliary building and turbine generator building. The compound building chilled water subsystem is located in the compound building and provides chilled water to the AHUs, cubicle coolers and gaseous radwaste system package located in the compound building. The central chilled water subsystem consists of four chillers, two chilled water pumps, an air separator, a compression tank, a chemical additive tank, associated piping, controls and instrumentation. The compound building chilled water subsystem consists of three chillers, two chilled water pumps, an air separator, a compression tank, a chemical additive tank, associated piping, controls and instrumentation.

To meet the above functional requirements, the PCWS is designed as follows:

1. The functional arrangement of the PCWS is as described in the Design Description of Subsection 2.7.2.4.1.

2.7.2.4.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.2.4-1 specifies the inspections, tests, analyses, and associated acceptance criteria for plant chilled water system.

The ITAAC related to the CIVs and the piping between the CIVs of the PCWS are described in Table 2.11.3-2.

Table 2.7.2.4-1

Plant Chilled Water System ITAAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the PCWS is as described in the Design Description of Subsection 2.7.2.4.1.	1. Inspection of the as-built PCWS will be performed.	1. The as-built PCWS conforms with the functional arrangement as described in the Design Description of Subsection 2.7.2.4.1.

2. The moderate-energy piping systems are reconciled with pipe rupture hazards analyses report to ensure that the safety-related SSCs are protected against or are qualified to withstand the environmental effects associated with postulate failures of these piping systems.

2. Inspections and analyses of the as-built moderate-energy piping and safety-related SSCs will be performed.

2. Pipe rupture hazard analysis report exists and concludes that the as-built safety-related SSCs are protected against or are qualified to withstand the effects of postulated pipe failures of the as-built moderate-energy piping system.

5. The safety-related divisional cabinet (SRDC) of the PERMSS provides an automatic ESF initiation signals, as shown on Table 2.7.6.4-2.

6. The seismic category I monitors identified in Table 2.7.6.4-1 can withstand seismic design basis loads without loss of safety function.

7. Separation is provided between Class 1E divisions, and between Class 1E division and non-Class 1E division.

8. The moderate-energy piping systems are reconciled with pipe rupture hazards analyses report to ensure that the safety-related SSCs are protected against or are qualified to withstand the environmental effects associated with postulate failures of these piping systems.

2.7.6.4.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.6.4-3 specifies the inspections, tests, analyses, and associated acceptance criteria for the process and effluent radiation monitoring and sampling system.

Table 2.7.6.5-3 (2 of 2)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The seismic Category I monitors of the ARMS identified in Table 2.7.6.5-1 can withstand seismic design basis loads without loss of safety function.	5.a. Inspections will be performed to verify that the as-built seismic Category I monitor identified in Table 2.7.6.5-1 is located in a seismic Category I structure(s).	5.a. The as-built seismic Category I monitor identified in Table 2.7.6.5-1 is located in a seismic Category I structure(s).
	5.b. Type test, analyses, or a combination of type tests and analyses of seismic Category I monitor identified in Table 2.7.6.5-1 will be performed.	5.b. A report exists and concludes that the seismic Category I monitor identified in Table 2.7.6.5-1 withstands seismic design basis loads without loss of safety function.
	5.c. Inspections and analyses will be performed to verify that the as-built seismic Category I monitor identified in Table 2.7.6.5-1 including anchorages is seismically bounded by the tested or analyzed conditions.	5.c. A report exists and concludes that the seismic Category I monitor identified in Table 2.7.6.5-1 including anchorages is seismically bounded by the tested or analyzed conditions.
6. The safety-related divisional cabinet (SRDC) of the ARMS provides an automatic ESF initiation signals, as shown in Table 2.7.6.5-2.	6. A Testing of the as-built SRDC will be performed using an integral activated check source.	6. Each as-built ESF initiation signal is sent to ESF-CCS group control cabinet upon detection of high radiation of containment operating area and fuel handling area defined in Table 2.7.6.5-2, if plant's radiation monitors exceed predetermined setpoints for containment purge isolation actuation signal (CPIAS) and fuel handling area emergency ventilation actuation signal (FHEVAS).
7. The moderate-energy piping systems are reconciled with pipe rupture hazards analyses report to ensure that the safety-related SSCs are protected against or are qualified to withstand the environmental effects associated with postulate failures of these piping systems.	7. Inspections and analyses of the as-built moderate-energy piping and safety-related SSCs will be performed.	7. Pipe rupture hazard analysis report exists and concludes that the as-built safety-related SSCs are protected against or are qualified to withstand the effects of postulated pipe failures of the as-built moderate-energy piping system.

MSIS	main steam isolation signal
MSIV	main steam isolation valve
MSLB	main steam line break
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
PACU	package air conditioning unit
PAR	passive autocatalytic recombiner
POSRV	pilot operated safety relief valve
PSI	preservice inspection
PSR	pneumatic actuated spring return
RCFC	reactor containment fan cooler
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RDT	reactor drain tank
RG	Regulatory Guide
RMI	reflective metal insulation
RSR	remote shutdown room
SCP	shutdown cooling pump
SCS	shutdown cooling system
~~SDS~~	~~safety depressurization system~~
SER	safety evaluation report
SG	steam generator
SGTR	steam generator tube rupture
SI	safety injection
SIAS	safety injection actuation signal

DCD Chapter 6 ACRONYM AND ABBREVIATION LIST

PWR	pressurized water reactor
PZR	pressurizer
RAP	reliability assurance program
RC	reactor coolant
RCB	reactor containment building
RCGVS	reactor coolant gas vent system
RCL	reactor coolant loop
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RDT	reactor drain tank
RHR	residual heat removal
RPS	reactor protection system
RRS	required response spectra
RSPT	reed switch position transmitter
RSR	remote shutdown room
RTD	resistance temperature detector
RT_{NDT}	reference temperature for nil-ductility transition
RTSS	reactor trip switchgear system
RV	reactor vessel
RVI	reactor vessel internals
SAFDL	specified acceptable fuel design limit
SAM	seismic anchor movement motion
SC	1) shutdown cooling 2) safety console
SCP	shutdown cooling pump
SCS	shutdown cooling system
SDCHX	shutdown cooling heat exchanger
~~SDS~~	~~safety depressurization system~~
~~SDVS~~	~~safety depressurization and vent system~~
SECY	office of the secretary of the commission

DCD Chapter 6 ACRONYM AND ABBREVIATION LIST

SDN	safety system data network
~~SDS~~	~~safety depressurization system~~
~~SDVS~~	~~safety depressurization and vent system~~
SECY	Secretary of the Commission, Office of the NRC
SER	safety evaluation report
SF	1) stratified flow 2) single failure
SFD	spent fuel damage
SFG	structural fill granular
SFHM	spent fuel handling machine
SFP	spent fuel pool
SFPCCS	spent fuel pool cooling and cleanup system
SFPCL	SFP cleanup loop
SFR	spent fuel rack
SG	steam generator
SGBDS	steam generator blowdown system
SGI	safeguard information
SGMSR	steam generator maximum steaming rate
SGTR	steam generator tube rupture
SI	safety injection
SI units	International System of Units
SIAS	safety injection actuation signal
SIF	stress intensification factor
SIFT	safety injection filling tank
SIP	safety injection pump
SIRCP	startup of an inactive reactor coolant pump
SIS	safety injection system
SIT	1) safety injection tank 2) structural integrity test
SIT-FD	safety injection tank with fluidic device
SKN	Shin-Kori nuclear power plant

DCD Chapter 6 ACRONYM AND ABBREVIATION LIST

Table 3.4-1 (2 of 3)

Item No.	Equipment No.	Equipment Description	Location	Building	Floor Elevation	SSC level from the Flood Elevation EL.102'-0"
24	1-491-V-003	PZR SURGE LINE SAMPLE CTMT ISOLATION VALVE	114-C01B	RCB	114'-0"	above
25	1-491-V-005	PZR STEAM SPACE SAMPLE CTMT ISOLATION VALVE	114-C01B	RCB	114'-0"	above
26	1-541-J-LT-1123A	STEAM GENERATOR 2 LEVEL TRANSMITTER (WIDE)	114-C01B	RCB	114'-0"	above
27	1-541-J-LT-1123B	STEAM GENERATOR 2 LEVEL TRANSMITTER (WIDE)	114-C01B	RCB	114'-0"	above
28	1-451-V-0203	PRESSURIZER AUXILIARY SPRAY VALVE	116-C04	RCB	116'-0"	above
29	1-431-J-PT-102A	PRESSURIZER PRESSURE TRANSMITTER	136-C01A	RCB	136'-0"	above
30	1-441-V-0614	SIT 4 OUTLET ISOLATION VALVE	136-C01A	RCB	136'-0"	above
31	1-441-V-0644	SIT 1 OUTLET ISOLATION VALVE	136-C01A	RCB	136'-0"	above
32	1-431-J-PT-102B	PRESSURIZER PRESSURE TRANSMITTER	136-C01B	RCB	136'-0"	above
33	1-441-V-0624	SIT 2 OUTLET ISOLATION VALVE	136-C01B	RCB	136'-0"	above
34	1-441-V-0634	SIT 3 OUTLET ISOLATION VALVE	136-C01B	RCB	136'-0"	above
35	1-431-V-130	~~SAFETY DEPRESSURIZATION AND VENT VALVE (POSRV)~~	136-C02	RCB	136'-0"	above
36	1-431-V-131	~~SAFETY DEPRESSURIZATION AND VENT VALVE (POSRV)~~	136-C02	RCB	136'-0"	above
37	1-431-V-132	~~SAFETY DEPRESSURIZATION AND VENT VALVE (POSRV)~~	136-C02	RCB	136'-0"	above
38	1-431-V-133	~~SAFETY DEPRESSURIZATION AND VENT VALVE (POSRV)~~	136-C02	RCB	136'-0"	above
39	1-431-V-134	~~SAFETY DEPRESSURIZATION AND VENT VALVE (POSRV)~~	136-C02	RCB	136'-0"	above
40	1-431-V-135	~~SAFETY DEPRESSURIZATION AND VENT VALVE (POSRV)~~	136-C02	RCB	136'-0"	above
41	1-431-V-136	~~SAFETY DEPRESSURIZATION AND VENT VALVE (POSRV)~~	136-C02	RCB	136'-0"	above
42	1-431-V-137	~~SAFETY DEPRESSURIZATION AND VENT VALVE (POSRV)~~	136-C02	RCB	136'-0"	above
43	1-433-V-410	PRESSURIZER VENT ISOLATION VALVE	156-C01	RCB	156'-0"	above
44	1-433-V-411	PRESSURIZER VENT ISOLATION VALVE	156-C01	RCB	156'-0"	above
45	1-433-V-412	PRESSURIZER VENT ISOLATION VALVE	156-C01	RCB	156'-0"	above
46	1-433-V-413	PRESSURIZER VENT ISOLATION VALVE	156-C01	RCB	156'-0"	above
47	1-441-V-0605	SIT 4 VENT ISOLATION VALVE	156-C01	RCB	156'-0"	above
48	1-441-V-0606	SIT 2 VENT ISOLATION VALVE	156-C01	RCB	156'-0"	above

POSRV (Typical)

Table 3.6-1

High- and Moderate-Energy Fluid Systems

System(1)(2)	High Energy	Moderate Energy
Reactor Coolant System	○	
~~Safety Depressurization and Vent System~~ ~~(3)~~ Reactor Coolant Gas Vent System	○	
Safety Injection~~/Shutdown Cooling~~ System	○	
Shutdown Cooling System	O	
Chemical and Volume Control System	○	
Steam Generator Blowdown System(4)	○	
Component Cooling System		○
Spent Fuel Pool Cooling and Cleanup System		○
~~Containment Air Monitoring/Sampling System~~ Process and Effluent Radiation Monitoring and Sampling System		○
Containment Spray System		○
Essential Service Water System		○
Main Steam System	○	
~~Auxiliary Feedwater Pump Turbine System~~	~~○~~	
Main Feedwater System	○	
Auxiliary Feedwater System (6)	O	~~○~~
Auxiliary Steam System	○	
Emergency Diesel Generator System(5)	○	
Diesel Fuel Oil Transfer System		○
Fire Protection System		○
~~Station Heating System~~		~~○~~
Drain System		○
Essential~~/Normal~~ Chilled Water System		○
Plant Chilled Water System		O
~~Breathing Air System~~ Compressed Air System		○
~~Instrument Air System~~		~~○~~
~~Service Air System~~		~~○~~

(1) Systems classified as high-energy are either totally or partially high-energy. If portions of system are high-energy, it is classified as high-energy system. The portions of high-energy piping are identified in Figures 3.6-1 through 3.6-12.

(2) Systems or portions of systems outside the containment building and auxiliary building are excluded from this table.

~~(3) The piping portion from 2-in piping located at downstream of POSRV to the first isolation valve is high-energy piping and the rest is moderate-energy piping.~~

(4) Wet layup recirculation system is classified as moderate-energy system.

(5) Subsystems other than an EDG engine starting air system are classified as moderate-energy systems.

(6) Subsystems other than an auxiliary feedwater pump turbine subsystem are classified as moderate-energy systems.

discharges from the POSRVs and the reactor coolant gas vent system (RCGVS).

3.8.3.1.8 In-Containment Refueling Water Storage Tank

The in-containment refueling water storage tank (IRWST) provides storage of refueling water, a single source of water for the safety injection and containment spray pumps, and a heat sink for ~~the safety depressurization system~~. The IRWST is annular and uses the lower section of the internal structure as its outer boundary. The IRWST is lined with a stainless steel liner plate to prevent leakage. The IRWST consists of the top and bottom slab and the exterior wall. The bottom slab of IRWST rests on the reactor containment building basemat, and the top and bottom slabs are rigidly connected to the secondary shield wall. The design of the IRWST considers pressurization as a result of the reactor containment building systems design basis accident. Refer to Section 6.8 for a description of the IRWST.

3.8.3.1.9 Holdup Volume Tank

The holdup volume tank (HVT) is a rectangular structural tank located between the primary shield wall and the IRWST inner wall. A screen is provided at the top of the HVT to prevent debris from getting into the tank. The HVT has a sump with pumps to measure the leakage rate and route the liquid to the liquid waste management system. During an accident, the water from breaks and the reactor containment building spray is collected in the HVT and overflows into the IRWST. Refer to Section 6.8 for a description of the HVT.

3.8.3.1.10 Operating and Intermediate Floors

The operating floor provides access for operating personnel functions and biological shielding. Intermediate floors provide access to equipment and components. The operating floor is located at elevation 156 ft 0 in, and intermediate floors are located at elevations 114 ft 0 in and 136 ft 6 in. These floors consist of reinforced concrete or steel grating supported by structural steel framing that spans between the containment wall and the secondary shield wall. The steel framing has a horizontally sliding connection at the containment wall side to allow axial displacement of framing due to seismic displacement and thermal expansion. Openings are provided in the floor for equipment removal.

monolithic ring that surrounds the reactor vessel (RV). Penetrations in the PSW are provided for the primary loop piping and cavity ventilation system.

3.8A.1.1.2.2 In-Containment Refueling Water Storage Tank (IRWST)

The in-containment refueling water storage tank (IRWST) provides storage of refueling water, a single source of water for the safety injection and containment spray pumps, and a heat sink for ~~the safety depressurization system~~ discharges from the POSRVs and the reactor coolant gas vent system (RCGVS). The IRWST is annulus shaped and uses the lower section of the internal structure as its outer boundary. The IRWST is provided with a stainless steel liner to prevent leakage. The IRWST consists of a top and bottom slab and exterior and interior walls. The bottom slab of the IRWST rests on the containment basemat and the top and bottom slab are rigidly connected to the secondary shield wall (SSW).

3.8A.1.1.2.3 Secondary Shield Wall

The SSW is a circular reinforced concrete structure that, together with the refueling pool walls, encloses the nuclear steam supply system (NSSS) equipment, except for the reactor vessel. The SSW major reinforcing, which is necessary for carrying the seismic overturning moment, is anchored into the containment basemat in a manner similar to the PSW. The SSW provides lateral support for the steam generators, reactor coolant pumps (RCPs), and pressurizer and supports the operating and intermediate floors.

3.8A.1.2 Structural Materials

The principal construction materials for the reactor containment building are concrete, liner plate, reinforcing steel, and prestressing steel. The minimum specified material properties are used for the conservative design of the reactor containment building. The detailed material properties considered in the analysis and design of the reactor containment building are as follows:

3.8A.1.2.1 Concrete

The concrete used in the analysis and design of the reactor containment building is normal weight concrete.

3.8A.1.4.3.1.4 Typical Rebar Arrangement

The typical rebar arrangements for PSW are presented in the Table 3.8A-21.

3.8A.1.4.3.1.5 Conclusion

The PSW concrete section strengths determined from the criteria in ACI 349 are sufficient to resist the design basis loads. It is feasible to design and construct the structural components that are considered. The assumptions envelop the given parameters so the design is adequate for any site-specific conditions within the parameters.

3.8A.1.4.3.2 IRWST

3.8A.1.4.3.2.1 Description

discharges from the POSRVs and the RCGVS.

The IRWST provides storage of refueling water, a single source of water for the safety injection and containment spray pumps, and a heat sink for ~~the safety depressurization system~~. The IRWST is annulus shaped and uses the lower section of the internal structure as its outer boundary. A stainless steel liner is provided inside the IRWST.

The IRWST consists of a top and bottom slab and exterior and interior walls. The bottom slab rests on the containment basemat and the top and bottom slabs are rigidly connected to the SSW. The interior wall is the reactor containment SSW. The fill concrete portion is placed between the SSW and PSW from El. 78 ft 0 in to El. 100 ft 0 in.

The exterior wall thickness of the IRWST is 0.91 m (3 ft), the same as the bottom and roof slabs. The inner radius of the exterior wall is 21.89 m (71 ft 10 in.) and the outer radius of the interior wall (SSW) is 16.15 m (53 ft 0 in.). The top elevation of the IRWST bottom slab is El. 81 ft 0 in. The top elevation of the IRWST roof slab is El. 100 ft 0 in. The normal water level in the IRWST is at El. 93 ft 0 in.

1) One narrow range and four wide range (0 to 100 percent) percent level indication channels are provided.

2) The narrow range level channel (non-class 1E) is used to detect the presence of fluid in the reactor cavity. The narrow range channel actuates an alarm in the MCR and RSR to alert the operator that the reactor cavity is being filled with water. The narrow range level channel is identified in Subsection 9.3.3.

3) The four wide range (0 to 100 percent) channels are powered by the vital instrument bus. Level indication is provided in the MCR and RSR to allow the operator to monitor reactor cavity fluid level after an accident.

6.8.3.2 Temperature

Four IRWST fluid temperature measurement channels are provided.

One channel is designated as non-class 1E and provides IRWST a fluid temperature indication inside the containment.

The three remaining channels are designed to electrical Class 1E requirements. These channels provide a IRWST fluid temperature indication in the MCR; a high/low temperature alarm of 10.0 °C (50 °F) and 48.88 °C (120 °F) is provided to alert the operator when IRWST temperature approaches the minimum or maximum allowable Technical Specification (Chapter 16) limits. The range of the channels is adequate to cover the maximum temperature expected to occur during operation of the POSRVs.

6.8.3.3 Pressure

Four wide range channels powered by the vital instrument bus are provided to indicate the IRWST pressure. Each channel provides an indication of IRWST pressure in the MCR and RSR. The range of the instrumentation is adequate to cover the maximum pressure expected to occur during ~~safety depressurization and vent system~~ operation.

the POSRVs and the RCGVS