APR1400

DESIGN CONTROL DOCUMENT TIER 2

CHAPTER 1
INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

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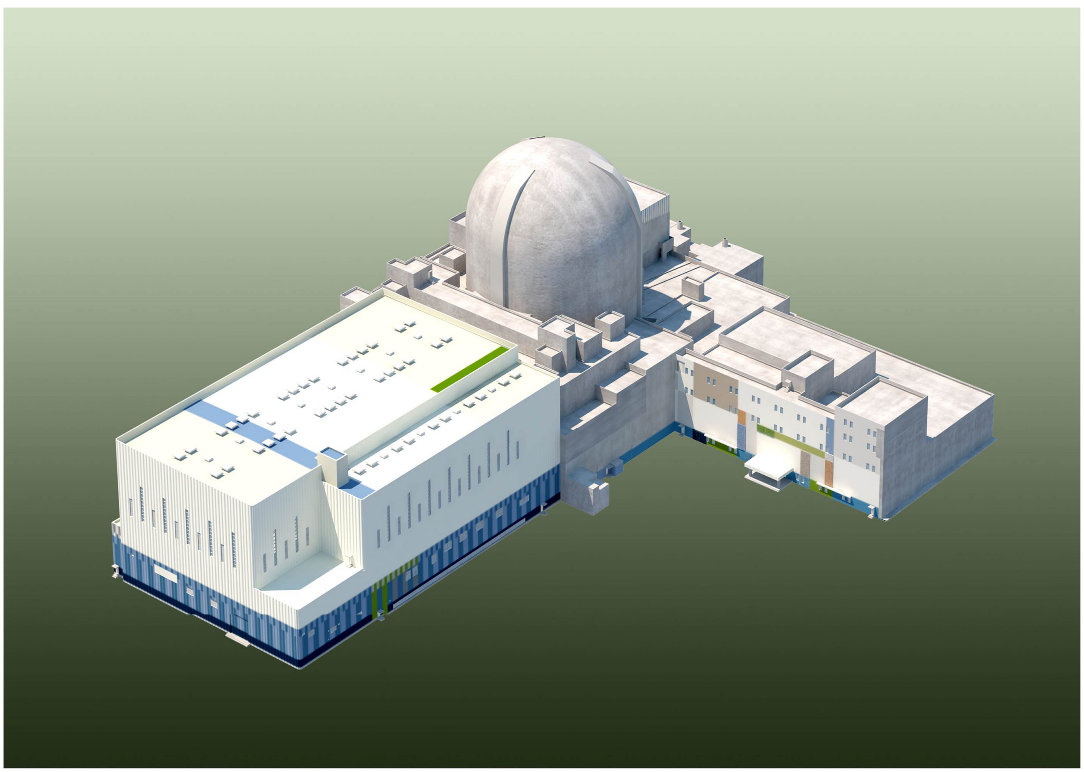

KOREA ELECTRIC POWER CORPORATION

KOREA HYDRO & NUCLEAR POWER CO., LTD

CHAPTER 1 – INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

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ACRONYM AND ABBREVIATION LIST

AAC	alternate alternating current
AAFAS	alternate auxiliary feedwater actuation signal
AB	auxiliary building
ABCAEES	auxiliary building controlled area emergency exhaust system
ABD	abnormal blow down
AC	alternating current
ACC	analysis computer cabinet
ACI	american concrete institute
ACP	auxiliary charging pump
ACR	advanced control room
ACU	air cleaning unit
ADV	atmospheric dump valve
AE	architect engineer
AEA	Atomic Energy Act
AEB	Atomic Energy Bureau
AF	auxiliary feedwater
AFAS	auxiliary feedwater actuation signal
AFW	auxiliary feedwater
AFWS	auxiliary feedwater system
AFWST	auxiliary feedwater storage tank
AHU	air handling unit
AI	analog input
AICC	Adiabatic Isochoric Complete Combustion
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
ALARA	as low as is reasonably achievable
ALI	annual limit on intake
ALMS	acoustic leak monitoring system
ALWR	advanced light water reactor

AM	accident management
AMCA	Air Movement and Control Association
AMI	accident monitoring instrumentation
AMS	Aerospace Material Specification
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOC	averted off-site property damage costs
AOE	averted occupational exposures
AOO	anticipated operational occurrence
AOP	abnormal operating procedure
AOSC	averted on-site costs
AOV	air operated valve
APC	auxiliary process cabinet
APC-S	auxiliary process cabinet-safety
APD	amplified probability distribution
APE	averted public exposure
API	American Petroleum Institute
APR	Advanced Power Reactor
APR1400	Advanced Power Reactor 1400
APSD	auto-power spectral density
APWR	advanced pressurized water reactor
ARI	Air-Conditioning and Refrigeration Institute
ARM	annunciator response model
ARMS	area radiation monitoring system
ARO	1) all rod out 2) additional reactor operator
ART	adjusted reference temperature
AS	1) accident sequence analysis 2) auxiliary steam
ASCE	American Society of Civil Engineers
ASD	alternate shutdown

ASEP	accident sequence evaluation program
ASHRAE	American Society of Heating, Refrigeration, and Air-Conditioning Engineers
ASI	axial shape index
ASIC	Application Specific Integrated Circuit
ASME	American Society of Mechanical Engineers
AST	alternative source term
ASTM	American Society of Testing and Materials
ATP	authorization to proceed
ATS	automatic turbine startup
ATWS	anticipated transient without scram
AUC	alarm unit cabinet
AUX	auxiliary
AVT	all volatile treatment
AWP	automatic withdrawal prohibit
AWS	American Welding Society
BABT	boric acid batching tank
BABT	boric acid bating tank
BAC	boric acid concentrator
BAMP	boric acid makeup pump
BAST	boric acid storage tank
BDAS	boron dilution alarm system
BDBE	beyond design basis event
BDBEE	beyond design basis external events
BDD	binary decision diagram
BDS	blowdown subsystem
BHP	brake horsepower
BISI	bypassed and inoperable status indication
BLOP	bearing lift oil pump
BLPB	branch line pipe break
BM	boronometer

BMT	basemat melt through
BOC	beginning of cycle
BOL	beginning of life
BOP	balance of plant
BP	bistable processor
BRL	Ballistic Research Laboratory
BTP	branch technical position
BWR	boiling water reactor
C&L	closing and latching
CAM	continuous air monitor
CAP	corrective action program
CAR	corrective action request
CAREM	code-accuracy-based realistic evaluation method
CAS	1) compressed air system 2) central alarm station
CAT	construction acceptance test
CAV	cumulative absolute velocity
CBD	continuous blowdown
CBDTM	cause-based decision tree methodology
CBP	computer based procedure
CBV	cation bed ion exchanger vessel
CC	component cooling water
CCDP	conditional core damage probability
CCF	common - cause failure
CCFP	conditional containment failure probability
CCG	control channel gateway
CCL	component control logic
CCP	centrifugal charging pump
CCS	component control system
CCTV	closed-circuit television
CCW	component cooling water

CCW HX	component cooling water heat exchanger
CCWLLSTAS	component cooling water low-low surge tank actuation signal
CCWPH	component cooling water pump house
CCWS	component cooling water system
CD	1) complete dependence (HRA) 2) condensate system
CDF	core damage frequency
CDI	conceptual design information
CEA	control element assembly
CEAC	1) control element assembly calculator 2) CEA calculator
CEAE	control elemnet assembly ejection
CEDE	committed effective dose equivalent
CEDM	control element drive mechanism
CEDMCS	control element drive mechanism control system
CET	core exit thermocouple
CEUS	central and eastern United States
CF	cavity flooding
CF/SPM	critical functions / success path monitoring
CFF	containment failure frequency
CFM	critical function monitoring
CFR	code of federal regulations
CFS	cavity flooding system
CGI	commercial grade item
CHCS	containment hydrogen control system
CHF	critical heat flux
CHMS	containment hydrogen monitoring system
CHR	containment heat removal
CHRS	containment heat removal system
CI	containment isolation
CIAS	containment isolation actuation signal

CILRT	containment integrated leakage rate test
CIM	component interface module
CIP	cleaning in place
CIS	containment isolation system
CIV	containment isolation valve
CLD	control logic diagram
CLVPS	containment low volume purge system
CM	1) condition monitoring 2) containment monitoring
CMAA	Crane Manufacturers Association of America
CMTR	Certified Material Test Report
CNMT	containment
COE	cost of enhancement
COL	combined license
COLA	combined license application
COLR	core operating limits report
COLSS	core operating limit supervisory system
COMP	compound
COO	chief operations officer
CP	condensate polishing
CPC	core protection calculator
CPCS	core protection calculator system
CPCSC	core protection calculator system cabinet
CPIAS	containment purge isolation actuation signal
CPM	control panel multiplexer
CPP	CEA position processor
CPS	condensate polishing system
CPU	central processing unit
CR	control room
CRC	cyclical redundancy check
CRDS	control rod drive system

CRE	control room envelope
CREACS	control room emergency makeup air cleaning system
CREVAS	control room emergency ventilation actuation signal
CRF	carryout rate fraction
CRTF	central receiver test facility
CRX	Canadian research reactor
CS	1) containment spray 2) core support 3) communication section
CSAS	containment spray actuation signal
CSB	core support barrel
CSDRS	certified seismic design response spectra
CSF	critical safety function
CSHX	containment spray heat exchanger
CSP	containment spray pump
CSS	containment spray system
CST	condensate storage tank
CT	condensate storage and transfer system
CTS	concentrate treatment system
CUF	cumulative usage factor
CV	1) control valve 2) chemical and volume control system
CVAP	comprehensive vibration assessment program
CVCS	chemical and volume control system
CVN	Charpy V-notch
CW	circulating water
CWP	circulating water pump
CWS	circulating water system
CWT	chemical waste tank
D3CA	diversity and defense-in-depth coping analysis
DA	data analysis

DAC	1) derived air concentration 2) design acceptance criteria
DAS	diverse actuation system
DAU	data acquisition unit
DAW	dry active waste
DBA	design basis accident
DBE	design basis event
DBFL	design basis flooding level
DBPB	design basis pipe break
DC	1) direct current 2) design certification
DCD	Design Control Document
DCF	1) dose conversion factor 2) dynamic containment failure
DCH	direct containment heating
DCN	design change notice
DCN-I	data communication network-information
DCS	distributed control system
DDCC	drawing and document control center
DDE	deep dose equivalent
DDT	deflagration to detonation transition
DE	dose equivalent
DEDLSB	double ended discharge leg slot break
DEG/PD	double-ended guillotine at the pump discharge leg
DEHLSB	double ended hot leg slot break
DELLOC	double-ended break of the letdown line outside containment
DESLSB	double ended suction leg slot break
DET	decomposition event tree
DF	decontamination factor
DFL	dynamic fluid loads
DFOT	diesel fuel oil tank

DG	diesel generator
DGA	diesel generator area
DI	1) digital input 2) design implementation
DIF	dynamic impact factor
DIS	diverse indication system
DIT	discrete integral transport
DLF	dynamic load factor
DLM	Diffusion Layer Model
DMA	diverse manual ESF actuation
DMDS	diagnostic monitoring and display system
DN	nominal diameter
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DO	1) dissolved oxygen 2) digital output
DOOSAN	Doosan Heavy Industries & Construction Co., Ltd
DOT	U.S. department of transportation
DPS	diverse protection system
DRC	dropped rod contact
DRCS	digital rod control system
DS	disconnect switch
DVI	direct vessel injection
DW	dead weight
DWST	demineralized water storage tank
EAB	exclusion area boundary
EAC	emergency alternating current
EBD	emergency blowdown
EBS	estimated break size
ECC	emergency core cooling
ECCS	emergency core cooling system

ECF	early containment failure
ECSA	electrical conductor sealing assembly
ECSBS	emergency containment spray backup system
ECT	Eddy current testing
ECWS	essential chilled water system
EDE	effective dose equivalent
EDECAIES	emergency diesel engine combustion air intake and exhaust system
EDECWS	emergency diesel engine cooling water system
EDEFOS	emergency diesel engine fuel oil system
EDELS	emergency diesel engine lubrication system
EDESS	emergency diesel engine starting system
EDG	emergency diesel generator
EDGB	emergency diesel generator building
EDH	electric duct heater
EDMG	extensive damage mitigation guideline
EDT	equipment drain tank
EF	1) error factor 2) engineered safety features actuation system
EFDS	equipment and floor drainage system
EFPD	effective full-power day
EFPY	effective full-power year
EHC	electro-hydraulic control
EJMA	Expansion Joint Manufacturers Association
ELAP	extended loss of ac power
EM	seismic monitoring
EMC	electromagnetic compatibility
EMI	electromagnetic interference
ENFMS	ex-core neutron flux monitoring system
EO	electrical operator
EOB	end of blowdown
EOC	end of cycle

EOF	emergency operation facility
EOL	end of life
EOP	emergency operating procedure
EOPR	end of post-reflood
EOR	end of reflood
EPA	1) electrical penetration assembly 2) U.S Environmental Protection Agency
EPD	external pressure differential
EPG	emergency procedure guideline
EPM	engineering procedures manual
EPRI	Electric Power Research Institute
EQ	environmental qualification
EQAP	engineering quality assurance procedure
ERDS	emergency response data system
ERF	emergency response facility
ERVC	external reactor vessel cooling
ES	equipment survivability
ESA	extension shaft assembly
ESCM	ESF-CCS soft control module
ESF	engineered safety features
ESF-CCS	engineered safety features – component control system
ESR	electro-hydraulic actuated spring return
ESW	essential service water
ESWPS	essential service water pump structure
ESWS	essential service water system
ET	event tree
ETAP	electrical transient analyzer program
ETS	emergency trip system
EVSE	ex-vessel steam explosion
EWT	equipment waste tank
FA	flame acceleration

FAC	flow-accelerated corrosion
FACT	fuel assembly compatibility test
FAP	fuel alignment plate
FATT	fracture appearance transition temperature
FC	fully closed
FCAW	flux cored arc welding
FCI	fuel-coolant interaction
FCR	field change request
FDS	floor drain system
FDT	1) floor drain tank 2) functional definition table
FEI	fluid-elastic instability
FEM	finite element model
FF	flash fraction
FHA	1) fuel handling accident 2) fuel handling area 3) fire hazards analysis
FHAEES	fuel handling area emergency exhaust system
FHEVAS	fuel handling area emergency ventilation actuation signal
FHS	fuel handling system
FIDAS	fixed in-core detector amplification system
FIRS	foundation input response spectra
FIV	flow-induced vibration
FLB	feedwater line break
FLC	factored load category
FLEX	diverse and flexible coping strategies
FME	foreign material exclusion
FMEA	failure modes and effects analysis
FO	fully open
FOM	fiber optic modem
FP	fire protection

FPD	flat panel display
FPDIL	full-power-dependent insertion limit
FPP	fire protection plan
FPS	fire protection system
FRA/FA	functional requirements analysis and function allocation
FS	factor of safety
FSAR	final safety analysis report
FSCEA	full-strength CEA
FSSA	fire safe shutdown analysis
FT	fault tree
FTC	fuel temperature coefficient
FV	Fussell-Vesely
FW	feedwater
FWCS	feedwater control system
FWCV	feedwater control valve
FWLB	feedwater line break
FWPB	feedwater pipe break
GC	group controller
GCB	generator circuit breaker
GCP	general control procedure
GDC	general design criteria (of 10 CFR Part 50, Appendix A)
GI	gastrointestinal
GIB	gas insulated bus
GIS	1) event-generated iodine spike 2) gas insulated substation
GL	Generic Letter
GMRS	ground motion response spectra
GOP	general operating procedure
GOTHIC	generation of thermal-hydraulic information for containment
GRID-LOOP	grid-centered loss of offsite power
GRID-SBO	grid-centered station blackout

GRS	gaseous radwaste system
GRV	gravity roof ventilator
GSE	gland steam packing exhauster
GSERMS	gas stripper effluent radiation monitoring system
GSI	Generic Safety Issue
GTAW	gas tungsten arc weld
GTG	gas turbine generator
GTRN	general transient
GWMS	gaseous waste management system
GWR	guided wave radar
HA	human action
HAZ	heat-affected zone
HCBD	high-capacity blowdown
HCLPF	high confidence of low probability of failure
HCOG	hydrogen control owner's group
HCR/ORE	human cognitive reliability / operator reliability experiment
HD	1) high dependence (HRA) 2) heater drain 3) HSI design
HDSR	historical data storage and retrieval
HE	human error
HED	human engineering discrepancy
HEI	Heat Exchange Institute
HELB	high-energy line break
HEP	human error probability
HEPA	high-efficiency particulate air
HF	human factors
HFE	human factors engineering
HFEPP	human factors engineering program plan
HFP	hot full power
HFT	hot functional test

HG	containment hydrogen control system
HHAS	high-humidity actuation signal
HI	hydrogen igniter
HIC	high-integrity container
HID	high-intensity discharge
HIS	Hydraulic Institute Standard
HJTC	heated junction thermocouple
HLI	hot leg injection
HMS	hydrogen mitigation system
HP	high pressure
HPCI	high-pressure coolant injection
HPME	high-pressure melt ejection
HPPT	high pressurizer pressure trip
HPS	Health Physics Society
HPSC	high-pressure seal cooler
HRA	human reliability analysis
HRAS	high radiation actuation signal
HRHF	hard rock high frequency
HRR	heat release rate
HSB	hot standby
HSD	hot shutdown
HSGL	high steam generator level
HSI	human-system interface
HSIS	human-system interface system
HSS	high safety significance
HT	high temperature
HVAC	heating, ventilation, and air conditioning
HVT	holdup volume tank
HX	heat exchanger
HZP	hot zero power
I&C	instrumentation and control

I/O	input/output
IA	instrument air
IAS	instrument air system
IBA	inner barrel assembly
ICC	inadequate core cooling
ICCMS	inadequate core cooling monitoring system
ICDP	incremental core damage probability
ICI	in-core instrumentation
ICR	information and control requirement
ICRP	International Commission on Radiological Protection
ID	1) inner diameter 2) identification
IE	1) initiating events analysis 2) Inspection and Enforcement
IEC	International Electrotechnical Commission
IED	internal effective dose
IEEE	Institute of Electrical and Electronics Engineers
IEPRA	internal events probabilistic risk assessment
IF	internal flooding analysis
IFPD	information flat panel display
IHA	1) integrated head assembly 2) important human action
ILRT	integrated leak rate test
INPO	Institute of Nuclear Power Operations
INVINJ	in-vessel injection
IOSGADV	inadvertent opening of a steam generator atmospheric dump valve
IP	implementation plan
IPB	isolated phase bus
IPS	information processing system
IRSF	interim radwaste storage facility
IRWST	in-containment refueling water storage tank

IS	internal structure
ISA	Instrument Society of America
ISG	Interim Staff Guidance
ISI	inservice inspection
ISLOCA	intersystem loss-of-coolant accident
ISM	independent support motion
ISRS	in-structure response spectra
IST	inservice testing
ISV	1) integrated system validation 2) intermediate stop valve
ITA	important to availability
ITAAC	inspections, tests, analyses, and acceptance criteria
ITC	isothermal temperature coefficient
ITP	1) interface and test processor 2) inspection and test plan
ITS	1) issue tracking system 2) important to safety
IV	intercept valve
IVMS	internal vibration monitoring system
IVSE	in-vessel steam explosion
IW	in-containment refueling water storage system
IWPP	independent water and power producer
IWSS	in-containment water storage system
IX	ion exchange
J-R	J-resistance
JOG	Joint Owner Group
KAERI	Korea Atomic Energy Research Institute
KEPCO	Korea Electric Power Corporation
KEPCO E&C	KEPCO Engineering & Construction Co., Inc.
KEPCO NF	KEPCO Nuclear Fuel Co., Ltd.
KHNP	Korea Hydro & Nuclear Power Co., Ltd.

KWU	Kraftwerk Union AG
LAN	local area network
LASRT	low-activity spent resin storage tank
LB	large break
LBB	leak before break
LBLOCA	large-break loss-of-coolant accident
LC	1) lock close 2) loop controller 3) load center
LCF	late containment failure
LCL	local coincidence logic
LCO	limiting conditions for operation
LCP	local control panel
LCS	local control station
LD	low dependence (HRA)
LDLB	letdown line break
LDP	large display panel
LE	LERF Analysis
LED	light-emitting diode
LEL	lower electrical limit
LERF	large early release frequency
LFW	loss of normal feedwater flow
LGS	lower group stop
LHGR	linear heat generation rate
LHR	linear heat rate
LHS	Latin hypercube sampling
LL	large LOCA
LLHS	light load handling system
LMFBR	liquid metal cooled fast breeder reactor
LO	local operator
LOAC	loss of nonemergency ac power

LOCA	loss-of-coolant accident
LOCCW	loss of component cooling water
LOCV	loss of condenser vacuum
LODC	loss of dc power
LOESW	loss of essential service water
LOF	1) left-out-force 2) loss of flow
LOFW	1) loss of normal feedwater flow 2) loss of main feedwater
LOIA	loss of instrument air
LOLA	loss of large areas
LOOP	loss of offsite power
LP	low pressure
LPD	local power density
LPLD	low PZR pressure and low DNBR
LPMS	loose parts monitoring system
LPSD	low power and shutdown
LPZ	low population zone
LRC	locked rotor current
LRF	large release frequency
LRS	liquid radwaste system
LSGL	low steam generator level
LSS	lower support structure
LSSB	large secondary side break
LSSS	limiting safety system setting
LST	lowest service temperature
LT	low temperature
LTC	long-term cooling
LTOP	low temperature overpressure protection
LUHS	loss of normal access to ultimate heat sink
LWMS	liquid waste management system

LWR	light water reactor
M&E	mass and energy
MAAP	modular accident analysis program
MBLOCA	medium break loss-of-coolant accident
MBV	mixed-bed ion exchanger vessel
MCA	multiple compartment analysis
MCC	motor control center
MCCI	molten corium concrete interaction
MCL	main coolant loop
MCR	main control room
MDNBR	minimum departure from the nucleate boiling ratio
MDS	makeup demineralizer system
MELB	moderate-energy line break
MF	membrane filter
MFIV	main feedwater isolation valve
MFLB	main feedwater line break
MFS	main feedwater system
MFW	main feedwater
MG	motor- generator
MG Set	motor-generator set
MI	minimum inventory
ML	manufacturing license
MMC	missing mass correction
MMI	modified Mercalli intensity
MOP	main oil pump
MORS	membrane oxygen removal subsystem
MOV	motor-operated valve
MRP	materials reliability program
MS	main steam
MSADV	main steam atmospheric dump valve
MSADVIV	MSADV isolation valve

MSE	main steam enclosure
MSGTR	multiple steam generator tube rupture
MSIS	main steam isolation signal
MSIV	main steam isolation valve
MSIVBV	main steam isolation valve bypass valve
MSLB	main steam line break
MSO	multiple spurious operation
MSPI	mitigating systems performance index
MSR	1) moisture separator reheater 2) maximum steaming rate
MSS	main steam system
MSSV	main steam safety valve
MSV	main steam valve
MSVH	main steam valve house
MT	main transformer
MTC	moderator temperature coefficient
MTP	maintenance test panel
MUX	multiplexer
MWD/MTU	megawatt-days per metric ton of Uranium
NA	not applicable
NDE	nondestructive examination
NDRC	national defense research council
NDTT	nil-ductility transition temperature
NEC	National Electrical Code
NEI	Nuclear Energy Institute
NEM	nodal expansion method
NEMA	National Electrical Manufacturers Association
NFPA	National Fire Protection Association
NFR	new fuel rack
NI	nuclear island
NIMS	NSSS integrity monitoring system

NLO	non-licensed operator
NNS	non-nuclear safety
NO	normal operation
NOP	normal operating procedure
NP	non-Class 1E 13.8 kV auxiliary power system
NPCS	NSSS process control system
NPP	nuclear power plant
NPS	nominal pipe size
NPSH	net positive suction head
NPSHA	net positive suction head available
NPSHR	net positive suction head required
NPSS	normal primary sampling system
NR	narrow range
NRC	United States Nuclear Regulatory Commission
NRV	1) non-return check valve 2) net present value
NS	non-seismic
NSA	neutron source assembly
NSAC	Nuclear Safety Analysis Center
NSSS	nuclear steam supply system
NT	normal torque
NTS	Nevada Test Site
NTTF	near term task force
NUREG	NRC technical report designation
OA	operational assessment
OBE	operating basis earthquake
OD	outside diameter
ODCM	offsite dose calculation manual
OECD	Organization for Economic Cooperation and Development
OER	operating experience review
OFAF	oil forced air forced

OHLHS	overhead heavy load handling system
OM	operator module
ONAF	oil natural air forced
ONAN	oil natural air natural
OPR	Optimized Power Reactor
ORNL	Oak Ridge National Laboratory
OSC	operational support center
P-CCS	process-component control system
P-T Limit	pressure-temperature limitation
P&ID	piping and instrumentation diagram
PA	public address
PABX	private automatic branch telephone exchange
PACU	packaged air conditioning unit
PAL	personnel air lock
PAR	passive autocatalytic recombiner
PASS	post-accident sampling system
PAT	power ascension test
PAU	physical analysis unit
PBX	plant telephone exchange
PC	prime contractor
PCA	primary coolant activity
PCB	power circuit breaker
PCMI	pellet cladding mechanical interaction
PCP	project control procedure
PCS	power control system
PCT	peak cladding temperature
PCWS	plant chilled water system
PDIL	power-dependent insertion limit
PDS	plant damage state
PED	piping evaluation diagram
PERMSS	process and effluent radiation monitoring and sampling system

PF	1) penalty factor 2) 4.16 kV Class 1E auxiliary power
PGA	peak ground acceleration
PI	pressure indicator
PIS	pre-accident iodine spike
PIV	pressure isolation valve
PLC	programmable logic controller
PLCS	pressurizer level control system
PLHGR	peak linear heat generation rate
PLM	priority logic module
PLOCCW	partial loss of component cooling water
PLOESW	partial loss of essential service water
PMF	probable maximum flood
PMP	probable maximum precipitation
PMWP	probable maximum winter precipitation
PNS	permanent non-safety
POL	power operating limit
PORV	power-operated relief valve
POS	plant operational state
POSRV	pilot operated safety relief valve
POV	power-operated valve
PPCS	pressurizer pressure control system
PPM	project procedures manual
PPS	1) plant protection system 2) preferred power supply
PPSC	plant protection system cabinet
PRA	probabilistic risk assessment
PRCSCD	RCS pressure at the time of core damage
PRM	process radiation monitor
PRMS	process radiation monitoring system
PRV	process representative value

PSA	probabilistic safety assessment
PSAR	preliminary safety analysis report
PSCEA	part-strength control element assembly
PSD	power spectral density
PSHA	probabilistic seismic hazard analysis
PSI	preservice inspection
PSR	pneumatically actuated spring return
PSW	primary shield wall
PTC	peak cladding temperature
PTLR	pressure and temperature limits report
PTS	1) pressurized thermal shock 2) primary to secondary
PV	preliminary validation
PVNGS	Palo Verde nuclear generating station
PVRC	Pressure Vessel Research Committee
PWR	pressurized water reactor
PWSCC	primary water stress corrosion cracking
PX	primary sampling system
PZR	pressurizer
QA	quality assurance
QAP	quality assurance procedure
QAPD	quality assurance program description
QIAS	qualified indication and alarm system
QIAS-N	qualified indication and alarm system – non-safety
QIAS-P	qualified indication and alarm system – p
QU	quantification
R/O	reverse osmosis
RADTRAD	Radionuclide Transport, Removal, and Dose
RAM	random access memory
RAP	reliability assurance program
RAW	risk achievement worth

RB	reactor building
RC	reactor coolant
RCA	radiologically contolled area
RCB	reactor containment building
RCC	remote control center
RCCA	rod cluster control assembly
RCFC	reactor containment fan cooler
RCGV	reactor coolant gas vent
RCGVS	reactor coolant gas vent system
RCIC	reactor core isolation cooling
RCL	reactor coolant loop
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCPS	reactor power cutback system
RCPSSSS	reactor coolant pump shaft speed sensing system
RCPVMS	reactor coolant pump vibration monitoring system
RCS	reactor coolant system
RCY	reactor critical-year
RD	rapid depressurization
RDS	radioactive drain system
RDT	reactor drain tank
REMP	Radiological and Environmental Monitoring Program
RFI	request for information
RFI	radio frequency interference
RG	Regulatory Guide
RHR	residual heat removal
RIA	reactivity-initiated accident
RIHA	risk-important human action
RLE	review level earthquake
RLS	radioactive laundry subsystem
RMI	reflective metal insulation

RMS	radiation monitoring system
RMTS	risk-managed technical specifications
RMWT	reactor makeup water tank
RO	reactor operator
RP	reactor protection
RPCB	reactor power cutback
RPCS	reactor power cutback system
RPS	reactor protection system
RPV	reactor pressure vessel
RRS	required response spectra
RSC	remote shutdown console
RSF	RCP seal LOCA
RSG	rapid ex-vessel steam generation
RSPT	reed switch position transmitter
RSR	remote shutdown room
RSSH	resin sluice supply header
RT	reactor trip
RTCB	reactor trip circuit breaker
RTD	resistance temperature detector
RTE	random turbulent excitation
RT_{NDT}	reference temperature for nil-ductility transition
RTNSS	regulatory treatment of non-safety systems
RTO	reactor trip override
RTOTT	reactor trip on turbine trip
RTP	1) rated thermal power 2) return to power
RT_{PTS}	reference temperature (pressurized thermal shock)
RTS	reactor trip system
RTSG	reactor trip switchgear
RTSS	reactor trip switchgear system
RV	reactor vessel

RVI	reactor vessel internals
RVLMS	reactor vessel level monitoring system
RVR	reactor vessel rupture
RVUH	reactor vessel upper head
RWP	radiation work permit
RWT	raw water tank
RY	reactor-year
S&Q	staffing and qualification
SAFDL	specified acceptable fuel design limit
SAM	seismic anchor movement
SAMA	severe accident mitigation alternative
SAMDA	severe accident mitigation design alternative
SAMG	severe accident management guideline
SAS	service air system
SAT	standby auxiliary transformer
SAW	submerged arc welding
SBCS	steam bypass control system
SBLOCA	small-break loss-of-coolant accident
SBO	station blackout
SC	1) shutdown cooling 2) safety console 3) safety critical
SCC	stress corrosion cracking
SCETCh	severe combined environment test chamber
SCP	shutdown cooling pump
SCS	shutdown cooling system
SCU	statistical combination of uncertainties
SDC	shutdown cooling
SDCHX	shutdown cooling heat exchanger
SDL	serial data link
SDM	shutdown margin

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

SL	surge line
SLB	steam line break
SLBFP	large steam line break during full-power operation
SLBZP	large steam line break during zero-power operation
SMA	seismic margin analysis
SMACNA	Sheet Metal and Air Conditioning Contractors' National Association
SMAW	shielded metal arc weld
SMS	seismic monitoring system
SODP	shutdown overview display panel
SOE	sequence of events
SOP	system operating procedure
SOV	solenoid-operated valve
SPADES+	safety parameter display and evaluation system+
SPAR-H	standardized plant analysis risk – human reliability
SPDS	safety parameter display system
SPERT	special power excursion reactor test program
SPM	success path monitoring
SPND	self-powered neutron detector
SPTA	standard post-trip action
SQSDS	seismic qualification summary data sheet
SRI	Stanford Research Institute
SRLST	Spent Resin Long-tern Storage Tank
SRM	1) Staff Requirements Memorandum 2) standard reference material
SRO	senior reactor operator
SRP	Standard Review Plan
SRS	solid radwaste system
SRSS	square root of the sum of the squares
SRST	spent resin storage tank
SRV	safety relief valve
SS	stainless steel

SSA	safety shutdown analysis
SSC	structure, system, or component
SSE	safe shutdown earthquake
SSI	soil-structure interaction
SSIE	supporting system initiating event
SSM	saturation margin monitor
SSS	secondary sampling system
SSW	secondary shield wall
ST	stud tensioner
STA	shift technical advisor
STC	source term category
STP	standard temperature & pressure
SV	suitability verification
SWGR	switchgear
SWMS	solid waste management system
SWYD	switchyard
SX	essential service water system
SY	systems analysis
T&M	test and maintenance
T/C	reactor inlet temperature, T(cold)
T/G	turbine-generator
T/H	reactor outlet temperature, T(hot)
TA	task analysis
TAA	transient and accident analysis
T_{AVG}	average temperature
TB	turbine building
TBS	turbine bypass system
TBV	turbine bypass valve
TCB	trip circuit breaker
TCE	two-cell equilibrium
T_{COLD}	cold leg temperature

TCS	turbine control system
TDAFWP	turbine-driven auxiliary feedwater pump
TDH	total dynamic head
TDR	time domain reflectometry
TEDE	total effective dose equivalent
TEMA	Tubular Exchanger Manufacturers Association
TEPCO	Tokyo Electric Power Company
TG	turbine generator
TGB	turbine generator building
TGBCCW	turbine generator building closed cooling water
TGBOCWS	turbine generator building open cooling water system
TGCS	turbine generator control system
TGSS	turbine gland sealing system
THD	total harmonic distortion
THERP	technique for human error rate prediction
TI	temperature indicator
TI-SGTR	temperature-induced steam generator tube rupture
TID	1) technical information document 2) total integrated dose
TIHA	treatment of important human action
TIV	temperature isolation valve
TLOCCW	total loss of component cooling water
TLOESW	total loss of essential service water
TLOFW	total loss of feedwater
TMI	Three Mile Island
T_{NDT}	nil-ductility transition temperature
TO	turbine operator
TRAN	transient
T_{REF}	reference temperature
TRS	test response spectrum
TS	technical specification

TSC	technical support center
TSO	transmission system operator
TSP	1) tri-sodium phosphate 2) transmission system provider
TSSS	turbine steam seal system
TT	thermally treated
UAT	unit auxiliary transformer
UEL	upper electrical limit
UGS	upper guide structure
UHS	ultimate heat sink
UHSRS	ultimate heat sink related structure
UL	Underwriters Laboratories
UPC	ultimate pressure capacity
UPS	uninterruptible power supply
URS	uniform response spectrum
USE	upper-shelf energy
USEPA	U.S. Environmental Protection Agency
USH	uniform support motion
USI	unresolved safety issue
UV	undervoltage
V&V	verification and validation
Vac	voltage alternating current
VB	vessel breach
VBPSS	vital bus power supply system
VCT	volume control tank
VD	emergency diesel generator area HVAC
Vdc	voltage direct current
VDU	visual display unit
VEWFDS	very early warning fire detection system
VFTP	ventilation filter testing program
VG	ESW intake structure/CCW heat exchanger building HVAC

VIPER	Vibration Investigation and Pressure Drop Experimental Research
VK	auxiliary building controlled area HVAC
VO	auxiliary building clean area HVAC
VOPT	variable overpower trip(signal)
VPN	virtual private network
VSP	variable setpoint
VU	miscellaneous building HVAC
VWO	valve wide open
WCT	waste collection tank
WDT	watchdog timer
WLS	wet layup subsystem
WO	chilled water system
WPS	welding procedure specification
WR	wide range
WRC	Welding Research Council
WT	turbine generator building closed cooling water system
WWTF	waste water treatment facility
WWTS	wastewater treatment system
ZD	zero dependence (HRA)
ZOI	zone of influence
ZPA	zero-period acceleration

CHAPTER 1 – INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

1.1 Introduction

Korea Hydro & Nuclear Power Co., Ltd. (KHNP) has designed the APR1400, an evolutionary light water reactor (LWR). Korea Electric Power Corporation (KEPCO) and KHNP submit the Design Control Document (DCD) of the APR1400 design for U.S. Nuclear Regulatory Commission (NRC) review and approval under the provisions of 10 CFR Part 52. KEPCO and KHNP request the issuance of a standard design certification for the APR1400 in accordance with 10 CFR Part 52, Subpart B. This DCD and application for design certification are based on KHNP design experience and the ABB-CE System 80+ certified design.

1.1.1 Plant Location

The APR1400 is designed for use at a site with the parameters that are described in Chapter 2 of this DCD. The combined license (COL) applicant that references the APR1400 design certification is to identify the actual plant site location.

1.1.2 Containment Type

The APR1400 containment is a steel-lined prestressed concrete structure that consists of a right circular cylinder with a hemispherical dome on a reinforced concrete common basemat. There is no structural connection between the free-standing portion of the containment and adjacent structures other than penetrations and associated supports. The containment retains integrity at the pressure and temperature conditions associated with the most limiting design basis event (DBE) without exceeding the design leakage rate. Access to the containment is provided by personnel air locks and an equipment hatch. Penetrations are also provided for electrical and mechanical components and for the transport of nuclear fuel.

1.1.3 Reactor Type

The APR1400 nuclear steam supply system (NSSS) is a KHNP-designed evolutionary two-loop pressurized water reactor (PWR).

1.1.4 Power Output

The APR1400 net electrical power output is approximately 1,400 MWe, depending on site conditions. The NSSS rated thermal power is 4,000 MWt with a core thermal power of 3,983 MWt.

1.1.5 Schedule

The COL applicant that references the APR1400 is to provide estimated schedules for the completion of construction and the start of commercial operation.

1.1.6 Format and Content

1.1.6.1 NRC Regulatory Guide 1.206

The format and content of this DCD are based on the guidance in NRC Regulatory Guide (RG) 1.206 (Reference 1). To the extent practical, the chapter, section, subsection, and paragraph headings in the APR1400 DCD are consistent with NRC RG 1.206.

1.1.6.2 Standard Review Plan

Preparation of this DCD has followed the guidance in the NRC Standard Review Plan (SRP), NUREG-0800 (Reference 2). An evaluation of the conformance with the SRP is provided in Section 1.9.

1.1.6.3 Text, Tables, and Figures

Tables and figures are identified by section or subsection number followed by a sequential number (for example, Table 1.3-1 is the first table of Section 1.3). Tables and figures are placed at the end of the applicable sections immediately following the text. Figures include drawings, graphs, and photographs.

1.1.6.4 Page Numbering

Pages are numbered sequentially in each section and are identified by the section number followed by a sequential number, and at the beginning of each chapter.

1.1.6.5 Proprietary Information

This document includes no information that is proprietary to KHNP. The portions of this document that are classified as sensitive and will be withheld from the public according to 10 CFR 2.390 are indicated and provided separately to the NRC.

As noted in Section 1.6, the DCD Tier 2 references topical and technical reports that contain proprietary information. In these cases, in Tables 1.6-1 and 1.6-2, the non-proprietary version of the topical or technical report is also identified.

1.1.6.6 Acronyms and Abbreviations

The acronyms and abbreviations used in this DCD are provided after the list of figures at the beginning of this chapter.

1.1.6.7 Amendments

The APR1400 DCD will be amended, if necessary, as the APR1400 design is finalized. The DCD will also be amended as a result of the NRC review. To identify the amendments, the following guidelines will be followed:

a. Amended portions will be indicated by vertical lines on the right hand side of the page. The vertical lines will identify only the latest amendment (i.e., amendments since the previous revision number).

b. Figure changes will be indicated by vertical lines on the right hand side of the page. Vertical lines will identify only the latest amendment.

c. Responses to NRC questions will be incorporated, as applicable, into revisions of the DCD.

1.1.7 Combined License Information

COL 1.1(1) The COL applicant that references the APR1400 is to identify the actual plant site location.

COL 1.1(2) The COL applicant that references the APR1400 is to provide estimated schedules for the completion of construction and the start of commercial operation.

1.1.8 References

1. Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," U.S. Nuclear Regulatory Commission, June 2007.

2. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," U.S. Nuclear Regulatory Commission, various dates and revisions.

1.2 General Plant Description

This section contains a summary of the principal design criteria, operating characteristics, safety considerations, and major structures and systems. This section also includes a site plan and the general arrangement of major structures and equipment. The scope of the certified design is described in Section 1.2.14 and is shown on the site plan in Figure 1.2-1. The site plan also shows site-specific structures and features.

The combined license (COL) applicant is to prepare a complete and detailed site plan.

1.2.1 Principal Design Criteria, Operating Characteristics, and Safety Considerations

1.2.1.1 Principal Design Objectives

The following subsection provides the principal design objectives for the safety, reliability, and performance of the plant. These objectives are the basis of the principal design criteria for the APR1400.

1.2.1.1.1 Safety Design Objectives

The safety design objectives of the APR1400 are as follows:

a. Simplify plant design and operation, as described in Subsection 1.2.1.2.1.

b. Provide the proper safety margin for a more forgiving and resilient plant, as described in Subsection 1.2.1.2.2.

c. Improve the human-system interface system to promote error-free normal operations and quick, accurate diagnosis of off-normal conditions.

d. Meet applicable NRC requirements related to engineered safety system design and analysis of plant and engineered safety system responses to regulatory transients and accidents.

e. Evaluate the mean annual core damage frequency (CDF) and large release frequency (LRF) for the APR1400 design using a probabilistic risk assessment (PRA). The design target for CDF is 1×10^{-5} events per reactor year, and the design target for LRF is 1×10^{-6} events per reactor year. These targets include an assessment of internal and external events, excluding seismic events, sabotage, and other external events, and an assessment of shutdown events.

f. Provide a large, rugged reactor containment building and associated containment systems for heat removal and retention of fission products for design basis events (DBEs) and beyond DBEs (BDBEs). Containment design pressure is based on the most limiting loss of coolant or steam line break accident.

g. Provide containment system components for which a change of state is necessary (e.g., containment isolation valves) that are redundant and sufficiently independent from the systems whose failure could lead to core damage in order to provide reasonable assurance of an intact containment and avoid significant vulnerability to common cause failure.

h. Design the containment systems so the applicable exposure limits are met assuming a reactor containment building design leak rate of no less than 0.1 volume percent per day.

i. Provide at least two separate and independent ac power connections to the grid to decrease the likelihood of a loss of offsite power (LOOP).

j. Reduce the risk of a station blackout (SBO) by providing an independent, safety-related, onsite ac power generation source for each division and by providing a non-safety-related, alternate ac (AAC) onsite power source.

k. Provide adequate severe accident protection through conservatisms inherent in the design and additional plant features that limit direct containment heating, provide reasonable assurance of core debris coolability, and avoid detonable concentrations of hydrogen.

1.2.1.1.2 Performance Design Objectives

The performance design objectives of the APR1400 are as follows:

a. Provide a lifespan of 60 years without the need for an extended refurbishment outage.

b. Provide the capability of operating on a fuel cycle, from post-refueling startup to the subsequent post-refueling startup, with a refueling interval of 18 months.

1.2.1.2 Fundamental Design Approach

The following subsections describe the fundamental design approaches that were used as the basis for the development of a comprehensive set of technical requirements for the APR1400 design.

1.2.1.2.1 Simplification

The approach to the APR1400 design emphasizes simplicity in all aspects of the plant design, construction, and operation. Simplicity is accomplished by pursuing simplification opportunities with high priority and placing greater importance on simplification in design decisions than has traditionally been done.

The APR1400 simplification approaches include the following:

a. Use a minimum number of systems, valves, pumps, instruments, and other types of mechanical and electrical equipment that are consistent with essential functional requirements.

b. Provide a human-system interface that simplifies plant operation and reflects operator needs and capabilities.

c. Provide system and component designs that provide reasonable assurance that the final plant design minimizes demands on the operator during normal operation as well as transient and emergency conditions.

d. Design equipment and arrangements that simplify and facilitate maintenance.

e. Provide protective logic and actuation systems that are more simplified than those in existing plants.

f. Use standardized components to facilitate operations and maintenance.

g. Design for ease and simplification of construction.

1.2.1.2.2 Design Margin

The APR1400 design approach includes a consideration of the proper margin that is needed to provide reasonable assurance of plant safety and operability, as follows:

a. Designed capability to accommodate transients without causing initiation of engineered safety systems

b. Ample operator time to assess and deal with upset conditions with minimum potential for damage

c. Enhancement of system and component reliability and minimization of the potential of exceeding limiting conditions for operation (LCO) limits that could cause derating or shutdown

1.2.1.2.3 Safety

The APR1400 safety design approach is that there will be excellence in safety to provide reasonable assurance of safety for the general public and personnel. The primary emphasis is on accident prevention, which includes accident resistance and core damage prevention. Emphasis is also placed on mitigation of the consequence of potential accidents so that a balanced approach to safety is achieved.

This design approach of excellence in safety is implemented through an integrated approach that includes three overlapping levels of safety protection-accident resistance, core damage prevention, and mitigation-and therefore, uses a deterministic analysis framework complemented by PRA.

1.2.1.2.4 Proven Technology

The APR1400 design approach uses successful, proven technology throughout the plant, including the design of systems and components, maintainability and operability features, and construction techniques.

1.2.2 Principal Site Characteristics

The APR1400 is a standard nuclear power plant design that can be constructed on a site with the parameters that are described in Chapter 2. These parameters are the basis for design certification. The site interface parameters presented in Chapter 2 are conservative enough to envelop most potential sites in the United States.

1.2.3 Nuclear Steam Supply System Summary

The scope of the APR1400 design covers an essentially complete nuclear power plant that includes all structures, systems, and components (SSCs) that can significantly affect safe operation. The primary design characteristics are summarized in the following subsections. The seismic category, safety classification, and quality assurance requirements of SSCs are listed in Table 3.2-1.

1.2.3.1 Reactor

1.2.3.1.1 Reactor Core

The reactor core is fueled by uranium dioxide pellets enclosed in fuel rods. The fuel rods are fabricated into assemblies with nozzles that limit axial motion and grids that limit lateral motion of the fuel rods. The control element assemblies (CEAs) consist of boron carbide (B_4C) or Inconel absorber rods that are guided by tubes located within the fuel assembly. The core consists of 241 fuel assemblies that are typically loaded in the first fuel cycles with different U-235 enrichments. The NSSS-rated thermal output is 4,000

MWt with a core thermal output of 3,983 MWt. The reactor core is described in Sections 4.2, 4.3, and 4.4.

1.2.3.1.2 Reactor Internals

The reactor internals include the core support barrel, lower support structure and in-core instrumentation nozzle assembly, core shroud, and upper guide structure assembly. The core support barrel is a right circular cylinder supported by a ring flange from a ledge on the reactor vessel. The lower support structure transmits the entire weight of the core to the core support barrel by means of a beam structure. Snubbers are provided at the lower end of the core support barrel to restrict lateral and torsional movement. The core shroud surrounds the core and minimizes the amount of bypass flow. The upper guide structure provides a flow shroud for the CEAs and limits upward motion of the fuel assemblies.

The principal design bases for the reactor internals are to provide vertical supports and horizontal restraints during all normal operating, upset, emergency, and faulted conditions. The core is supported and restrained during normal operation and postulated accidents to provide reasonable assurance that coolant can be supplied to the coolant channels for heat removal. The reactor internals are described in further detail in Sections 3.9 and 4.5.

1.2.3.2 Reactor Coolant System and Connecting System

1.2.3.2.1 Reactor Coolant System

The reactor coolant system (RCS) is arranged as two closed loops connected in parallel to the reactor vessel. Each loop consists of one outlet hot leg, one steam generator (SG), two cold legs, and two reactor coolant pumps (RCPs). A pressurizer (PZR) is connected to one of the RCS loops.

The RCS operates at a nominal pressure of 158.2 kg/cm^{2}A (2,250 psia). The reactor coolant enters the reactor vessel, flows downward between the reactor vessel shell and core barrel, flows up through the core, leaves the reactor vessel, and flows through the tube side of the two SGs where heat is transferred to the secondary system. RCPs return the reactor coolant to the reactor vessel.

Two SGs, using heat generated by the reactor core, produce steam for driving the plant turbine generator (T/G). Each SG is a vertical U-tube heat exchanger with an integral economizer that operates with the reactor coolant on the tube side and secondary coolant on the shellside. Each unit is designed to transfer heat from the RCS to the secondary system to produce saturated steam when provided with the proper feedwater (FW) input.

Moisture separators and steam dryers on the shellside of the SG limit the moisture content of the steam during normal operation. An integral flow restrictor is included in each SG nozzle to restrict flow in the event of a steam line break.

The SG incorporates high-performance steam dryers to limit the moisture content to below 0.25 percent in the steam flow. The heat transfer tubes are made of Alloy 690 TT, which is resistant to stress corrosion cracking in high-temperature conditions. The secondary FW inventory is increased to extend the dry-out time to enhance the NSSS capability to tolerate upset conditions and improve operational flexibility. The heat transfer area is large enough to allow the NSSS to maintain a rated output even if 10 percent of the tubes are plugged.

The RCS is described in further detail in Chapter 5.

1.2.3.2.2 Reactor Coolant System High Point Vents

The high point vent system is a dedicated safety system designed to perform the following functions:

a. A safety-grade means of venting non-condensable gases and steam from the PZR and the reactor vessel closure head.

b. A safety-grade means to depressurize the RCS in the event the PZR spray is unavailable during plant cooldown to cold shutdown.

The reactor coolant gas vent system (RCGVS) is described in further detail in Subsection 5.4.12.

1.2.4 Engineered Safety Features

Engineered safety features (ESF) are provided to mitigate the consequences of design basis accidents. These ESFs are designed to localize, control, mitigate, or terminate such accidents in order to hold exposure levels below the limits of 10 CFR 50.34 (Reference 1).

1.2.4.1 Reactor Containment Building

General arrangements for the reactor containment building are shown in Subsection 1.2.14.

The APR1400 reactor containment building is a steel-lined prestressed concrete structure that consists of a right circular cylinder with a hemispherical dome on a reinforced concrete basemat. The cylindrical portion of the containment structure is prestressed by a post-tensioning system that consists of horizontal (hoop) and vertical (inverted, U-shaped) tendons. The interior surfaces of the containment shell, dome, and basemat are lined with a carbon steel plate for leak-tightness. A protective layer of concrete (filled slab) covers the portion of the liner over the foundation slab. The containment building provides biological shielding for normal and accident conditions.

The containment building completely encloses the reactor and RCS, and is designed to provide a barrier that is essentially leak-tight to the release of radioactive materials subsequent to postulated accidents. The internal structures and compartment arrangement provide equipment missile protection and biological shielding for maintenance personnel.

The containment building is designed for all credible loading combinations, including normal loads during a LOCA, test loads, and loads due to adverse environmental conditions.

1.2.4.2 Safety Injection System

The safety injection system (SIS) is designed to satisfy NRC requirements. The requirements are specified as the licensing design basis for the APR1400 design.

In the unlikely event of a LOCA, the SIS injects borated water into the RCS. The SIS incorporates a four-train safety injection configuration and an in-containment refueling water storage tank (IRWST).

The SIS uses four safety injection (SI) pumps to inject borated water directly into the reactor vessel. In addition, four safety injection tanks (SITs) are provided. The SI pumps are aligned to the IRWST, and realignment for recirculation following a LOCA is not required. The SIT provides cooling to limit core damage and fission product release and reasonable assurance of an adequate shutdown margin. The fluidic device (FD) in the SIT regulates the flow rate into the reactor vessel to improve cooling effectiveness.

The SIS also provides continuous long-term, post-accident cooling of the core by recirculating borated water from the IRWST. Water drawn from the IRWST by the SI pumps and containment spray (CS) pumps is injected into the reactor vessel and containment. The SI water then enters the containment through the primary pipe break. This water and the CS water return through floor drains and the holdup volume tank (HVT) to the IRWST. During this process, heat is removed from the IRWST water by the CS heat exchanger.

The SIS is capable of providing an alternate means of decay heat removal for the events beyond the licensing design basis in which the SGs are not available. Decay heat removal is accomplished by feeding and bleeding the RCS, using the SIS to feed and the pressurizer pilot operated safety relief valve (POSRV) to bleed, and by cooling the IRWST water using the shutdown cooling system (SCS).

The SIS and the IRWST are described in further detail in Sections 6.3 and 6.8, respectively.

1.2.4.3 Auxiliary Feedwater System

The auxiliary feedwater system (AFWS) provides feedwater from the auxiliary feedwater storage tanks (AFWSTs) to the SGs for heat removal when the FW system is inoperable for a transient or postulated accident condition.

The AFWS consists of two 100 percent capacity motor-driven pumps, two 100 percent capacity turbine driven pumps, two 100 percent AFWSTs, valves, two cavitating flow-limiting venturis, and instrumentation. Each pump takes suction from the respective AFWST and has a respective discharge header. Each pump discharge header contains a pump discharge check valve, flow modulating valve, auxiliary feedwater (AFW) isolation valve, and SG isolation check valve.

The AFWS components are located in seismic Category I structures, which protect the components from external environmental hazards such as earthquakes, tornados, floods, and external missiles. Each train of the AFWS is physically separated from the others within these structures.

One motor-driven pump train and one turbine-driven pump train are configured into one mechanical division and joined inside the containment to feed their respective SG through a common AFW header, which connects to the steam generator downcomer feedwater line. Each common AFW header contains a cavitating venturi to restrict the maximum AFW flow rate to each SG.

The AFWS is designed to be manually or automatically actuated by an auxiliary feedwater actuation signal (AFAS). At the low water level setpoint of the SG, the AFAS associated with that SG is designed to actuate the AFWS.

For design basis considerations, sufficient feedwater can be provided at the required temperature and pressure even if a secondary pipe break event occurs. Because the AFWS is the only safety-related source of makeup water to the SGs for heat removal when the FW system is inoperable for a transient or postulated accident condition, it has been designed with redundancy, diversity, and separation to provide reasonable assurance of its ability to function.

The AFWS is described further in Subsection 10.4.9.

1.2.4.4 Containment Spray System

The containment spray system (CSS) is designed to maintain containment pressure and temperature within the design limits in the unlikely design basis accidents (DBAs) that result in mass-energy releases to the containment atmosphere. The CSS also provides a containment air cleanup function to reduce the concentration of fission products in the containment atmosphere after an accident.

The CSS consists of two independent trains, each containing a CS pump, a CS heat exchanger, a CS pump mini-flow heat exchanger, spray headers, spray rings and nozzles, and associated valves, piping, and instrumentation.

The CS pumps are automatically actuated on receipt of a safety injection actuation signal (SIAS) or a containment spray actuation signal (CSAS). Upon a CSAS, the containment spray isolation valves open and the CS flow starts. The essential components of the CSS are powered from the emergency power sources to provide assurance of the reliability of the safety function for a loss of offsite power. The suction isolation valve from the IRWST is locked open during power operation. Two CS pumps supply water through two CS heat exchangers to the spray headers in the upper region of the containment. Spray headers are used to provide a relatively uniform distribution of spray over the cross-sectional area of the containment. The IRWST provides a continuous suction source for the CS pumps, thus eliminating the switchover from the IRWST to the containment recirculation sump for conventional PWR plants.

The CS pumps can be manually aligned and used as residual heat removal pumps during SCS operation. Likewise, the SC pumps can be manually aligned to perform the containment spray function.

The CS pumps can also be used as a backup to the SC pumps to provide cooling of the IRWST water during post-accident feed and bleed operations when the SGs are not available to cool the RCS.

The CSS is further discussed in Subsections 6.2.2 and 6.5.2.

1.2.4.5 Containment Isolation System

The containment isolation system (CIS) provides means of isolating fluid systems that pass through the containment penetrations to confine the release of any radioactivity from the containment following a postulated DBA.

In conformance to 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 54 (Reference 2), the piping systems and related components penetrating the containment are provided with leak detection, isolation, and containment capabilities with redundancy, reliability, and performance capabilities that reflect the safety-related importance of isolating these fluid systems.

Isolation design is achieved by applying acceptable common criteria to penetrations in many different fluid systems and by using containment pressure to provide a containment isolation actuation signal (CIAS) to actuate appropriate valves.

The CIS is described further in Subsection 6.2.4.

1.2.4.6 Engineered Safety Features Filter Systems

ESF filters are provided for the systems that are required to perform safety-related functions subsequent to a DBA, as follows:

a. Control room emergency makeup air cleaning system

   The system is part of the control room heating, ventilation, and air conditioning (HVAC) system and is used to clean up the makeup air that has potential to carry radioactive iodine and particulates following a DBA.

   The system is normally shut down and starts automatically in response to any one of the following signals:

   1) SIAS

   2) Control room emergency ventilation actuation signal (CREVAS)

   3) Remote manual activation from the main control room (MCR)

b. Auxiliary building controlled area emergency exhaust system

   The system is part of the auxiliary building controlled area HVAC system and is used to filter radioactive elemental iodines and particulates in the exhaust air from the safety-related mechanical equipment rooms, which are cooled by safety-related cubicle coolers after a DBA.

   The system is normally shut down and starts automatically in response to any one of the following signals:

1) SIAS

2) Remote manual activation from the MCR

c. Fuel handling area emergency exhaust system

The system is part of the fuel handling area HVAC system and is used to reduce the radioactive elemental iodines and particulates in the exhaust air from the fuel handling area following a fuel handling accident.

The system is normally shut down and starts automatically in response to one of the following signals:

1) High radiation signal from the radiation monitor located in the common discharge duct of the fuel handling area exhaust air cleaning units (ACUs)

2) Fuel handling area emergency ventilation action signal (FHEVAS)

3) Remote manual activation from the MCR

ESF filter systems are described further in Subsection 6.5.1.

1.2.5 Instrumentation and Control

1.2.5.1 Reactor Trip System

The reactor trip system (RTS) is a safety system that initiates reactor trips. The RTS consists of four channels of sensors, auxiliary process cabinet-safety (APC-S), ex-core neutron flux monitoring system (ENFMS), core protection calculator system (CPCS), the reactor protection system (RPS) portion of the plant protection system (PPS), and reactor trip switchgear system (RTSS).

Four independent channels of the RPS monitor the selected plant parameters. The RPS logic is designed to initiate protective action whenever the signals of any two channels of a given parameter reach the setpoint. If this occurs, the power supplied to the control element drive mechanisms (CEDMs) is interrupted through the RTSS. The CEDMs release the CEAs, which drop into the core to shut down the reactor.

1.2.5.2 Engineered Safety Features System

The engineered safety features (ESF) system consists of four channels of sensors, APC-S, the engineered safety features actuation system (ESFAS) portion of PPS, and the engineered safety features – component control system (ESF-CCS).

The ESF-CCS accepts ESFAS initiation signals from the ESFAS portion of the PPS and radiation monitoring system (RMS). The ESF actuation logic is used to activate ESF system components of the plant. Emergency diesel generator (EDG) loading sequencer logic is also included in the ESF-CCS. The component control logic in the ESF-CCS is described in Subsection 1.2.5.3. The ESF actuation logic has a selective 2-out-of-4 coincidence logic for the NSSS ESFAS or 1-out-of-2 logic for the BOP ESFAS so that no single failure can preclude the system from providing the safety function. The ESF actuation signal actuates ESF system components through the ESF-CCS.

1.2.5.3 Component Control System

The component control system (CCS) is designed to provide control of plant process components and to acquire data on the process components. The CCS provides discrete and continuous control of plant components.

The CCS consists of the ESF-CCS and process-CCS (P-CCS) assemblies to provide control for the different divisions of safety equipment, as well as non-safety equipment. Although the safety and non-safety CCS assemblies perform different plant control functions, they use diverse software and software-dependent electronic components.

1.2.5.4 Diverse Protection System

The diverse protection system (DPS) augments the plant protection function by initiating a reactor trip signal, turbine trip signal, AFAS, and SIAS that are separate and diverse from the PPS.

The DPS is provided to address the design requirements of 10 CFR 50.62 (Reference 3) and the Staff Requirements Memorandum (SRM) regarding SECY-93-087, II.Q (Reference 4). The DPS equipment provides a simple and diverse mechanism to significantly decrease risk from anticipated transient without scram (ATWS) events and assist the mitigation of the effects of a postulated common-cause failure (CCF) of the digital computer logic within the PPS and ESF-CCS.

The DPS initiates a reactor trip when the PZR or containment pressure exceeds a predetermined value. For implementation of the reactor trip function, the DPS circuitry is diverse from the PPS, from sensor output to interruption of power to control rods. The DPS design uses a 2-out-of-4 logic to open trip circuit breakers of the reactor trip switchgear system (RTSS).

The DPS initiates the AFAS when the level in either SG decreases below a predetermined value and initiates the SIAS when the PZR pressure decreases below a predetermined value. From sensor output to, but not including, the final actuation device, the DPS circuitry for the AFAS and SIAS is independent and diverse from the circuitry of the PPS and ESF-CCS.

1.2.5.5 Reactor Control Systems

The startup, operation, and shutdown of the reactor are accomplished through integrated control system actions. These control systems regulate reactor power and respond to plant transients to maintain the NSSS within its normal operating conditions. Reactor control functions are performed by the power control system (PCS) and NSSS process control system (NPCS) of the P-CCS, as described in Section 7.7. The PCS performs digital rod control system (DRCS), reactor power cutback system (RPCS), and required response spectra functions to adjust the reactor power response to turbine load demand. The NPCS performs steam bypass control system (SBCS), feedwater control system (FWCS), and PZR control functions.

Reactor power control is normally accomplished by the automatic movement of CEAs in response to a change in reactor coolant temperature, with manual control that is capable of overriding the automatic signal at any time. If the reactor coolant temperature is different from a programmed value, the CEAs are adjusted until the difference is within the prescribed control band. Regulation of the reactor coolant temperature, in accordance with this process, maintains the secondary steam pressure within operating limits and matches reactor power to load demand.

The reactor is controlled by a combination of CEA motion and dissolved boric acid in the reactor coolant. Boric acid is used for reactivity changes associated with large but gradual changes in water temperature, xenon concentration, and fuel burnup. The addition of boric acid also provides an increased shutdown margin during the initial fuel loading and subsequent refuelings. The boric acid solution is prepared and stored at a temperature that prevents precipitation.

CEA movement provides changes in reactivity for shutdown or power changes. The CEAs are moved by CEDMs mounted on the reactor vessel head. The CEDMs are designed to permit rapid insertion of the CEAs into the reactor core by gravity. CEA motion can be initiated manually or automatically.

The pressure in the RCS is controlled by regulating the temperature of the coolant in the PZR where steam and water are maintained in thermal equilibrium. Steam is formed by the PZR heaters or condensed by the PZR spray to reduce variations caused by expansion and contraction of the reactor coolant because of temperature changes.

The SBCS is used to dump steam in case of a large mismatch between the power being produced by the reactor and the power being used by the turbine. Dumping steam allows the reactor to remain at power instead of tripping. The water level in each SG is maintained by the FWCS. The RPCS is used to drop selected CEAs into the core to reduce reactor power rapidly during the large loss of load or failure of 2-out-of-3 operating FW pumps. Dropping selected CEAs allows the SBCS and the FWCS to maintain the NSSS in a stable condition without a reactor trip and without lifting any safety valves during the transients after the loss of load.

1.2.5.6 Nuclear Instrumentation

The nuclear instrumentation includes ex-core and in-core neutron flux detectors and associated signal processing equipment. Eight channels of ex-core instrumentation monitor the power. Two startup channels are provided for startup, two control channels are provided for power control, and four safety channels are provided for protection.

The startup channels are used to monitor the power that is used during the initial reactor startup, extended shutdown periods, startup after extended periods of shutdown, and after refueling operations. The control channels are used to control the reactor power during power operation. The safety channels are used to provide inputs to the variable overpower, high logarithmic power, low departure from nucleate boiling ratio (DNBR), and high local power density (LPD) trips in the RPS.

The in-core nuclear instrumentation consists of fixed in-core nuclear instrumentation detectors distributed throughout the core. The instrumentation is used to monitor the power distribution in the core and evaluate fuel burnup in each fuel assembly and thermal margins in the core.

1.2.5.7 Process Monitoring Systems

Temperature, pressure, flow, and liquid level are monitored as required to keep operating personnel informed of plant operating conditions. Protection channels indicate the various parameters used for protective action and provide trip and pre-trip alarms from the RPS.

Plant liquid and gaseous effluents are monitored to provide reasonable assurance that they are maintained within applicable radioactivity limits. Additional information is provided in Section 11.5.

1.2.6 Human-System Interface System

1.2.6.1 Main Control Room

The MCR is provided with a redundant, compact, workstation-type human-system interface (HSI); large display panel (LDP); safety console; voice communication equipment; and other equipment that is necessary for safe and reliable plant operation.

Qualified indication and alarm displays are provided to permit normal and accident plant operations in the unlikely event that the information processing system (IPS) becomes unavailable.

The arrangements and layouts for all controls and displays in the MCR are designed, verified, and validated in accordance with human factors design guidelines and the requirements in the APR1400 human factors engineering program plan, which is described in Section 18.1. The layout of an MCR is shown in Figure 7.7-13.

An operator workstation, including sufficient desk space, is provided to support the plant monitoring and daily operational needs for each operator.

1.2.6.2 Remote Shutdown Room

The RSR design includes the remote shutdown console (RSC), which is similar to the reactor operator (RO) workstation in the MCR, and the shutdown overview display panel (SODP) to achieve cold shutdown (Mode 5 plant conditions) when operators are to evacuate the MCR. The layout of the RSR is shown in Figure 7.4-4.

For a safe shutdown from the RSR, controls and indications are available through information on flat panel displays (FPDs) and soft controls on the RSC. The SODP provides information that the operator uses during plant shutdown operation.

For consistency, the information displays and soft controls on the RSC are the same as in the MCR.

1.2.6.3 Qualified Indication and Alarm System

The qualified indication and alarm system (QIAS) is composed of the qualified indication and alarm system – P (QIAS-P) and qualified indication and alarm system – non-safety (QIAS-N).

The QIAS-P provides a continuous and dedicated display of NRC RG 1.97 (Reference 5) Type B and C parameters for accident monitoring.

The QIAS-N receives analog and digital data from both safety and non-safety systems, analyzes the data, and relays the results of the analysis to the operator via FPDs and the mini-LDP in the safety console and the SODP in the RSR. The system interfaces with the IPS to integrate alarm and process information.

Additional information is provided in Section 7.5.

1.2.6.4 Information Processing System

The IPS is a fault-tolerant, multi-processor, computer-based system that provides plant data and status information to the operating staff. The IPS monitors the NSSS and balance-of-plant (BOP) steam and electrical production processes. The IPS provides plant operating staff the ability to obtain detailed process data via FPDs and LDPs.

The major functions performed by the IPS include plant-wide data acquisition through dedicated data links to plant systems, validation of sensed parameters, execution of application programs and performance calculations, monitoring of general plant status and plant safety status, generation of logs and reports, determination of alarm conditions, recording of the sequence of events, and generation of a post-trip review.

FPD and LDP formats incorporate human factors engineering design principles that permit quick operator recognition of information that is necessary to allow the operator to monitor, control, and diagnose plant conditions.

The IPS is designed to provide the plant operating staff with reliable, complete, and timely information for the safe and efficient operation of the plant. The IPS is designed to tolerate the loss of any single major system component without total loss of functionality. The design includes automatic fail-over and sufficient redundant peripherals to minimize the effects of an IPS component failure during plant operations.

The IPS is described further in Section 7.7.

1.2.7 Electrical System

Offsite and onsite power systems are provided to supply electrical power to unit auxiliaries that are necessary during normal operation and the RPS and ESF that are necessary in abnormal and accident conditions.

The offsite power system consists of transmission lines, transmission line towers, switchyard components and a control system, switchyard battery systems, transmission tie lines, main generator, generator circuit breaker (GCB), main transformer, unit auxiliary transformers (UATs), and standby auxiliary transformers (SATs).

Under normal operating conditions, the main generator supplies power through an isolated phase bus and the GCB to the main transformer and UATs. The UATs are connected to the isolated phase bus between the GCB and main transformer. Additional information on the offsite power system is provided in Section 8.2.

The onsite power system for the unit auxiliaries consists of four EDGs, an alternate alternating current (AAC) gas turbine generator (GTG), and two onsite power distribution systems (a Class 1E system and a non-Class 1E system). The onsite power distribution system is connected to the site-specific switchyard via two separate and independent transmission tie circuits. One circuit is connected to the switchyard through the main transformer and UATs, and the other circuit is connected to the switchyard via the SATs.

During normal operation, onsite power is supplied from the main generator through the UATs. During startup and shutdown, the GCB is open, and the onsite power is supplied from the transmission system through the main transformer and UATs.

The onsite power system is described further in Section 8.3. A description of the AAC generator system is provided in Section 8.4.

1.2.8 Steam and Power Conversion System

The function of the steam and power conversion system is to convert heat energy generated by the nuclear reactor into electrical energy. The heat energy produces steam in two SGs capable of driving a turbine-generator (T/G) unit.

The steam and power conversion system consists of the T/G, main steam system (MSS), condensate and FW system, and other support systems. The steam and power conversion system uses a condensing cycle with regenerative FW heating.

The steam generated in the two SGs is supplied to the high-pressure turbine by the MSS. The steam is expanded through the high-pressure turbine, passes through the two moisture separator reheaters (MSRs), and then flows to the three low-pressure turbines.

The exhaust steam from the low-pressure turbines is condensed in a conventional surface-type condenser. The condenser removes air and other non-condensable gases from the condensate and transfers heat to the circulating water system.

The condensate from the steam is returned to the SGs through the condensate and FW system. The condensate from the condenser hotwell is transferred through the low-pressure (LP) heaters to the deaerator storage tank by the condensate pumps.

The FW booster pumps take suction from the deaerator storage tank and discharge to the FW pumps. FW is discharged from the FW pumps, passes through two trains of high-pressure FW heaters, and is delivered to the SGs.

The steam and power conversion system is described further in Chapter 10.

1.2.8.1 Turbine Generator

The T/G converts the thermal energy of the steam produced in the SGs into mechanical shaft power and then into electrical energy.

The T/G consists of a double-flow, high-pressure turbine and three double-flow, low-pressure turbines driving a direct-coupled generator and two external MSRs.

The flow of main steam is directed from the SGs to the high-pressure turbine through main stop and control valves. After expanding through the high-pressure turbine, exhaust steam passes through the MSRs. Extraction from the high-pressure turbine and main steam from the equalization header is supplied to the first and second stage of reheater tube bundles in each reheater. The hot reheat steam is admitted to the low-pressure turbines through combined intermediate valves and expands through the low-pressure turbines to the main condensers.

The T/G control system is designed to be compatible with the plant control system for reactor operation. The T/G is designed to accept a sudden loss of full load or LOOP without exceeding design overspeed.

The T/G is described further in Section 10.2.

1.2.8.2 Main Steam System

The MSS delivers steam generated in the SGs to the high-pressure turbine where the thermal energy of the steam is converted to mechanical energy to drive the main T/G. The MSS also provides steam to the FW pump turbines, AFWP turbines, second-stage reheater of the MSRs, turbine steam seal system, auxiliary steam system, and process sampling system.

The major components of the MSS are the main steam piping, main steam isolation valves (MSIVs), main steam isolation valve bypass valves (MSIVBVs), main steam safety valves (MSSVs), main steam atmospheric dump valves (MSADVs), turbine bypass valves (TBVs), and AFWP turbine steam supply valves and warmup valves.

An MSIV is installed on each of the main steam lines downstream of the MSSVs, outside the reactor containment building. The MSIVs are provided to isolate the SGs upon receipt of a main steam isolation signal (MSIS). MSIVs are remote-operated and fail-closed valves with a hydraulic actuator.

Overpressure protection for the secondary side of the SGs is provided by spring-loaded MSSVs. Modulation of the TBVs would normally prevent the safety valves from opening.

Following the load rejection of any magnitude from full load to house load, including a turbine trip from 100 percent power, the TBS controls main steam pressure automatically by the SBCS.

During a turbine or reactor trip, the TBS dissipates heat from the reactor coolant system to the condensers. The system has the capability of relieving 55 percent of full load main steam flow to the main condenser.

The MSS is provided with MSADVs to remove reactor decay heat during hot standby and emergency cooldown in conjunction with AFWS.

The MSS is described in further detail in Section 10.3.

1.2.8.3 Condensate and Feedwater System

The condensate and feedwater system delivers feedwater from the condenser to the SG. The entire condensate system is non-safety related. The portions of the feedwater system that are required to mitigate the consequences of an accident and allow safe shutdown of the reactor are safety-related.

The condensate and feedwater system is described further in Subsection 10.4.7.

1.2.9 Heating, Ventilation, and Air Conditioning System

The HVAC systems for all plant buildings are designed for personnel comfort and equipment operation. In addition, the following systems are provided with the protection features described as follows:

a. The control room HVAC system is designed to maintain the environment in the control room envelope and limit the radiation exposure of personnel in the control room during all plant operation conditions. The system maintains positive pressure to provide habitability and prevent uncontrolled incoming air leakage.

b. The fuel handling area HVAC system is a once-through ventilation system designed to limit the radiation release following a fuel-handling accident to meet 10 CFR 50.34 guidelines. This system maintains the area under negative pressure and airflow from less-contaminated to more-contaminated areas.

c. The compound building controlled-area HVAC system is a once-through ventilation system with filtered exhausts. This system maintains negative building pressure and airflow from less-contaminated to more-contaminated areas.

d. The auxiliary building controlled-area HVAC system is a once-through ventilation system designed to filter post-accident contaminated leakages before exiting to meet 10 CFR 50.34 guidelines. This system maintains the building under negative pressure and airflow from less-contaminated to more-contaminated areas.

e. The containment purge system is provided with post-accident containment isolation features and filtration units for air cleanup during normal and refueling operations. This system limits the radiation release to meet 10 CFR 50.34 guidelines in case of a fuel handling accident inside the containment.

Other HVAC systems are described in Section 9.4.

1.2.10 Fuel Handling and Storage

1.2.10.1 Fuel Handling

Fuel handling equipment provides for the safe handling of fuel assemblies and CEAs under all specified conditions and for the required assembly, disassembly, and storage of the integrated head assembly and reactor internals during refueling.

The major components of the system are the refueling machine, CEA change platform, fuel transfer system, spent fuel handling machine, and new fuel and CEA elevators. The fuel handling equipment is provided to transfer new and spent fuel between the fuel storage facility, containment building, and fuel shipping and receiving areas during initial core loading and refueling operations. Fuel is inserted into or removed from the core using the refueling machine. During normal operations, irradiated fuel and CEAs are maintained in a water environment.

The principal design criteria specify the following:

a. Fuel is inserted, removed, and transported in a safe manner.

b. Subcriticality is maintained during all operations.

Fuel handling is described in further detail in Section 9.1.

1.2.10.2 Fuel Storage

The new fuel storage facility provides onsite storage capacity of 112 new fuel assemblies. This capacity, which represents 46 percent of the fuel assemblies in the core, envelops a reload batch based on a refueling cycle of 18 months. The spent fuel storage is divided into two regions of the fuel. The fresh or partially burnt fuel assemblies are stored in Region I, which has storage capacity for one full core, one refueling batch, and five damaged fuel assemblies. The Region I storage area is designed to accommodate fuel assemblies with an initial enrichment up to 5 weight percent U-235. Region II has a storage capacity of spent fuel assemblies generated during a plant operation of 20 years. The maximum initial enrichment of 5 weight percent U-235 and the minimum burnup are applied to the Region II design.

Criticality and safety analyses are addressed in Subsection 9.1.1. The new fuel and spent fuel storage facilities are addressed in Subsection 9.1.2.

1.2.11 Cooling Water Systems

1.2.11.1 Circulating Water System

The circulating water system provides cooling water for the condensers and T/G building closed cooling water heat exchangers and rejects heat to the normal plant heat sink. The normal plant heat sink is site-specific, but a mechanical draft cooling tower is used as a preferable heat sink.

The circulating water system is described further in Subsection 10.4.5.

1.2.11.2 Essential Service Water System

The essential service water system (ESWS) is an open system that takes suction from the ultimate heat sink (UHS) and provides cooling water to remove heat released from plant SSCs. The ESWS returns the heated water to the UHS. The ESWS cools the component cooling water system (CCWS), which in turn cools essential and non-essential reactor auxiliary loads.

The ESWS consists of two independent, redundant, safety-related divisions. Each division consists of two ESW pumps, three CCW heat exchangers, three ESW debris filters, and associated piping, valves, controls and instrumentation.

During normal operation, one ESW pump and two CCW heat exchangers per division are in service.

During plant shutdown operations, two ESW pumps and three CCW heat exchangers in each division operate to remove heat from the components required for plant shutdown operation.

During plant abnormal operation, one ESW pump and two CCW heat exchangers in a single division operate to remove heat from the essential components required for safe shutdown or mitigation of plant abnormal conditions.

In the event of a LOOP, each division of the ESWS is automatically powered from the EDGs in accordance with emergency load sequencing.

The ESWS is described further in Subsection 9.2.1.

1.2.11.3 Component Cooling Water System

The CCWS is a closed-loop cooling water system that, in conjunction with the ESWS and the UHS, removes heat generated from essential and non-essential plant components connected to the CCWS. Heat transferred by these components to the CCWS is rejected to the ESWS via the CCW heat exchangers.

The CCWS consists of two independent, redundant closed loop divisions. Each division consists of two CCW pumps, three CCW heat exchangers, a CCW surge tank, a CCW chemical addition tank, a CCW makeup pump, and associated piping, valves, and instruments.

During normal power operation, one CCW pump and two CCW heat exchangers in each division are in service to supply cooling water to safety-related and non-safety-related components in the division required for normal power operation.

During the plant shutdown operation, two CCW pumps and three CCW heat exchangers in each division operate to supply cooling water to the components required for plant shutdown operation.

During abnormal plant operations, one CCW pump and two CCW heat exchangers in a single division operate to supply cooling water to the essential components required for the safe shutdown of the plant or mitigation of the abnormal condition.

In the event of a LOOP, each division of the CCWS is automatically powered from the EDGs in accordance with emergency load sequencing.

The CCWS is described further in Subsection 9.2.2.

1.2.11.4 Chilled Water System

The chilled water system is designed to provide and distribute a sufficient quantity of chilled water, through a group of dedicated piping systems, to air handling units (AHUs) and cubicle coolers in specific plant areas. The system is divided into two subsystems: an essential chilled water system (ECWS) that serves safety-related HVAC cooling loads and a plant chilled water system (PCWS) that serves primarily non-safety-related HVAC cooling loads.

The chilled water system is described further in Subsection 9.2.7.

1.2.11.5 Spent Fuel Pool Cooling and Cleanup System

The spent fuel pool cooling and cleanup system (SFPCCS) consists of the spent fuel pool (SFP) cooling system and the SFP cleanup system.

The safety-related SFP cooling system consists of two redundant trains that are independent of each other. The SFP cooling system removes decay heat generated by one full core offloaded after 100 hours following shutdown, plus the spent fuel assemblies accumulated from the previous refueling operations. Spent fuel is placed in the pool during the refueling operation and stored there until shipped offsite. Heat is transferred from the SFP cooling system, through an SFP cooling heat exchanger, to the CCWS. When a cooling train is in operation, water flows from the SFP to the SFP cooling pump suction, is pumped through the hot side of the heat exchanger, and is returned to the SFP. The suction line is located at an elevation above the required minimum water level, while the return line contains an anti-siphon device to prevent gravity drainage of the SFP.

The non-safety-related SFP cleanup system consists of pumps, demineralizers, and filters to maintain SFP water clarity and purity. Fuel transfer canal and refueling pool water is circulated through the same demineralizers and filters. These cleanup loops are sufficient for removing fission products and other contaminants that may be introduced if a leaking fuel assembly is transferred to the SFP.

The demineralizer and filter of the cleanup train are used to clean and purify the SFP water or refueling water while SFP heat removal operations proceed. Connections are provided so that the water may be pumped from either the IRWST or the SFP through a filter and demineralizer and discharged to IRWST or the SFP. To assist further in maintaining SFP optical clarity, the SFP surface is cleaned by a skimmer.

The SFP receives borated makeup water from the boric acid storage tank (BAST) through the chemical and volume control system (CVCS). The seismic Category I backup source is provided from the AFWST via the CCW makeup pumps. The non-seismic Category source of nonborated demineralized water to the SFP is available during normal plant conditions.

The SFPCCS is described further in Subsection 9.1.3.

1.2.12 Auxiliary Systems

1.2.12.1 Shutdown Cooling System

The SCS is used to reduce the temperature of the reactor coolant, at a controlled rate, from the hot shutdown operating temperature to the refueling temperature and to maintain the proper reactor coolant temperature during refueling. The system uses SC pumps to circulate the reactor coolant through two SC heat exchangers and return it to the RCS. The CCWS supplies cooling water for the SC heat exchangers.

The SCS has a design pressure of 63.28 kg/cm^{2}G (900 psig). The system pressure provides for greater operational flexibility and simplifies concerns about system overpressurization.

The SCS is described in further detail in Subsection 5.4.7.

1.2.12.2 Chemical and Volume Control System

The CVCS controls the purity, volume, and boric acid content of the reactor coolant. The CVCS is not required for any safe shutdown or accident mitigation function.

The coolant purity level in the RCS is controlled by continuous purification of a bypass stream of reactor coolant. Water removed from the RCS is cooled in the regenerative heat exchanger and letdown heat exchanger.

From there, the coolant flows through a filter and a demineralizer where corrosion and fission products are removed. The coolant is then sprayed into the volume control tank (VCT) and returned by the charging pumps to the regenerative heat exchanger for heating prior to returning to the RCS loops. A portion of the flow downstream of the charging pump is diverted for RCP seal injection. The charging flow is controlled by centrifugal charging pumps and a charging flow control valve on the discharge of the pumps.

The CVCS automatically adjusts the amount of reactor coolant in order to maintain a programmed level in the PZR.

The CVCS controls the boric acid concentration in the coolant by “feed and bleed” where the purified letdown stream is diverted to a boron recovery subsystem, and either concentrated boric acid or demineralized water is sent to the charging pumps. The diverted coolant stream is processed by ion exchange and degasification and flows to a concentrator. The concentrator bottoms are sent to the BAST for reuse as boric acid solution, and the distillate is passed through an ion exchanger and stored for reuse as demineralized water in the reactor makeup water tank.

Moving accident mitigation and safe-shutdown functions to other dedicated safety systems has permitted simplification of plant systems. Although not a safety-related system, the CVCS could provide makeup and depressurization capabilities.

A CVCS is described in further detail in Subsection 9.3.4.

1.2.12.3 Primary Sampling System

The primary sampling system is designed to collect and deliver representative samples for inline and laboratory analyses. Typical results of the analyses include reactor coolant boron and chloride concentrations, fission product radioactivity level, radionuclide gamma-spectrum, dissolved gas concentrations, fission gas content, conductivity, pH, corrosion product concentration, and chemical additive concentration. The analysis results are used in regulating boron concentration, evaluating fuel element integrity and demineralizer performance, maintaining acceptable hydrogen levels, detecting radioactive material leakage, and regulating additions of corrosion-controlling chemicals to the systems.

The system consists of sampling lines, a normal primary sample sink, a normal primary sample cooler rack, post-accident primary sample cooler rack, post-accident primary sample sink, normal/post-accident primary sample control panels, primary off-gas hydrogen/oxygen analyzer, analysis equipment, and associated valves and instrumentation.

The system permits sampling during reactor operation, cooldown, and post-accident modes without requiring access to containment. Remote samples of fluids can be taken from high radiation areas without requiring access to these areas. Local sampling points are provided at various locations throughout the plant. Samples from the containment flow through containment isolation valves to the post-accident primary sample room in the auxiliary building or the normal primary sample room in the compound building. High-temperature sample lines also contain sample coolers in the normal and post-accident primary sample cooler racks.

The primary sampling system is described further in Subsection 9.3.2.

1.2.12.4 Condensate Polishing System

The condensate polishing system (CPS) is designed to remove dissolved and suspended impurities that could cause corrosion damage to secondary system equipment. Condensate polishing demineralizers are also used to remove impurities that enter the system as a result of a condenser circulating water tube leak.

The condensate polishing system is described further in Subsection 10.4.6.

1.2.12.5 Steam Generator Blowdown System

The SG blowdown system (SGBS) is designed to assist in maintaining the chemical characteristics of the secondary side water within permissible limits during normal operation and anticipated operational occurrences (AOOs) such as a main condenser tube leak or SG primary-to-secondary tube leakage. The SGBS is also designed to remove impurities concentrated in SGs by continuous blowdown (CBD), periodical high-capacity blowdown (HCBD), and emergency blowdown (EBD).

The SGBS consists of the blowdown subsystem (BDS) and wet layup subsystem (WLS). The BDS consists of blowdown piping connected to each SG, a blowdown flash tank, a regenerative heat exchanger, two pre-filters, two demineralizers, a post-filter, and control valves. The WLS consists of two recirculation trains (one for each SG) and shares filters and demineralizers with the BDS.

During normal operations, the CBD (0.2% or 1% of the full-power main steam flow) flows from each SG are maintained in order to keep SG the secondary side water chemistry within the specified limits.

The blowdown is directed into a flash tank where the flashed steam is returned to the cycle via the high-pressure FW heaters. The liquid portion flows to a heat exchanger for cooling and is directed through a blowdown filter where a major portion of the suspended solids is removed. After filtration, the blowdown fluid is processed by blowdown demineralizers and returned to the condenser. During long-term shutdown periods, the WLS is used to control water chemistry in the SGs. Following draining or dry layup, the WLS is used to refill the SGs.

The blowdown lines from the SGs are automatically isolated by closing isolation valves in the event of abnormal conditions.

The SGBS is described further in Subsection 10.4.8.

1.2.12.6 Compressed Air and Gas Systems

The compressed air and gas systems comprise the compressed air system, the compressed gas system, and the breathing air system. The compressed air and gas systems are non-safety related with the exception of containment penetration portion.

The instrument air system supplies clean, oil-free, dried air to all air-operated instrumentation and valves. The service air system supplies compressed air for air-operated tools, miscellaneous equipment, and various maintenance purposes.

The compressed gas system comprises the nitrogen subsystems, hydrogen subsystem, and carbon dioxide subsystem.

The breathing air system supplies emergency breathing air for control room personnel.

The compressed air and instrument air systems are described further in Subsection 9.3.1.

1.2.12.7 Equipment and Floor Drainage System

The equipment and floor drainage system (EFDS) provides the means by which wastes are appropriately segregated and transported to the liquid waste management system (LWMS) to minimize liquid and gaseous radioactive releases.

The EFDS is described further in Subsection 9.3.3.

1.2.12.8 Fire Protection Program

The fire protection program protects SSCs important to safety from the effects of a potential fire. The plant achieves safe shutdown with the assumption that fire will render all equipment in any one fire area inoperable, recognizing that postfire reentry for repairs or operator action will not be possible. The plant also maintains the ability to minimize the potential for radioactive releases to the environment in the event of a fire.

The fire protection program includes administrative controls, emergency lighting, fire barriers, fire detection and suppression systems, fire brigade personnel, and other features provided for fire protection purposes.

The fire protection program is described further in Subsection 9.5.1.

1.2.12.9 Communication Systems

The communication systems are designed to provide effective communications between all areas of the plant and the plant site, including all vital areas of the plant. In addition, the communication systems are designed to provide an effective means to communicate to plant personnel and offsite utility and regulatory officials during normal conditions, abnormal, and accident conditions.

The communication systems are described further in Subsection 9.5.2.

1.2.12.10 Lighting System

The lighting system is designed to provide adequate and effective illumination throughout the plant and plant site, including all vital areas of the plant.

The normal lighting system is used to provide normal illumination under normal plant operation, maintenance, and test conditions.

Upon loss of the normal lighting system, the emergency lighting system is used to provide acceptable levels of illumination throughout the station and particularly in areas where emergency operations are performed, such as control rooms, fuel handling area, remote shutdown area, and Class 1E switchgear rooms.

The lighting system is described further in Subsection 9.5.3.

1.2.12.11 Emergency Diesel Generator System

The EDG system is a safety-related system consisting of four EDGs and their respective support systems such as fuel oil, lube oil, engine cooling water, starting air, and combustion air intake and exhaust systems. Each EDG provides Class 1E power to one of the four independent Class 1E buses during a LOOP. EDGs are normally in standby mode.

Each EDG is designed to attain the rated voltage and frequency within 17 seconds of a loss of voltage, and to be connected to the 4.16 kV Class 1E bus within 19 seconds of a loss of voltage.

Once the EDG reaches rated voltage and speed, the EDG breaker closes and the sequencer generates the proper signal to connect ESF equipment to the Class 1E bus in a programmed time sequence.

The EDG support systems are described further in Subsections 9.5.4 through 9.5.8.

1.2.12.12 Gas Turbine Generator Facility

One GTG is used as an AAC source to cope with an SBO. The GTG is independent from the EDGs. The GTG manually starts from a standby condition, accelerates to the required speed, reaches nominal voltage and frequency, and is ready to accept load within 2 minutes of receipt of a start signal in the event of an SBO. The GTG is also designed to start automatically and to be connected manually to non-Class 1E cables in the event of a LOOP.

The major components of the GTG are a combustion turbine, generator, and auxiliary systems such as fuel oil, lube oil, start system, and combustion air intake and exhaust systems.

The GTG support systems are described further in Subsection 9.5.9.

1.2.12.13 Domestic Water and Sanitary System

The domestic water and sanitary system provides water for drinking and sanitary purposes.

The sanitary system is designed to receive and treat sewage. This system serves no safety functions and any malfunction has no adverse effect on any safety-related system. The requirements of 10 CFR Part 50, Appendix A, GDC 60 (Reference 6) are met as related to the design provisions provided to control the release of liquid effluents containing radioactive material from contaminating the domestic water and sanitary system.

The domestic water and sanitary system is described further in Subsection 9.2.4.

1.2.13 Radioactive Waste Management Systems

The radioactive waste management systems are designed to control radioactive liquid, gaseous, and solid wastes. The systems consist of three principal systems:

a. Liquid waste management system

b. Gaseous waste management system

c. Solid waste management system

The solid, gaseous, and liquid waste management systems are located in the compound building. The design of the radioactive waste management systems provides reasonable assurance that the total offsite dose resulting from radioactive releases is as low as is reasonably achievable (ALARA).

1.2.13.1 Liquid Waste Management System

The liquid waste management system (LWMS) is designed to monitor, control, collect, process, handle, store, and dispose of liquid radioactive waste generated during normal plant conditions, including AOOs. The LWMS is divided into the floor drain subsystem, equipment waste subsystem, chemical waste subsystem, and detergent waste subsystem. The LWMS treats liquid waste using a reverse osmosis (R/O) package system that reduces radioactivity to levels acceptable for release or reuse. The processed liquid radioactive waste is sampled prior to release from monitor tanks. The LWMS is designed to monitor radioactivity levels in the processed liquid waste prior to release.

The LWMS meets the following design requirements:

a. Capability to process floor drain wastes, equipment wastes, chemical wastes, and detergent wastes to meet release radionuclide concentration limits in accordance with 10 CFR Part 20, Appendix B (Reference 7), prior to discharge to the environment.

b. Capability to recycle treated water in order to minimize the liquid radwaste effluent releases to the environment.

c. Capability to segregate the liquid waste streams by the use of separate waste drain headers and waste collection sumps or tanks for each waste stream category.

The LWMS provides sufficient capacity, redundancy, and flexibility to treat the liquid radwaste in a manner that reduces the radionuclide concentrations to levels that do not exceed the effluent concentration limits in 10 CFR Part 20, Appendix B, and 10 CFR Part 50, Appendix I (Reference 8) dose objectives for liquid effluents.

A description of this system is presented in Subsection 11.2.

1.2.13.2 Gaseous Waste Management System

The gaseous waste management system (GWMS) is designed to monitor, control, collect, process, handle, store, and dispose of gaseous radioactive waste generated during normal plant conditions, including AOOs.

The GWMS manages radioactive gases collected from the off-gas system and other tank vents containing radioactive materials. The gaseous waste from the above sources is treated to reduce the quantity of radioactive material prior to release to the environment.

The radiation level in the processed gases is verified with radiation monitors prior to release to the environment.

The GWMS meets the following design requirements:

a. Provide the capability to monitor, control, collect, process, handle, store, and dispose of radioactive gaseous waste generated as the result of normal operation including AOOs to meet release radionuclide concentration limits in accordance with 10 CFR Part 20, Appendix B, prior to discharge to the environment.

b. Provide reasonable assurance that the release of radioactive material in gaseous effluents is kept as low as (is) reasonably achievable (ALARA).

c. Remove and reduce radioactive materials to the environment to meet the requirements of 10 CFR Part 50, Appendix I.

The gaseous radwaste subsystem uses charcoal at ambient temperature to delay the passage of radioactive gases. When operating at design conditions, the mass of charcoal provided in the absorber beds is sufficient to provide a delay of 45 days for xenon and a delay of 3.5 days for krypton.

The GWMS operates at pressures slightly above atmospheric, and therefore limits the potential for oxygen inleakage. Leakage from the GWMS is further limited through the use of welded connections wherever they are not restricted due to maintenance requirements. Control valves are provided with bellows seals to minimize leakage through the valve stems. The GWMS is designed to prevent the formation or buildup of explosive mixtures of hydrogen and oxygen by continuous monitoring and controlling the concentrations of hydrogen and oxygen through one of the two gas analyzers. The concentrations are confirmed by periodic sampling and analysis at several routing locations. When the oxygen concentration is detected to be higher than the predetermined setpoint, nitrogen is injected to dilute the concentration to below the lower flammable limit, which is 4 percent.

A description of this system is presented in Section 11.3.

1.2.13.3 Solid Waste Management System

The solid waste management system (SWMS) is designed to provide the means to monitor, control, collect, process, handle, and temporarily store the following prior to shipment: wet, dewatered, and dry solid radioactive waste generated during normal plant conditions, including AOOs. The SWMS processes both wet solid active waste and dry active waste (DAW) for onsite interim storage and shipment to the offsite disposal facility.

The SWMS meets the following design requirements:

a. Collect, segregate, treat, package, and store various solid radioactive wastes generated from the normal operation, maintenance, refueling, and AOOs.

b. Store, treat, and package the radioactive spent resin transported from the LWMS, CVCS, SFPCCS, and steam generator blowdown system (SGBDS).

c. Temporarily store the high- and low-activity waste, and to retrieve and ship wastes.

d. Treat and package wastes into drums or high-integrity containers (HICs) that satisfy the required regulations of the U.S. Department of Transportation (DOT) and the disposal facility.

e. Satisfy federal regulations, and protect the workers and the general public from radiation exposures ALARA.

The SWMS is subdivided into a spent resin transfer subsystem, packaging and storage subsystem, filter handling subsystem, dry active waste subsystem, concentrate treatment subsystem, and waste storage subsystem.

In order to reduce occupational radiation exposure, operations for processing and transfer of low- and intermediate-level radioactive waste are conducted remotely. Operator access is required for work related to low-level radioactive waste such as DAW.

A description of this system is presented in Section 11.4.

1.2.14 Plant Arrangement Summary

The APR1400 plant is composed of the following buildings:

a. Reactor containment building

b. Auxiliary building including two emergency diesel generator rooms

c. Turbine generator building

d. Compound building

e. Emergency diesel generator building with two emergency diesel generator rooms

f. Alternate alternating current gas turbine generator building

g. Essential service water intake structure and ultimate heat sink related structure

h. Component cooling water heat exchanger building

A standard plot of the APR1400 is shown in Figure 1.2-1, and the general arrangement drawings are shown in Figures 1.2-2 through 1.2-49.

1.2.14.1 Reactor Containment Building

The reactor containment building is designed using a post-tensioned concrete containment wall with a reinforced concrete internal structure. The reactor containment building houses a reactor, two steam generators, a pressurizer, reactor coolant loops, an IRWST, and portions of the auxiliary systems. The reactor containment building is designed to provide biological shielding and external missile protection, as well as to sustain all internal and external loading conditions that are reasonably expected to occur during the life of the plant.

The interior arrangement of the reactor containment building is designed to meet the requirements for all anticipated conditions during operation and maintenance, including new and spent fuel handling.

The equipment hatch is located at the operating floor level. The hatch is sized to accommodate the one-piece replacement of a steam generator. A polar bridge crane is supported from the wall of the reactor containment building. The polar bridge crane has the capability to install and remove the steam generators. Personnel access from the auxiliary building to the reactor containment building is through two hatches: one at the operating floor and the other one at the ground floor.

1.2.14.2 Auxiliary Building

The auxiliary building encompasses the reactor containment building and is on the common basemat that forms a monolithic structure with the reactor containment building. The auxiliary building houses the MCR, two EDG rooms, emergency core cooling system (ECCS) equipment area, fuel handling area, safety-related electrical and instrumentation and control (I&C) equipment areas, and two auxiliary feedwater storage tanks.

The auxiliary building is designed as a seismic Category I reinforced concrete structure. It houses safety-related equipment required to provide safe shutdown capability. Redundant divisions of systems essential for safe shutdown are physically separated from one another to prevent a common failure of both systems.

The remote shutdown console (RSC) is located in a separate fire area from the MCR and contains all controls necessary for safe shutdown.

For the convenience of operation and maintenance, including for installation work, there is a staging service area in the auxiliary building in front of the equipment hatch of the reactor containment building.

The auxiliary building is physically separated into Division I and Division II. The divisions are subdivided with quadrant walls (Quadrant A through Quadrant D). Quadrants A and C belong to Division I, and Quadrants B and D belong to Division II.

The APR1400 safety-related systems, including their components, are divided into the two divisions in the auxiliary building, which are physically and electrically independent of each other. The components are further divided into the four quadrants of the auxiliary building. A fire or flood in one quadrant does not affect the other quadrants.

The fuel handling area houses the following facilities:

a. Vehicle loading and unloading area

   A vehicle loading and unloading area is provided adjacent to the decontamination area. Space is provided within the building to permit inspection, tiedown adjustments, radiation monitoring, and storage of removable tiedown equipment. Doors are sized to allow the traffic of personnel and the vehicles ingressing and exiting the building. A vehicle cleanup and maintenance area is provided outside the building to service incoming vehicles that includes removing road dirt, cinders, salt, oil, and similar materials.

b. Decontamination pit

A decontamination pit is sized to permit the storage of a shipping cask, shipping cask head, and all other necessary rigging. Space also is allowed for portable scaffolds, elevated platforms, or ladders to gain access to the upper parts of the cask. Ample room is provided in the decontamination pit for the free passage of operating personnel around this equipment. Fixed or movable splash curtains or barriers are included to prevent splashing or accidental spillage out of the decontamination pit.

c. Spent fuel pool

The spent fuel pool is designed to allow the installation of the number of spent fuel storage racks that are required to accommodate 20 years of discharged fuel plus one full core storage. The pool is designed to allow the installation of underwater lighting around the periphery of the pool to enhance visibility during fuel handling. Fuel assemblies are to be placed in vertical cells (storage racks) and grouped in parallel rows. Cranes used in moving spent fuel have a lift height limit in order to maintain the required water shielding above the spent fuel storage racks during transfer operations.

d. Overhead crane

Spent fuel is loaded into a shipping cask in the cask loading pit using the spent fuel handling machine. The shipping cask is moved from the cask loading pit to the decontamination pit using an overhead crane. The cask travel path and the lifting height are restricted by limiting the distance above or adjacent to stored fuel within the spent fuel pool and the height of cask lift.

1.2.14.3 Turbine Generator Building

The turbine generator building is designed as a seismic Category II steel frame building and has the following four main levels: a basement, a grade floor, an operating floor, and a deaerator and storage tanks floor. The basement consists of concrete and steel structures, and the above-grade floor consists of steel structures.

The turbine generator building provides support and housing for the turbine generator and auxiliary equipment. The generator-associated equipment includes condensers, feedwater heaters, feedwater and condensate pumps, and the condensate cleanup system. Auxiliary equipment includes the lube oil system, hydrogen supply and cooling system, stator cooling system, seal oil system, and electro-hydraulic control system.

1.2.14.4 Compound Building

The compound building houses the systems and components related to radwaste management, access control, and the operation support center (OSC). The compound building consists of an access control facility, radwaste management facility, hot machine shop, and sampling facilities and laboratory.

The compound building is adjacent to the auxiliary building, and is classified as non-safety-related seismic Category II reinforced concrete structures. The compound building is supported by a reinforced concrete foundation that is separated from the foundation of the auxiliary building and designed so that it will not affect safety-related structures system and components in the auxiliary building under the safe shutdown earthquake (SSE) condition.

The compound building is designed to be protected from natural phenomena such as flooding, snow, and earthquakes and to accommodate loadings associated with environmental conditions to the extent necessary to retain within the building the spillage of potentially contaminated solids or liquids.

1.2.14.5 Emergency Diesel Generator Building

There are four EDG units. Two are located in the auxiliary building and two in the emergency diesel generator buildings (EDGBs). The two EDG units in the auxiliary building are separated so they are on opposite sides of the building in a mirror configuration.

The EDGB houses two EDG units and supporting equipment in two separate compartments. The EDGB is a seismic Category I reinforced concrete structure and each EDG compartment is designed to be physically separate to provide protection from fire, aircraft, missiles, and the environment. The EDGB is also designed to withstand the effects of internal and external hazards.

1.2.14.6 Alternate Alternating Current Gas Turbine Generator Building

The AAC gas turbine generator building is located on the north side of the plant site. The AAC gas turbine system provides an AAC power source during an SBO.

1.2.14.7 Essential Service Water Building

Two ESW buildings separated by division are classified as seismic Category I buildings with a concrete structure. The ESW building houses essential service water pumps, cooling tower, and cooling tower basin.

1.2.14.8 Component Cooling Water Heat Exchanger Building

Two CCW heat exchanger buildings next to each ESW building are classified as a seismic Category I building with a concrete structure. The CCW heat exchanger building houses CCW heat exchangers, debris filters.

1.2.14.9 Storage Tanks

The following storage tanks are located on the site, outside of building structures:

a. Reactor makeup water tank (seismic Category III)

b. Holdup tank (seismic Category III)

c. Boric acid storage tank (seismic Category I)

d. Condensate storage tank (seismic Category III)

e. Demineralized water storage tank (seismic Category III)

f. Fresh water storage tank (seismic Category III)

The reactor makeup water tank capacity is based on providing dilution to allow total recycle. The tank also provides dilution for one cold shutdown operation and subsequent startup at the most limiting time in core cycle. The reactor makeup water tank is described further in Subsection 9.3.4.

The holdup tank is sized to store all recoverable reactor coolant generated by one cold shutdown operation with the most reactive CEA withdrawn and subsequent startup at the most limiting time in core cycle. The holdup tank is described further in Subsection 9.3.4.

The boric acid storage tank is sized to permit one shutdown operation to cold shutdown, followed by a shutdown for refueling at the most limiting time in core cycle with the most reactive CEA withdrawn. The maximum concentration of boric acid in the tank is 2.5 weight percent (4,400 ppm boron). The boric acid storage tank is described further in Subsection 9.3.4.

The fresh water storage tank is sized to contain sufficient water for 2-hour operation of the largest design demand of any sprinkler system plus a 1,900 L/min (500 gpm) manual hose stream allowance to support fire suppression activities, or at least 1,135,500 liters (300,000 gallons) in ground-level storage tanks. Two 100 percent fresh water storage tanks are arranged separately so that the fire pumps can take suction from either or both tanks. The fresh water storage tank is described further in Subsection 9.5.1.

One 100 percent capacity demineralized water storage tank is provided for APR1400. The demineralized water storage tank stores and supplies demineralized water to the auxiliary feedwater storage tank for makeup and to other systems for various services during all modes. The demineralized water storage tank is described further in Subsection 9.2.6.

Two 50 percent capacity condensate storage tanks store and supply the condensate, as a readily available source of deaerated condensate for makeup, to the condenser. The condensate storage tank is described further in Subsection 9.2.6.

1.2.15 Combined License Information

COL 1.2(1) The COL applicant is to prepare a complete and detailed site plan.

1.2.16 References

1. 10 CFR 50.34, "Contents of Applications; Technical Information," U.S. Nuclear Regulatory Commission.

2. 10 CFR Part 50, Appendix A, General Design Criterion 54, "Systems Penetrating Containment," U.S. Nuclear Regulatory Commission.

3. 10 CFR 50.62, "Requirements for the Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants," U.S. Nuclear Regulatory Commission.

4. Staff Requirements Memorandum to SECY-93-087, II.Q, "Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems," " U.S. Nuclear Regulatory Commission, 1993

5. Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Rev.4, U.S. Nuclear Regulatory Commission, June 2006.

6. 10 CFR Part 50, Appendix A, General Design Criterion 60, "Control of Release of Radioactive Materials to the Environment," U.S. Nuclear Regulatory Commission.

7. 10 CFR Part 20, Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," U.S. Nuclear Regulatory Commission.

8. 10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to meet the Criterion 'As Low As is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," U.S. Nuclear Regulatory Commission.

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-1 Typical APR1400 Site Arrangement Plan

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-2 General Arrangement Reactor Containment Building Section A-A

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-3 General Arrangement Reactor Containment Building Section B-B

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-4 General Arrangement Reactor Containment Building El. 69′-0″ and El. 78′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-5 General Arrangement Reactor Containment Building El. 100′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-6 General Arrangement Reactor Containment Building El. 114′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-7 General Arrangement Reactor Containment Building El. 136′-6″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-8 General Arrangement Reactor Containment Building El. 156′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-9   General Arrangement Auxiliary Building Section A-A

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-10 General Arrangement Auxiliary Building Section B-B

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-11 General Arrangement Auxiliary Building El. 55′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-12 General Arrangement Auxiliary Building El. 68′-0″ and El. 86′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-13 General Arrangement Auxiliary Building El. 78′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-14 General Arrangement Auxiliary Building El. 100′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-15 General Arrangement Auxiliary Building El. 120′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-16   General Arrangement Auxiliary Building El. 137′-6″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-17   General Arrangement Auxiliary Building El. 156′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-18   General Arrangement Auxiliary Building El. 174′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-19 General Arrangement Auxiliary Building Roof El. 195′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-20 General Arrangement EDG Building El. 63′-0″, Sections A-A and B-B

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-21 General Arrangement EDG Building El. 100′-0″ and El. 121′-6″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-22 General Arrangement EDG Building El. 135′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-23 General Arrangement Compound Building Sections A-A and B-B

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-24 General Arrangement Compound Building El. 63′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-25 General Arrangement Compound Building El. 77′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-26 General Arrangement Compound Building El. 85′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-27   General Arrangement Compound Building El. 100′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-28 General Arrangement Compound Building El. 120′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-29 General Arrangement Compound Building El. 139′-6″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-30 General Arrangement Compound Building Roof El. 156′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-31 General Arrangement Turbine Generator Building Section A-A

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-32   General Arrangement Turbine Generator Building Section B-B

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-33 General Arrangement Turbine Generator Building El. 73′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-34 General Arrangement Turbine Generator Building El. 100′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-35 General Arrangement Turbine Generator Building El. 136′-6″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-36 General Arrangement Turbine Generator Building El. 170′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-37 General Arrangement Turbine Generator Building Roof Plan

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-38 AAC Gas Turbine Generator Building Section A-A

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-39   AAC Gas Turbine Generator Building Plan El. 79′-0″ and El. 85′-6″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-40 AAC Gas Turbine Generator Building Plan El. 100′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-41 AAC Gas Turbine Generator Building Plan El. 120′-0″

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-42 General Arrangement ESW/CCW Hx Building EL. 81'-0" (DIV. I)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-43 General Arrangement ESW/CCW Hx Building EL. 100'-0" (DIV. I)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-44 General Arrangement ESW/CCW Hx Building Roof Plan (DIV. I)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-45 General Arrangement ESW/CCW Hx Building Section (DIV. I)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-46 General Arrangement ESW/CCW Hx Building EL. 81'-0" (DIV. II)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-47 General Arrangement ESW/CCW Hx Building EL. 10'-0" (DIV.II)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-48 General Arrangement ESW/CCW Hx Building Roof Plan (DIV. II)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 1.2-49 General Arrangement ESW/CCW Hx Building Section (DIV.II)

1.3 Comparison with Other Facilities

1.3.1 Comparison with Similar Facility Designs

This section highlights the principal features of the APR1400 design and provides a comparison of the major plant design features with other pressurized water reactor (PWR) facilities. Table 1.3-1 summarizes the comparison of design and operating characteristics for the nuclear steam supply system (NSSS) and demonstrates that the APR1400 NSSS design is similar to both a previously certified design and a design now nearing completion of construction in Korea. Table 1.3-2 summarizes major plant features other than those in the NSSS.

Table 1.3-1 (1 of 13)

Comparison of NSSS Components

Design Feature	APR1400	System 80+	SKN 3&4	DCD Tier 2 Section
Control Characteristics				
Dissolved boron content for criticality, ppm (CEAs withdrawn, BOC)				4.3
Cold, 20 °C (68 °F)	1,238	1,431	1,238	
Hot, zero power, clean, 291.3 °C (556.3 °F)	1,187	1,414	1,187	
Hot, full power, equilibrium Xe	817	100	817	
Hot, full power, clean, 308.9 °C (588 °F)	1,067	1,270	1,067	
Nuclear Design Data				
Structural Characteristics				
Core equivalent diameter, m (in)	3.647 (143.6)	3.647 (143.6)	3.647 (143.6)	4.1
Number of fuel assemblies	241	241	241	
Core average H_2O/UO_2 volume ratio, first cycle, hot (core cell)	2.12	2.06	2.12	4.3
UO_2 fuel rod locations per assembly	236 (Batch A)	236 [1]	236 (Batch A)	
	236/224/220 (Batch B)	-	236/224/220 (Batch B)	
	236/224/220 (Batch C)	-	236/224/220 (Batch C)	

Table 1.3-1 (2 of 13)

Design Feature	APR1400	System 80+	SKN 3&4	DCD Tier 2 Section
Nuclear Design Data (cont.)				
Performance Characteristics				
Fuel management	3 Batch Mixed Central Zone	3 Batch Mixed Central Zone	3 Batch Mixed Central Zone	4.3
Fuel discharge burnup, MWd/MtU				
Core average burnup, 10 ppm soluble boron	17,571	15,300	17,571	
First core average discharge burnup	28,914	31,700	28,914	
Fuel rod enrichment without U-235				
Region 1	1.71	1.8	1.71	
Region 2	3.14/2.64	2.9	3.14/2.64	
Region 3	3.64/3.14	3.7	3.64/3.14	
Region 4	-	-	-	
Control Element Assemblies				
Material (full strength/part strength)	B_4C/Inconel	B_4C or Ag-In-Cd/Inconel (2)	B_4C/Inconel	4.2
Number of control element assemblies (full strength/part strength)	81/12	68/25 (3)	81/12	
Number of absorber rods per CEA (or rod cluster control assembly [RCCA])	4 or 12	4 or 12	4 or 12	
Total rod worth (all CEAs inserted, hot, 308.9 °C (588 °F)), %Δρ	16.70	16.4 (typical)	16.70	4.3

Table 1.3-1 (3 of 13)

Design Feature	APR1400	System 80+	SKN 3&4	DCD Tier 2 Section
Nuclear Design Data (cont.)				
Kinetic Characteristics Range Over First Cycle				
Moderator temperature coefficient $\Delta\rho$/°C (hot, full power, BOC/EOC)	-1.71×10^{-4} / -4.34×10^{-4}	-1.3×10^{-4} / -4.7×10^{-4}	-1.71×10^{-4} / -4.34×10^{-4}	4.3
Moderator pressure coefficient $\Delta\rho$/psi (hot, operating, BOC)	$+0.44\times 10^{-6}$	$+0.4\times 10^{-5}$	$+0.44\times 10^{-6}$	
Moderator void coefficient $\Delta\rho$/% void (hot, operating, BOC)	-0.21×10^{-3}	-0.22×10^{-3}	-0.21×10^{-3}	
Doppler coefficient $\Delta\rho$/°C (hot operating range, BOC/EOC)	-2.54×10^{-5} / -2.95×10^{-5}	-2.74×10^{-5} / -2.93×10^{-5}	-2.54×10^{-5} / -2.95×10^{-5}	
Thermal and Hydraulic Design Parameters				
Total core heat output, MWt	3,983	3,914	3,983	4.4
Total core heat output, 10^6 kcal/hr (MBtu/hr)	3,425 (13,590)	3,367 (13,360)	3,425 (13,590)	
Average fuel rod energy deposition fraction	0.975	0.975	0.975	
Primary system pressure, kg/cm^{2}A (psia)	158.2 (2,250)	158.2 (2,250)	158.2 (2,250)	

Table 1.3-1 (4 of 13)

Design Feature	APR1400	System 80+	SKN 3&4	DCD Tier 2 Section
Engineering Factors				
Engineering heat flux factor	1.03	1.03	1.03	4.4
Engineering enthalpy rise factor	1.03	1.03	1.03	
Engineering factor on LHR	1.03	1.03	1.03	
Coolant Flow				
Total coolant flow, 10^6 kg/hr (10^6 lb/hr)	75.6 (166.6)	75.2 (165.8)	75.6 (166.6)	4.4
Core flow, 10^6 kg/hr (10^6 lb/hr)	73.3 (161.6)	73.0 (160.8)	73.3 (161.6)	
Core flow area, m^2 (ft^2)	5.83 (62.7)	5.65 (60.8)	5.83 (62.7)	
Core average coolant velocity, m/s (ft/s)	4.94 (16.2)	5.10 (16.7)	4.94 (16.2)	
Core average mass velocity, 10^6 kg/hr-m^2 (10^6 lbm/hr-ft^2)	12.60 (2.58)	12.94 (2.65)	12.60 (2.58)	

Table 1.3-1 (5 of 13)

Design Feature	APR1400	System 80+	SKN 3&4	DCD Tier 2 Section
Reactor Coolant Temperatures				
Reactor inlet coolant temperature, °C (°F) (7)	290.6 (555)	291.1 (556)	290.6 (555)	4.4
Reactor outlet coolant temperature, °C (°F) (7)	323.9 (615)	323.9 (615)	323.9 (615)	
Core-exit average coolant temperature, °C (°F) (7)	325.0 (617)	325.0 (617)	325.0 (617)	
Average rise in vessel, °C (°F) (7)	33.3 (60)	32.8 (59)	33.3 (60)	
Average rise in core, °C (°F) (7)	34.4 (62)	33.9 (61)	34.4 (62)	
Average temperature in core, °C (°F) (7)	307.8 (586)	308.3 (587)	307.8 (586)	
Average temperature in vessel, °C (°F) (7)	307.2 (585)	307.8 (586)	307.2 (585)	
Characteristics of Rod and Channel with Minimum DNBR				
Outlet temperature, °C (°F) (7)	340.6 (645)	340.0 (644)	340.6 (645)	
Minimum DNBR at nominal conditions (CHF correlation)	2.44(KCE-1)(4)	2.00(CE-1)(4)	2.44(KCE-1)(4)	

Table 1.3-1 (6 of 13)

Design Feature	APR1400	System 80+	SKN 3&4	DCD Tier 2 Section
Core Characteristics at Full Power				
Total heat transfer area, m^2 (ft^2)	6,454 (69,470)	6,592 (70,960)	6,454 (69,470)	4.4
Core average fuel rod heat flux, kcal/hr-m^2 (Btu/hr-ft^2)	517,361 (190,735)	497,200 (183,300)	517,361 (190,735)	
Maximum fuel rod heat flux, kcal/hr-m^2 (Btu/hr-ft^2)	1,215,000 (448,000)	1,164,000 (429,100)	1,215,000 (448,000)	
Average fuel rod LHR, W/cm (kW/ft)	179.2 (5.46)	175.9 (5.36)	179.2 (5.46)	
Maximum fuel rod LHR, W/cm (kW/ft)	420.8 (12.8)	413.4 (12.6)	420.8 (12.8)	
Maximum fuel centerline temperature at 100 % power, °C (°F)(7)	1,712 (3,114)	1,748 (3,179)	1,712 (3,114)	
Mechanical Design Parameters				
Fuel Assemblies				
Fuel rod array	square, 16 × 16	square, 16 × 16	square, 16 × 16	4.2
Fuel rod pitch, cm (in)	1.2852 (0.506)	1.2852 (0.506)	1.2852 (0.506)	
Fuel rod to fuel rod, cm (in)	20.23 × 20.23 (7.964 × 7.964)	20.25 × 20.25 (7.972 × 7.972)	20.23 × 20.23 (7.964 × 7.964)	
Total fuel weight, kg UO_2 (lb UO_2) (assuming all rod locations are fuel rods)	117.8×10^3 (259.7×10^3)	120.0×10^3 (264.5×10^3)	117.8×10^3 (259.7×10^3)	
Number of grids per assembly	12	11	12	

Table 1.3-1 (7 of 13)

Design Feature	APR1400	System 80+	SKN 3&4	DCD Tier 2 Section
Fuel Rods				
Number of locations	56,876 [(5)]	56,876 [(5)]	56,876 [(5)]	4.1
Clad outside diameter (OD), cm (in)	0.950 (0.374)	0.970 (0.382)	0.950 (0.374)	4.2
Diametral gap (cold), cm (in)	0.01651 (0.0065)	0.01651 (0.0065)	0.01651 (0.0065)	
Clad thickness, cm (in)	0.05715 (0.0225)	0.06350 (0.025)	0.05715 (0.0225)	
Cladding material	ZIRLO	Zircaloy-4	ZIRLO	
Fuel Pellets				
Material	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered	4.2
Diameter, cm (in)	0.8192 (0.3225)	0.827 (0.3255)	0.8192 (0.3225)	
Length (enriched uranium), cm (in)	0.98 (0.387)	0.991 (0.390)	0.98 (0.387)	
Control Element Assemblies				
Clad material	Inconel 625	Inconel 625	Inconel 625	4.2
Clad thickness, cm (in)	0.089 (0.035)	0.089 (0.035)	0.089 (0.035)	

Table 1.3-1 (8 of 13)

Design Feature	APR1400	System 80+	SKN 3&4	DCD Tier 2 Section
Reactor Coolant System Code Requirements				
Reactor vessel	ASME III, Class 1	ASME III, Class 1	ASME III, Class 1	5.2, 5.3, 5.4
Steam generator, tube side	ASME III, Class 1	ASME III, Class 1	ASME III, Class 1	
Steam generator, shellside	ASME III, Class 2	ASME III, Class 2	ASME III, Class 2	
Pressurizer	ASME III, Class 1	ASME III, Class 1	ASME III, Class 1	
Pilot operated safety relief valves	ASME III, Class 1	ASME III, Class 1	ASME III, Class 1	
Reactor coolant piping	ASME III, Class 1	ASME III, Class 1	ASME III, Class 1	
Principal Design Parameters of the Reactor Vessel				
Material	Low-alloy, steel with austenitic SS cladding	Low-alloy, steel with austenitic SS cladding	Low-alloy, steel with austenitic SS cladding	5.3
Design pressure, kg/cm^{2}A (psia)	175.8 (2,500)	175.8 (2,500)	175.8 (2,500)	
Design temperature, °C (°F)	343.3 (650)	343.3 (650)	343.3 (650)	
Normal operating pressure, kg/cm^{2}A (psia)	158.2 (2,250)	158.2 (2,250)	158.2 (2,250)	
Inside diameter at shell, m (in)	4.63 (182-1/4)	4.63 (182-1/4)	4.63 (182-1/4)	
Outside diameter across inlet nozzles, m (in)	6.88 (271)	6.88 (271)	6.88 (271)	
Overall height of vessel and head, m (in) to top (including closure head, CEDM nozzles and bottom head instrumentation nozzles)	14.83 (583-7/8)	14.83 (583-7/8)	14.83 (583-7/8)	
Minimum cladding thickness, m (in)	0.318 (1/8)	0.318 (1/8)	0.318 (1/8)	-

Table 1.3-1 (9 of 13)

Design Feature	APR1400	System 80+	SKN 3&4	DCD Tier 2 Section
Principal Design Parameters of the Reactor Coolant Piping				
Material	Carbon steel internally clad with stainless steel	Carbon steel internally clad with stainless steel	Carbon steel internally clad with stainless steel	5.4.3
Hot leg – ID, mm (in)	1,066.8 (42)	1,066.8 (42)	1,066.8 (42)	
Cold leg – ID, mm (in)	762 (30)	762 (30)	762 (30)	
Between pump and steam generator – ID, mm (in)	762 (30)	762 (30)	762 (30)	
Design pressure, kg/cm^{2}A (psia)	175.8 (2,500)	175.8 (2,500)	175.8 (2,500)	
Principal Design Parameters of the Reactor Coolant System				
Operating pressure, kg/cm^{2}A (psia)	158.2 (2,250)	158.2 (2,250)	158.2 (2,250)	5.1, 5.4
Reactor inlet temperature, °C (°F) (7)	290.6 (555)	291.1 (556)	290.6 (555)	
Reactor outlet temperature, °C (°F) (7)	323.9 (615)	323.9 (615)	323.9 (615)	
Number of loops	2	2	2	
Design pressure, kg/cm^{2}A (psia)	175.8 (2,500)	175.8 (2,500)	175.8 (2,500)	
Design temperature, °C (°F)	343.3 (650)	343.3 (650)	343.3 (650)	
Hydrostatic test pressure (cold), kg/cm^{2}A (psia)	219.7 (3,125)	219.7 (3,125)	219.7 (3,125)	
Total coolant volume, m^3 (ft^3)	455.3 (16,079)	448.1 (15,825.5) (8)	455.3 (16,079)	
Total reactor flow, L/min (gal/min) (6)	1,689,000 (446,300)	1,683,000 (444,650)	1,689,000 (446,300)	

Table 1.3-1 (10 of 13)

Design Feature	APR1400	System 80+	SKN 3&4	DCD Tier 2 Section
Nuclear Design Data				
Principal Design Parameters of the Reactor Coolant Pumps				
Number of units	4	4	4	-
Type	Vertical, single-stage centrifugal with bottom suction and horizontal discharge	Vertical, single-stage centrifugal with bottom suction and horizontal discharge	Vertical, single-stage centrifugal with bottom suction and horizontal discharge	5.4.1
Design pressure, kg/cm^2A (psia)	175.8 (2,500)	175.8 (2,500)	175.8 (2,500)	
Design temperature, °C (°F)	343.3 (650)	343.3 (650)	343.3 (650)	
Operating pressure, nominal, kg/cm^2A (psia)	158.2 (2,250)	158.2 (2,250)	158.2 (2,250)	
Suction temperature, °C (°F) (7)	290.6 (555)	291.1 (556)	290.6 (555)	
Rated flow, L/min (gal/min)	460,256 (121,600)	437,014 (115,447)	460,256 (121,600)	
Rated head, m (ft)	109.7 (360)	114.0 (374)	109.7 (360)	
Hydrostatic test pressure, (cold) kg/cm^2A (psia)	219.7 (3,125)	219.7 (3,125)	219.7 (3,125)	
Motor type	AC Induction Single Speed	AC Induction Single Speed	AC Induction Single Speed	
Motor rating (cold), kW (hp)	10,067 (13,500)	8,948 (12,000)	10,067 (13,500)	

Table 1.3-1 (11 of 13)

Design Feature	APR1400	System 80+	SKN 3&4	DCD Tier 2 Section
Nuclear Design Data				
Principal Design Parameters of the Steam Generators				
Number of units	2	2	2	5.4.2
Type	Vertical U-tube with integral economizer	Vertical U-tube with integral economizer	Vertical U-tube with integral economizer	
Tube material	SB-163 NiCrFe alloy 690	SB-163 NiCrFe alloy 690	SB-163 alloy 690	
Shell material	SA-533 Gr. B, Class 1 or SA-508, Class 3	Primary side low-alloy steel clad with austenitic stainless steel	SA-533 Gr. B, Class 1 or SA-508, Class 3	
Tube-side design pressure, kg/cm^2A (psia)	175.76 (2,500)	175.76 (2,500)	175.76 (2,500)	
Tube-side design temperature, °C (°F)	343.33 (650)	343.33 (650)	343.33 (650)	
Tube side design flow, kg/hr (lb/hr) per steam generator	37.78×10^6 (83.3×10^6)	37.6×10^6 (82.9×10^6)	37.78×10^6 (83.3×10^6)	
Shellside design pressure, kg/cm^2A (psia)	84.36 (1,200)	84.36 (1,200)	84.36 (1,200)	
Shellside design temperature, °C (°F)	298.88 (570)	298.88 (570)	298.88 (570)	
Operating pressure, tube-side, nominal, kg/cm^2A (psia)	158.2 (2,250)	158.2 (2,250)	158.2 (2,250)	
Operating pressure, shellside, maximum, kg/cm^2A (psia)	77.3 (1,100)	77.3 (1,100)	77.3 (1,100)	

Table 1.3-1 (12 of 13)

Design Feature	APR1400	System 80+	SKN 3&4	DCD Tier 2 Section
Maximum moisture at outlet at full load, %	0.25	0.25	0.25	5.4.2
Hydrostatic test pressure, tube-side (cold), kg/cm^2A (psia)	219.7 (3,125)	219.7 (3,125)	219.7 (3,125)	5.2.2.1.2
Steam pressure, kg/cm^2A (psia), at full power	70.3 (1,000)	70.3 (1,000)	70.3 (1,000)	5.4.2
Steam temperature, °C (°F) at full power	285 (545)	285 (545)	285 (545)	
Steam flow, at full power, lb/hr per steam generator, kg/hr (lb/hr)	4.070×10^6 (8.975×10^6)	4.0×10^6 (8.82×10^6)	4.070×10^6 (8.975×10^6)	
pressurizer				
Design pressure, kg/cm^2A (psia)	175.8 (2,500)	175.8 (2,500)	175.8 (2,500)	5.4.10
Design temperature, °C (°F)	371.1 (700)	371.1 (700)	371.1 (700)	
Normal operating pressure, kg/cm^2A (psia)	158.2 (2,250)	158.2 (2,250)	158.2 (2,250)	
Normal operating temperature, °C (°F)	344.8 (652.7)	344.8 (652.7)	344.8 (652.7)	
Internal free volume, $m^3(ft^3)$	68.0 (2,400)	68.0 (2,400)	68.0 (2,400)	
Normal (full power) operating water volume, $m^3(ft^3)$	33.2 (1,171)	34.0 (1,200)	33.2 (1,171)	
Normal (full power) steam volume, $m^3(ft^3)$	35.7 (1,260)	34.9 (1,234)	35.7 (1,260)	
Installed heater capacity, kW	2,400	2,400	2,400	

Table 1.3-1 (13 of 13)

Design Feature	APR1400	System 80+	SKN 3&4	DCD Tier 2 Section
ESFAS				
Initiating ESFAS				
Number of manual switches	2 sets of 2 each	2 sets of 2 each	2 sets of 2 each	7.3
Automatic initiation parameter channels/logic	4 channels provided, coincidence of 2 required for each function	4 channels provided, coincidence of 2 required for each function	4 channels provided, coincidence of 2 required for each function	
Initiating Reactor Trip				
Number of manual switches	2 sets of 2 in MCR 1 set of 2 in RSR	2 sets of 2 each in both MCR and at RSP	2 sets of 2 in MCR 1 set of 2 in RSR	7.2
Automatic initiation parameter channels/ logic	4 channels provided, coincidence of 2 required for trip	4 channels provided, coincidence of 2 required for trip	4 channels provided, coincidence of 2 required for trip	

(1) In the first core, some UO_2 rods may be replaced by burnable absorber rods.

(2) Inconel part-strength CEAs in System 80+

(3) Locations are provided for eight additional CEAs.

(4) Minimum DNBR at nominal conditions

(5) Some of the rod locations are occupied by burnable absorber rods.

(6) Design minimum

(7) Temperatures are given to the nearest degree.

(8) Cold condition including pressurizer

Table 1.3-2 (1 of 5)

Comparison of Plant Components Other Than NSSS

Design Feature	APR1400	System 80+	SKN 3&4	DCD Tier 2 Section
Containment System				
Containment				
Type	Steel-lined prestressed concrete cylinder with hemispherical dome	Steel spherical containment shell, surrounded by reinforced concrete shield building	Steel-lined prestressed concrete cylinder with hemispherical dome	3.8, 6.2
Leak rate (%/d)	0.1 (24 hr) 0.05 (after 24 hr)	0.5 (24 hr) 0.25 (after 24 hr)	0.15 (24 hr) 0.075 (after 24 hr)	
Design pressure, kg/cm^2 (psig)	4.22 (60)	3.726 (53)	4.22 (60)	
Net Free volume, 10^3 m^3 (10^6 ft^3)	88.576 (3.128)	95.626 (3.337)	88.576 (3.128)	
Containment Spray				6.2.2
Number of pumps	2	2	2	
Number of heat exchangers	2	2	2	
Design capacity, each, lpm (gpm)	18,927 (5,000)	18,927 (5,000)	18,927 (5,000)	
Containment Coolers				9.4.6
Type	Normal and loss of offsite power	Normal and loss of offsite power	Normal and loss of offsite power	
Number of units	4	4	4	
Capacity, Kcal/hr (Btu/hr)	1,725,264 (6,846,400)	755,988 (3,000,000)	1,725,264 (6,846,400)	

Table 1.3-2 (2 of 5)

Design Feature	APR1400	System 80+	SKN 3&4	DCD Tier 2 Section
Electric System				
Onsite Power Systems, AC				8.3.1
Generator prime mover	Diesel engine	Diesel engine	Diesel engine	
Number of units	4	2	2	
Capacity, each (kW)	9,100 (EDG A, B) 7,500 (EDG C, D)	5,500	8,000	
Other Systems				
Essential Service Water				9.2.1
Number of trains	2	2	2	
Number of pumps/train	2	2	2	
Pump type	Vertical turbine type wet pit	Vertical centrifugal type wet pit	Vertical turbine type wet pit	
Rated flow rate, each, lpm (gpm)	75,708 (20,000)	54,889 (14,500)	64,352 (17,000)	
Component Cooling Water				9.2.2
Number of trains	2	2	2	
Number of pumps/train	2	2	2	
Design capacity, each, lpm (gpm)	94,635 (25,000)	57,358 (15,200)	70,030 (18,500)	
Number of heat exchangers/train	3	2	3	
Heat exchanger type	Plate	Shell and tube	Plate	

Table 1.3-2 (3 of 5)

Design Feature	APR1400	System 80+	SKN 3&4	DCD Tier 2 Section
Spent Fuel Pool Cooling and Cleanup				9.1.3
Cooling pump design capacity each, lpm (gpm)	15,142 (4,000)	13,249 (3,500)	15,142 (4,000)	
Condensate Storage Facility				9.2.6
Total Capacity, L (10^3 gal)	1,930,560 (510,000)	--	1,930,560 (510,000)	
Plant Fire Protection				9.5.1
Water source	Fresh water tank	Fresh water tank	Fresh water tank	
Backup source	Seismic Category I fire water storage tank	Seismic Category I fire water storage tank	Seismic Category I fire water storage tank	
Essential chilled water				9.2.7
Number of pump and chiller/divisions	2 (one per quad)	2	2	
Emergency Diesel Generators				9.5.4
Fuel oil storage capacity per diesel operating at full power (days)	7 days, plus a margin for periodic testing	7 days, plus a margin for periodic testing	7 days, plus a margin for periodic testing	
Turbine Generator				10.2
Output, guaranteed (MWe)	1,425	1,391	1,455	

Table 1.3-2 (4 of 5)

Design Feature	APR1400	System 80+	SKN 3&4	DCD Tier 2 Section
Main Steam Supply				10.3,
Total steam flow, kg/hr (1b/hr)	8.14×10^6 (17.95×10^6)	8.0×10^6 (17.64×10^6)	8.14×10^6 (17.95×10^6)	10.4.1
Steam generator dome pressure, kg/cm^2 A (psia)	70.3 (1,000)	70.3 (1,000)	70.3 (1,000)	
Steam generator dome temperature °C (°F)	284.2 (543.6)	284.8 (544.6)	284.8 (544.6)	
Condenser Type	Single pressure	Multi-pressure	Single pressure	
Design Operating Pressure, kg/cm^2A (in HgA)	0.09 (2.6)	0.06/0.08/0.10 (1.8/2.3/3.0)	0.05 (1.5)	
Turbine Bypass				10.4.4
Capacity (% of rated load main steam flow)	55 (to condenser)	55 (to condenser)	55 (to condenser)	
Auxiliary Feedwater				10.4.9
Pump prime movers	2 turbine-driven, 2 motor-driven	2 turbine-driven, 2 motor-driven	2 turbine-driven, 2 motor-driven	
Rated flow rate, each, L/min (gpm)	2,461 (650)	1,892 (500)	2,461 (650)	

Table 1.3-2 (5 of 5)

Design Feature	APR1400	System 80+	SKN 3&4	DCD Tier 2 Section
Radwaste Systems				
Liquid radwaste system tank volume, L (gal)	590,524 (156,000) [(2)] (one units)	2,175,491 (574,704) [(1)] (one unit)	635,949 (168,000) [(2)] (two units)	11.2
Gaseous radwaste system holdup time (days)	45 (xenon) 3.5 (krypton)	30 (xenon) 3 (krypton)	45 (xenon) 3.5 (krypton)	11.3
Solid radwaste system process type solidification agent	Solidification polymer	Dewatering N/A	Solidification polymer	11.4
Emergency Support Facilities				
Technical support center (TSC)	Dedicated TSC for each unit, located in the auxiliary building of the respective unit	Dedicated TSC for each unit, located in the auxiliary building of the respective unit	Dedicated TSC for each unit, located in the auxiliary building of the respective unit	13.3

(1) The neutralization tank is included (435,322 L (115,000 gal) × 2).

(2) APR1400 liquid radwaste systems have no neutralization tanks.

1.4 Identification of Agents and Contractors

1.4.1 Applicant – Program Manager

Korea Electric Power Corporation (KEPCO) was founded with the objective to facilitate the development of electric power supply in Korea, meet the country's power supply and demand needs, and contribute to the national economy in accordance with the Korea Electric Power Corporation Act. KEPCO is classified as a market-oriented public corporation under the Act on the Management of Public Institutions. KEPCO's areas of business are based on the aforementioned objectives and include the development of electric power resources, electric power generation, transmission, transformation, and distribution, as well as related marketing, research, technological development, overseas business, investment, corporate social responsibility, and use of its property. KEPCO provides funds for the APR1400 design certification and reviews top-level policy issues.

Korea Hydro & Nuclear Power (KHNP) is responsible for the APR1400 design. The design is based on the Korean standard nuclear reactor, the OPR1000. The OPR1000 design is based on more than 30 years of experience in the construction, operation, and design of nuclear power plants beginning with Kori Unit 1. Following the construction of Hanbit Units 3 & 4, which marked the beginning of the Korean standard nuclear power plant, KHNP has constructed and operated Hanbit Units 5 & 6, Hanul Units 3 through 6, and Shin-Kori Units 1 & 2. This application for design certification of the APR1400 design is also based on the ABB-CE System 80+ certified design. Using the design expertise accumulated while developing the OPR1000, KHNP upgraded the capacity for its standard design to 1,400 MW, making it a globally competitive reactor.

KHNP relies on the following three primary organizations for support of the APR1400 design certification: Korea Electric Power Corporation (KEPCO) Engineering & Construction Company, Inc. (KEPCO E&C); KEPCO Nuclear Fuel Co., Ltd. (KEPCO NF); and Doosan Heavy Industry & Construction Co., Ltd. (Doosan).

KEPCO is located in Seoul of the Republic of Korea, and KHNP is headquartered in Gyeongju. KEPCO E&C is headquartered in Yongin, KEPCO NF is located in Daejeon, and Doosan is located in Changwon.

1.4.2 Architect Engineer (A/E) – KEPCO E&C

KEPCO E&C is the prime contractor to KHNP for architectural and engineering services and other related services.

KEPCO E&C provides engineering and engineering management services, project management assistance, and support services.

KEPCO E&C was established in 1975 to meet the increasing demands for architectural and engineering capabilities in Korea. Since its inception, KEPCO E&C has played the leading role in consulting and engineering activities in all of the Korean nuclear projects.

The nuclear experience of KEPCO E&C dates to 1976, when it undertook several design tasks on the first nuclear power plant in Korea. Subsequently, KEPCO E&C has participated in the construction projects of all Korean nuclear power plants and has provided various engineering services. Recently, KEPCO E&C has been actively expanding business areas that include international projects.

1.4.3 Major Equipment Supplier – DOOSAN

DOOSAN was established in 1962 under the name of Hyundai International Inc., and its primary business was the manufacture of various industrial machinery and equipment.

In the course of its development, the company changed management in November 1980 in conformance with the Korea Governmental Policy on Heavy Industry Distribution and was renamed "Korea Heavy Industries and Construction Co., Ltd. (HANJUNG)."

The structural reform of electric equipment was completed in November 1999 based on the policy as to the management structural improvement of public enterprise in April 1998. HANJUNG was unificated in the division of electric equipment. Under the privatization policy of public enterprise, HANJUNG was renamed DOOSAN in March 2001 in its initial public offering in October 2000.

This government action and subsequent transformation in the ownership of the company has strengthened DOOSAN’s capabilities to assume sole responsibility for all nuclear power plants to be installed in Korea.

Under the above policy, DOOSAN was designated by KHNP as the prime contractor for the supply of equipment, materials, and related services of the nuclear steam supply system (NSSS) and turbine generators (T/Gs) for HUN 3&4, HBN 5&6, HUN 5&6, Shin-Kori 1&2, Shin-Wolsong 1&2, and Shin-Kori 3&4 in Korea.

Currently, DOOSAN is committed to the manufacturing of NSSSs and T/Gs for Shin-Hanul 1&2 in Korea.

1.4.4 Nuclear Steam Supply System Designer – KEPCO E&C

KEPCO E&C – Nuclear Steam Supply System (KEPCO E&C-NSSS) division worked with Asea Brown-Boveri Combustion Engineering (ABB-CE) on the 1,000 MW pressurized water reactor (PWR) Hanbit Units 3 & 4 in 1987 and was able to develop the technology to achieve self-reliance of the NSSS design.

Since 1991, KEPCO E&C-NSSS has been designing the NSSS of all nuclear power plants built in Korea, including the 1,000 MW PWR, and has developed the NSSS design of the 1,400 MW APR1400 plant.

1.4.5 Nuclear Fuel Design and Manufacturing – KEPCO NF

KEPCO NF is a fuel design and fabrication company that has been responsible for the fuel supplied to all nuclear power plants in Korea for decades. Its major activities include initial and reload core design, fuel development, fuel assembly and component manufacture, and fuel services.

1.4.6 Combined License Information

COL 1.4(1) The combined license (COL) applicant that references the APR1400 design certification is to identify major agents, contractors, and participants for the construction and operation of the nuclear power plant.

1.5 Requirements for Additional Technical Information

This section describes additional technical information for the unique design features of the APR1400.

1.5.1 Fluidic Device Design

Conventional nuclear power plants are designed to deliver cooling water into a reactor vessel from safety injection tanks (SITs) in the refill phase and to deliver the cooling water by safety injection pumps (SIPs) during the reflood phase in the event of a large-break loss-of-coolant accident (LBLOCA). During an LBLOCA, the fuel cladding temperature increases because the liquid around the core is carried away by a significant loss of reactor coolant from the RCS.

The safety injection system (SIS) is required to inject cooling water into the core to limit the fuel temperature increase. Safety injection (SI) water in the SIT plays the role of rapidly raising the water level in the downcomer up to the cold leg bottom in the refill stage, removing decay heat of the reactor core and sensible heat of the nuclear fuels and metal structures in the early reflood phase. The SITs of conventional nuclear power plants deliver excessive cooling water to the reactor vessel after the water level has been raised to the cold leg bottom elevation, causing SI water to flow into the containment atmosphere. The excess flow to the containment limits the usefulness of SI water and can cause a decrease in the reflood rate.

The fluidic device (FD) is installed inside the pressurized SIT of the APR1400 and passively controls the water injection flow rate from the SIT. The FD consists primarily of a standpipe and vortex chamber. The vortex chamber can receive injection water through control ports as well as through the standpipe. When the SIT water level is above the top of the standpipe, water enters the vortex chamber through both the top of the standpipe and the control ports, resulting in injection of water at a large flow rate. When the water level drops below the top of the standpipe, water only enters the vortex chamber through the control ports, resulting in vortex formation in the vortex chamber and a relatively small flow injection. Therefore, the SIT provides short-term large flow injection to refill the reactor vessel, and a smaller flow injection, which, in conjunction with the SIP, adequately supports the core reflooding phase. The FD improves the LOCA safety analysis by reducing the amount of SI water that would spill into the containment once the downcomer is full, thereby improving the overall reliability of SI water injection.

The performance of the FD has also been evaluated by repeated experiments in the full-scale valve performance evaluation rig (VAPER) Test Facility at the Korea Atomic Energy Research Institute (KAERI). The experimental results confirm that the currently developed FD satisfies the major performance requirements of the plant design regarding the injection flow rate, pressure loss coefficient (K-factor), and injection duration time. The pressure loss coefficient of the small flow rate period is almost 10 times higher than that of the large flow rate period due to the strong vortex motion in the FD. The K-factor of the SIS has been evaluated based on the K-factor obtained from the tests, and this value essentially matches the target design value of the SIS in the APR1400.

A quality assurance program for the FD tests has been applied to provide reasonable assurance of high-quality documentation of the testing.

1.5.2 Pilot Operated Safety Relief Valve Design

The APR1400 adapts the pilot operated safety relief valve (POSRV) to provide overpressure protection of the reactor coolant system (RCS). Four POSRVs are connected to the top of the pressurizer by separate inlet lines. These valves also provide rapid depressurization functions during the beyond-design-basis event of a total loss of feedwater event for feed-and-bleed operations and for severe accidents to reduce RCS pressure prior to vessel breach.

The four POSRVs are designed to maintain the RCS pressure below 110 percent of design pressure during the worst-case transient, a loss-of-load event with a delayed reactor trip.

The pressurizer POSRVs perform the overpressure protection function with two pairs of spring-loaded pilot assemblies and a main valve and perform the rapid depressurization function using two motor-operated pilot valves in series and a main valve.

Each pressurizer POSRV provides the overpressure protection function with a main valve and two spring-loaded pilot valves in assembly. Each spring-loaded pilot valve in the assembly consists of a motor-operated isolation valve, spring-loaded pilot valve, check valve, and manual isolation valve. The spring-loaded pilot valve of each POSRV opens automatically if the system pressure increases to the POSRV set pressure, thus opening the main valve. The motor-operated isolation valves are normally open but are manually closed by an operator to prevent discharge when the spring-loaded pilot valves fail to close. The manual isolation valves are normally open but are manually closed to block the main valve from opening when repairing a spring-loaded pilot valve or conducting a setpoint test.

Each pressurizer POSRV provides the rapid depressurization function with a main valve and two motor-operated pilot valves installed in series. The motor-operated pilot valves are normally closed, but an operator remotely and manually opens the valves to open the main valve for the rapid depressurization of the RCS.

The pressurizer POSRVs are capable of discharging steam, water, and steam-water mixture.

A typical POSRV is illustrated in Figure 5.4.14-1. The design parameters are given in Table 5.4.14-1.

1.5.3 Direct Vessel Injection

One of the advanced design features in the safety injection system (SIS) for the APR1400 is direct vessel injection (DVI), which reduces the loss of injection water during cold leg pipe breaks. The SIS is designed to provide DVI. The discharge of each safety injection (SI) pump and safety injection tank (SIT) is piped directly to a reactor vessel nozzle instead of a cold leg nozzle. The flow is directed into the reactor vessel downcomer region through the DVI nozzle. The APR1400 design places the four DVI nozzles on the reactor vessel above the hot leg / cold leg centerline at angles of 45°, 135°, 225°, and 315° from the referenced hot leg.

1.5.4 Instrumentation and Control System

The APR1400 instrumentation and control (I&C) systems consist of the safety I&C system, non-safety control and monitoring system, diverse actuation system (DAS), and human-system interface (HSI) system.

The safety I&C system consists of the plant protection system (PPS), core protection calculator system (CPCS), engineered safety features – component control system (ESF-CCS), and the qualified indication and alarm system – P (QIAS-P). The control and monitoring system includes the power control system (PCS), process-component control system, qualified indication and alarm system – non-safety (QIAS-N), and information processing system (IPS). The DAS is composed of the diverse protection system (DPS), diverse indication system (DIS), and diverse manual ESF actuation (DMA) switch. The HSI system includes the compact workstation-based operator console with an information flat panel display and ESF-CCS soft control module (ESCM), large display panel, safety console with ESCM / manual switches / operator module / display device in the main control room, compact workstation-based operator console with ESCM, and a shutdown overview panel in the remote shutdown room.

The safety I&C system is implemented on the four channels of common programmable logic controller qualified for Class 1E grade in accordance with IEEE Std. 603 (Reference 1) and IEEE Std. 7-4.3.2 (Reference 2), and each channel is located in the separate I&C equipment room.

The software for the digital I&C system is designed, verified, and validated in accordance with software life-cycle process conforming with NRC RG 1.152 (Reference 3).

The control and monitoring system is implemented on a distributed control system.

The diversity and defense-in-depth analysis is performed to demonstrate that the DAS and control system meet SECY 93-087, II.Q (Reference 4) in case of software common-cause failure in the safety I&C system. The DAS is implemented on the platform diverse from the safety I&C system and control system.

The data communication system provides a high-speed and error-free communication path between each system and within a system.

The HSI system is designed in accordance with the human factors engineering (HFE) program to provide reasonable assurance that the HFE design is properly developed and effectively implemented. The HFE program objectives for the NPP design are that the design is human-centered, it incorporates HFE principals and methods, and is developed according to a systematic top-down approach. In accordance with applicable requirements of the HFE process elements, the HFE program plan provides reasonable assurance that the HSI design effectively supports the operator and allows consequential operator errors to be minimized. The HFE program is in effect at least from the start of the design cycle through completion of initial plant startup test program to conform with NUREG-0711 (Reference 5).

1.5.5 References

1. IEEE Std. 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, 1991.

2. IEEE Std. 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, 2003

3. Regulatory Guide 1.152, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants," Rev.3, U.S. Nuclear Regulatory Commission, July 2011.

4. SECY 93-087, II.Q, "Defense against Common-Mode Failures in Digital Instrumentation and Control Systems, " U.S. Nuclear Regulatory Commission, July 1993.

5. NUREG-0711, "Human Factors Engineering Program Review Model," Rev.3, U.S. Nuclear Regulatory Commission, November 2012.

1.6 Material Referenced

Tables 1.6-1 and 1.6-2 contain lists of topical reports and technical reports, respectively, that are incorporated by reference in this document.

Table 1.6-1

List of Topical Reports

Report Number[1]	Title	DCD Tier 2 Section
APR1400-F-A-TR-12004-P APR1400-F-A-TR-12004-NP	Realistic Evaluation Methodology for Large-Break LOCA of the APR1400, Rev. 0	6.2.1.5.1, 15.6
APR1400-F-C-TR-12002-P APR1400-F-C-TR-12002-NP	KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design, Rev. 0	4.4, 15.0~15.6
APR1400-F-M-TR-13001-P APR1400-F-M-TR-13001-NP	PLUS7 Fuel Design for the APR1400, Rev. 0	4.2, 4.4
APR1400-K-Q-TR-11005-NP	QAPD for the APR1400 DC, Rev. 4	17.1, 17.2, 17.3, 17.5
APR1400-Z-M-TR-12003-P APR1400-Z-M-TR-12003-NP	Fluidic Device Design for the APR1400, Rev. 0	6.3.2.1

(1) P – denotes document is proprietary.
NP – denotes document is non-proprietary.

Table 1.6-2 (1 of 2)

List of Technical Reports

Report Number [1]	Title	DCD Tier 2 Section
APR1400-E-A-NR-14002-P-SGI	Physical Security Design Features	13.6.2
APR1400-E-I-NR-14001-P APR1400-E-I-NR-14001-NP	Human Factors Engineering Program Plan	18.1
APR1400-E-I-NR-14002-P APR1400-E-I-NR-14002-NP	Operating Experience Review Implementation Plan	18.2
APR1400-E-I-NR-14003-P APR1400-E-I-NR-14003-NP	Functional Requirements Analysis and Function Allocation Implementation Plan	18.3
APR1400-E-I-NR-14004-P APR1400-E-I-NR-14004-NP	Task Analysis Implementation Plan	18.4
APR1400-E-I-NR-14006-P APR1400-E-I-NR-14006-NP	Treatment of Important Human Actions Implementation Plan	18.6
APR1400-E-I-NR-14007-P APR1400-E-I-NR-14007-NP	Human-System Interface Design Implementation Plan	18.7
APR1400-E-I-NR-14008-P APR1400-E-I-NR-14008-NP	Human Factors Verification and Validation Implementation Plan	18.10
APR1400-E-N-NR-14001-P APR1400-E-N-NR-14001-NP	Design Features to Address GSI-191	6.2.1.1.2.2 6.8.2.2.1
APR1400-E-P-NR-14005-P APR1400-E-P-NR-14005-NP	Evaluations and Design Enhancements to Incorporate Lessons Learned from FUKUSHIMA DAI-CHI Nuclear Accident	1.9.6, 19.3
APR1400-E-S-NR-14004-P APR1400-E-S-NR-14004-NP	Evaluation of Effects of HRHF Response Spectra on SSCs	3.7B.1
APR1400-E-S-NR-14005-P APR1400-E-S-NR-14005-NP	Evaluation of Structure-Soil-Structure Interaction (SSSI) Effects	3.7.2.8
APR1400-E-S-NR-14006-P APR1400-E-S-NR-14006-NP	Stability Check for NI Common Basemat	3.8.5.4.3
APR1400-F-A-NR-14001-P APR1400-F-A-NR-14001-NP	Small Break LOCA Evaluation Model	15.6

Table 1.6-2 (2 of 2)

Report Number [(1)]	Title	DCD Tier 2 Section
APR1400-F-A-NR-14002-P APR1400-F-A-NR-14002-NP	The Effect of Thermal Conductivity Degradation on APR1400 Design and Safety Analyses	15.4 15.6
APR1400-F-A-NR-14003-P APR1400-F-A-NR-14003-NP	Post-LOCA Long Term Cooling Evaluation Model	15.6
APR1400-H-N-NR-14012-P APR1400-H-N-NR-14012-NP	Mechanical Analysis for New and Spent Fuel Storage Racks	9.1.2
APR1400-K-I-NR-14005-P APR1400-K-I-NR-14005-NP	Staffing and Qualifications Implementation Plan	18.5
APR1400-K-I-NR-14009-P APR1400-K-I-NR-14009-NP	Design Implementation Plan	18.11
APR1400-Z-A-NR-14006-P APR1400-Z-A-NR-14006-NP	Non-LOCA Safety Analysis Methodology	15.0.2
APR1400-Z-A-NR-14007-P APR1400-Z-A-NR-14007-NP	LOCA Mass and Energy Release Methodology	6.2.1.3
APR1400-Z-J-NR-14001-P APR1400-Z-J-NR-14001-NP	Safety I&C System	7.1, 7.2, 7.3, 7.4, 7.5, 7.8, 7.9
APR1400-Z-J-NR-14003-P APR1400-Z-J-NR-14003-NP	Software Program Manual	7.1.4, 7.2.2.2, 7.3.1
APR1400-Z-J-NR-14004-P APR1400-Z-J-NR-14004-NP	Uncertainty Methodology and Application for Instrumentation	7.2.2.7, 7.3.2.7
APR1400-Z-J-NR-14005-P APR1400-Z-J-NR-14005-NP	Setpoint Methodology for Plant Protection System	7.2.2.7, 7.3.2.7
APR1400-Z-M-NR-14008-P APR1400-Z-M-NR-14008-NP	Pressure-Temperature Limits Methodology for RCS Heatup and Cooldown	5.2, 5.3

(1) P – denotes document is proprietary.
NP – denotes document is non-proprietary.

1.7 Drawings and Diagrams

1.7.1 Electrical, Instrumentation, and Control Drawings

The line drawings of offsite and onsite electrical systems are provided in Chapter 8. Instrumentation, control, and other electrical drawings are provided in Chapter 7. Table 1.7-1 is a list of the electrical, instrumentation, and control drawings in Chapters 7 and 8.

1.7.2 Flow Diagrams

The flow diagrams are listed in Table 1.7-2. The symbols used in the flow diagrams are defined in Figure 1.7-1.

Table 1.7-1 (1 of 4)

Safety-Related Electrical, Instrumentation, and Control Drawings

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7.2-7	Core Protection Calculator System Functional Block Diagram	7.2
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7.2-31	Functional Logic Diagram for High Logarithmic Power Level Operating Bypass Permissive	7.2
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7.3-2	Block Diagram of the ESF-CCS	7.3
7.3-3	ESF-CCS Simplified Logic Diagram for 2-out-of-4 Actuation	7.3
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7.3-5	ESFAS Functional Logic (CSAS)	7.3
7.3-6	ESFAS Functional Logic (CIAS)	7.3
7.3-7	ESFAS Functional Logic (MSIS)	7.3
7.3-8	ESFAS Functional Logic (AFAS-1 and AFAS-2)	7.3
7.3-9	ESFAS Functional Logic (FHEVAS)	7.3
7.3-10	ESFAS Functional Logic (CPIAS)	7.3
7.3-11	ESFAS Functional Logic (CREVAS)	7.3
7.3-12	CLD for a Solenoid-Operated Valve	7.3
7.3-13	CLD for a Modulating Valve with Solenoid Operator	7.3
7.3-14	Motor-Operated Valve Functional Interface Design	7.3
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7.3-16	CLD for a Throttling Motor-Operated Valve	7.3
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7.3-18	CLD for a Circuit Breaker Operated Component	7.3
7.3-19	CLD for a Modulating Component	7.3
7.3-20	CLD for a Electro-Hydraulic Motor Damper	7.3

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5.1.2-2	Reactor Coolant Pump Flow Diagram	5.1
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5.4.12-1	Reactor Coolant Gas Vent System Flow Diagram	5.4
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6.3.2-1	Safety Injection / Shutdown Cooling System Flow Diagram	6.3
6.4-1	Control Room Envelope Flow Diagram	6.4
6.4-2	Control Room HVAC System Flow Diagram (Normal Mode)	6.4
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Figure No.	Title	DCD Tier 2 Section
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9.5.4-1	Diesel Fuel Oil Transfer System Flow Diagram	9.5
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Figure No.	Title	DCD Tier 2 Section
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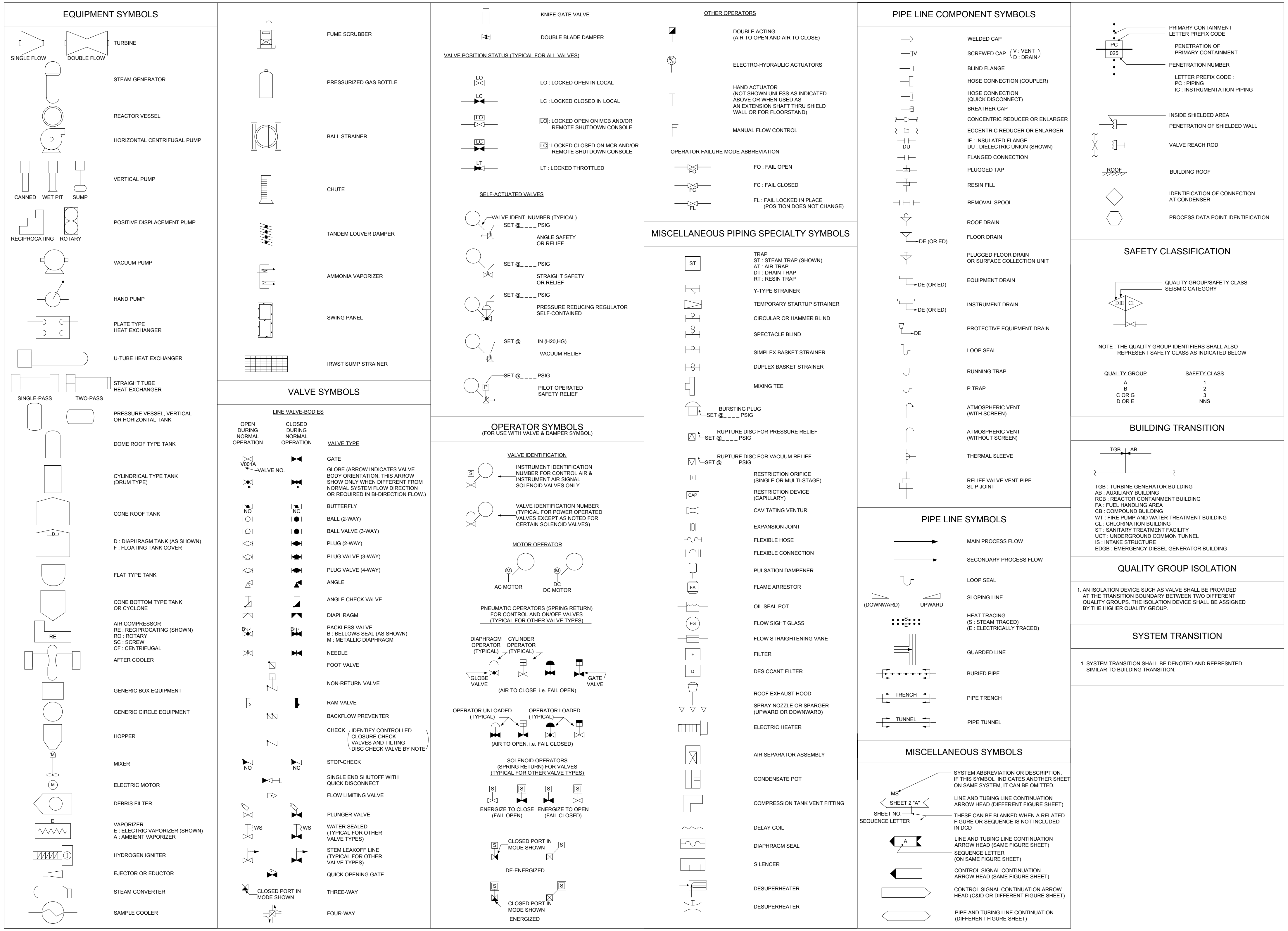

Figure 1.7-1 Flow Diagram Symbols and Legend (1 of 4)

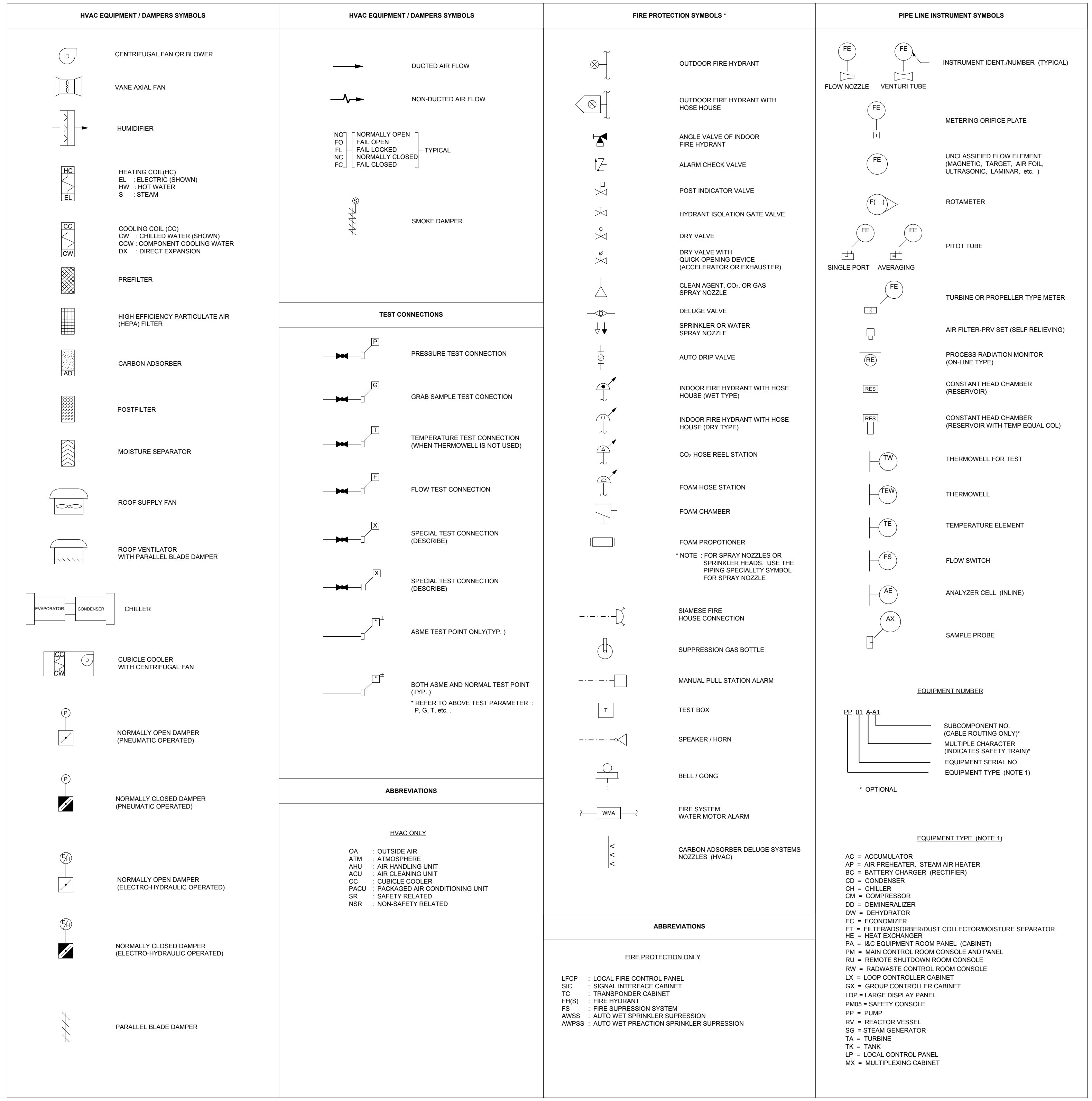

Figure 1.7-1 Flow Diagram Symbols and Legend (2 of 4)

INSTRUMENT & CONTROL DEVICE SYMBOLS

- ** *** SHARED HUMAN-SYSTEM INTERFACE (HSI) (HAND SWITCH, INDICATOR, CONTROL, ALARM, RECORDER) WITHIN POWER BLOCK (1)
- ** SHARED HSI (HAND SWITCH, INDICATOR,CONTROL, ALARM, RECORDER) OUT OF POWER BLOCK (1)
- ** MAIN CONTROL ROOM PANEL MOUNTED DEVICE (FIXED)
  - * 1PM05 PANEL NUMBER (TYPICAL)
- CONTROL DEVICE AND / OR CCS CABINET MOUNTED DEVICE INSTALLED WITHIN POWER BLOCK
  - * 1PA01 PANEL NUMBER (TYPICAL)
- CONTROL DEVICE AND / OR CCS CABINET MOUNTED DEVICE INSTALLED OUT OF POWER BLOCK
- LOCALLY MOUNTED DEVICE NOT ON INSTRUMENT PANEL OR ASSEMBLY
- LOCAL INSTRUMENT PANEL MOUNTED DEVICE
  - * 1LP01 PANEL NUMBER (TYPICAL)
- MULTIPLE DEVICES IN ONE ASSEMBLY (i.e. TEMPERATURE ELEMENT IN THERMOWELL)

* PLANT UNIT NUMBER NOT REQUIRED IF SAME AS DRAWING PLANT UNIT NUMBER

** THE INDICATED POWER SEGREGATION REPRESENTS THE POWER SEGREGATION OF COMPONENT BEING CONTROLLED BY THE SHARED CONTROL.

*** TYPE OF HSI DEVICE SUCH AS QIAS-N, QIAS-P OR LDP IS PRESENTED AS SUFFIX. HOWEVER, THE TYPE OF HSI DEVICE IS NOT REQUIRED IF THE SYMBOL IS USED FOR THE SHARED CONTROL.

THE SHARED SYMBOL WITHOUT SUFFIX MEANS THIS PARAMETER IS ONLY REPRESENTED ON THE INFORMATION FPD. THE QIAS-N MEANS THIS PARAMETER IS REPRESENTED ON BOTH INFORMATION FPD AND QIAS-N FPD. THE LDP MEANS THIS PARAMETER IS REPRESENTED ON THE INFORMATION FPD, QIAS-N AND LDP. FOR A MAN SWITCH, ONLY THE COMPONENT STATUS IS DISPLAYED ON QIAS-N AND LDP.

(1) COMPOUND BUILDING & SWITCH YARD ARE EXCLUDED FROM POWER BLOCK AREA.

COMPUTER INPUT - OUTPUT SIGNAL NUMBER

ALL OF INPUT POINTS TO CCS ARE REGARED AS THE COMPUTER INPUT DATA FOR IPS OR QIAS-N.

LOCAL PANEL INPUT SYMBOL

TO CCS	INPUT SIGNAL FROM LOCAL PANEL TO COMPUTER THROUGH CCS
TO QIAS-P	INPUT SIGNAL TO QIAS-P COMPUTER
TO DIS	INPUT SIGNAL TO DIS COMPUTER

SYSTEM LEVEL ACTUATION SYMBOL

ESF-SIAS START	SYSTEM LEVEL ESF ACTUATION SIGNAL
DMA-SIAS START	DIVERSE MANUAL ACTUATION SIGNAL

CONTROL & INSTRUMENTATION FUNCTION SYMBOLS & CODES

FUNCTION SYMBOLS

- ELECTRICAL SIGNAL BETWEEN LOCAL & MAIN CONTROL ROOM / I & C EQUIPMENT ROOM FUNCTIONS CONNECTION FOR THE PROCESS VARIABLE (TYPICAL)
- LOCAL FUNCTION

FUNCTION SYMBOL CODES

PTIS 001 XCIR O_2

- VARIABLE (MEASURED)
- LOCAL FUNCTION OF VARIABLE
- INSTRUMENT NUMBER
- CLARIFYING SYMBOL PER ABOVE SYMBOL LIST (IF REQUIRED)
- FUNCTION IN MAIN CONTROL ROOM AREA
- VARIABLE IN MAIN CONTROL ROOM AREA (USED WHEN NO LOCAL VARIABLE EXISTS)

PF-TIS 001 CIR

- MULTIPLE VARIABLE WHEN APPLICABLE
- DASH USED WHEN MULTIPLE VARIABLES EXIST
- ALTERNATE WHEN REQUIRED FOR MULTIPLE VARIABLES

INSTRUMENT CLARIFYING SYMBOL LIST

CLARIFYING SYMBOL LIST

A	ANALOG SIGNAL
ADD	ADD
AVG	AVERAGE
BI	BIAS
C	CONDUCTIVITY
CC	CATION CONDUCTIVITY
CL-	CHLORIDE
CL_2	CHLORINE
CO_2	CARBON DIOXIDE
D	DIGITAL
DIFF	SUBTRACT
DIR	DIRECT ACTING
E/I	CONVERTER (VOLTAGE TO CURRENT)
FC	FAIL CLOSED
F/H	FOLLOW / HOLD
FI	FAIL INTERMEDIATE
FL	FAIL LOCK
FO	FAIL OPEN
H_2	HYDROGEN
HE	HELIUM
HHL	HIGH HIGH LIMIT
HL	HIGH LIMIT
HP	HIGH PASS
I/E	CONVERTER (CURRENT TO VOLTAGE)
INT	INTEGRATOR
I/P	CONVERTER (ELECTRIC CURRENT TO PNEUMATIC)
ISO	ISOLATOR
LL	LOW LIMIT
LLL	LOW LOW LIMIT
LP	LOW PASS
M	MANUAL
MAX	MAXIMUM
M/A	MANUAL / AUTOMATIC
MIN	MINIMUM
N_2	NITROGEN
NA	SODIUM
NAK	SODIUM POTASSIUM
N_2H_2	HYDRAZINE
O_2	OXYGEN
pH	pH ANALYSIS
RES	RESET
REN	REVERSE ACTING
SC	SPECIFIC CONDUCTIVITY
SI	SILICA
SM	SMOKE OR IONIZATION
SQRT	SQUARE ROOT
TURB	TURBIDITY
VOL	VOLUME
X	MULTIPLY
REF	REFRIGERANT
DC	DEDICATED CONTROL
IPS	INFORMATION PROCESSING SYSTEM
QIAS-N	QUALIFIED INDICATION AND ALARM SYSTEM - NON-SAFETY
QIAS-P	QUALIFIED INDICATION AND ALARM SYSTEM - P
SC	SOFT CONTROL

INSTRUMENT TYPE CODES

FIRST LETTER VARIABLE

(MEASURED OR INITIATING)

A	ANALYSIS
B	BURNER
C	CONDUCTIVITY
D	DENSITY
E	VOLTAGE
F	FLOW
H	HAND (MANUAL)
I	CURRENT (ELEC.)
J	POWER (ELEC.)
K	TIME
L	LEVEL(LIQUID OR SOLID)
M	MOISTURE (HUMIDITY)
N	NEUTRON FLUX
P	PRESSURE OR VACUUM
PD	DIFFERENTIAL PRESSURE
R	RADIOACTIVITY
S	SPEED OR FREQUENCY
T	TEMPERATURE
U	MULTIVARIABLE
V	VIBRATION
W	WEIGHT OR FORCE
X	SPECIAL (REFER TO CLARIFYING SYMBOL LIST)
Y	STRAIN
Z	POSITION

SECOND OR SUCCEEDING LETTERS INSTRUMENT FUNCTION

A	ALARM
AC	ALARM CLOSED
AH	ALARM HIGH
AHH	ALARM HIGH HIGH
AHL	ALARM HIGH OR LOW
AL	ALARM LOW
ALL	ALARM LOW LOW
AO	ALARM OPEN
C	CONTROLLER (BLIND)
E	PRIMARY ELEMENT
EW	PRIMARY ELEMENT WELL
G	GLASS
I	INDICATOR
IC	INDICATING CONTROLLER
IK	INDICATING CONTROL STATION (MANUAL/AUTO)
IR	INDICATING RECORDER
IS	INDICATING SWITCH
IT	INDICATING TRANSMITTER
IY	INDICATING RELAY
JR	MULTIPOINT RECORDER (MORE THAN 3 PENS)
K	CONTROL STATION-VARIABLE TYPE (MANUAL/AUTO) WITHOUT CONTROLLER
L	LIGHT (PILOT)
Q	INTEGRATOR
R	RECORDER
RC	RECORDER CONTROLLER
S	SWITCH
T	TRANSMITTER
U	MULTIFUNCTION
W	WELL
X	UNCLASSIFIED
Y	RELAY, SIGNAL CONVERTER, COMPUTING SOLENOID, ETC.
Z	POWER POSITIONER / CONTROL DRIVE (EXCEPT VALVE MOUNTED)

SYSTEM CLARIFYING SYMBOL LIST

CLARIFYING SYMBOL LIST

AF	AUXILIARY FEEDWATER SYSTEM
AS	AUXILIARY STEAM
CA	CONDENSER VACUUM SYSTEM
CC	COMPONENT COOLING WATER SYSTEM
CD	CONDENSATE SYSTEM
CM	CONTAINMENT MONITORING SYSTEM
CP	CONDENSATE POLISHING SYSTEM
CS	CONTAINMENT SPRAY SYSTEM
CT	CONDENSATE STORAGE AND TRANSFER SYSTEM
CV	CHEMICAL AND VOLUME CONTROL SYSTEM
CW	CIRCULATING WATER SYSTEM
DA	GAS TURBINE GENERATOR FACILITY SYSTEM
DE	RADIOACTIVE DRAIN SYSTEM
DG	EMERGENCY DIESEL GENERATOR SYSTEM
DO	DIESEL FUEL OIL TRANSFER SYSTEM
ED	NON-RADIOACTIVE EQUIPMENT VENT & DRAIN SYSTEM
FC	SPENT FUEL POOL COOLING AND CLEANUP SYSTEM
FP	FIRE PROTECTION SYSTEM
FW	FEEDWATER SYSTEM
GW	GASEOUS RADWASTE MANAGEMENT SYSTEM
IA	INSTRUMENT AIR SYSTEM
IW	IN-CONTAINMENT WATER STORAGE SYSTEM
MS	MAIN STEAM SYSTEM
PR	PROCESS RADIATION MONITORING SYSTEM
PS	PROCESS SAMPLING SYSTEM
RC	REACTOR COOLANT SYSTEM
RG	REACTOR COOLANT GAS VENT SYSTEM
SA	SERVICE AIR SYSTEM
SD	STEAM GENERATOR BLOWDOWN SYSTEM
SI	SAFETY INJECTION / SHUTDOWN COOLING SYSTEM
SX	ESSENTIAL SERVICE WATER SYSTEM
VB	COMPOUND BUILDING HVAC SYSTEM
VC	CONTROL ROOM HVAC SYSTEM
VD	EMERGENCY DIESEL GENERATOR AREA HVAC SYSTEM
VE	ELECTRICAL AND I&C EQUIPMENT AREAS HVAC SYSTEM
VF	FUEL HANDLING AREA HVAC SYSTEM
VK	AUXILIARY BUILDING CONTROLLED AREA HVAC SYSTEM
VO	AUXILIARY BUILDING CLEAN AREA HVAC SYSTEM
VP	REACTOR CONTAINMENT BUILDING HVAC SYSTEM
VQ	REACTOR CONTAINMENT BUILDING PURGE SYSTEM
VT	TURBINE GENERATOR BUILDING HVAC SYSTEM
WD	DOMESTIC WATER SYSTEM
WH	TURBINE GENERATOR BUILDING OPEN COOLING SYSTEM
WI	PLANT CHILLED WATER SYSTEM
WO	ESSENTIAL CHILLED WATER SYSTEM
WT	TURBINE GENERATOR BUILDING COMPONENT COOLING SYSTEM
WV	LIQUID RADWASTE MANAGEMENT SYSTEM
WX	SOLID RADWASTE MANAGEMENT SYSTEM

Figure 1.7-1 Flow Diagram Symbols and Legend (3 of 4)

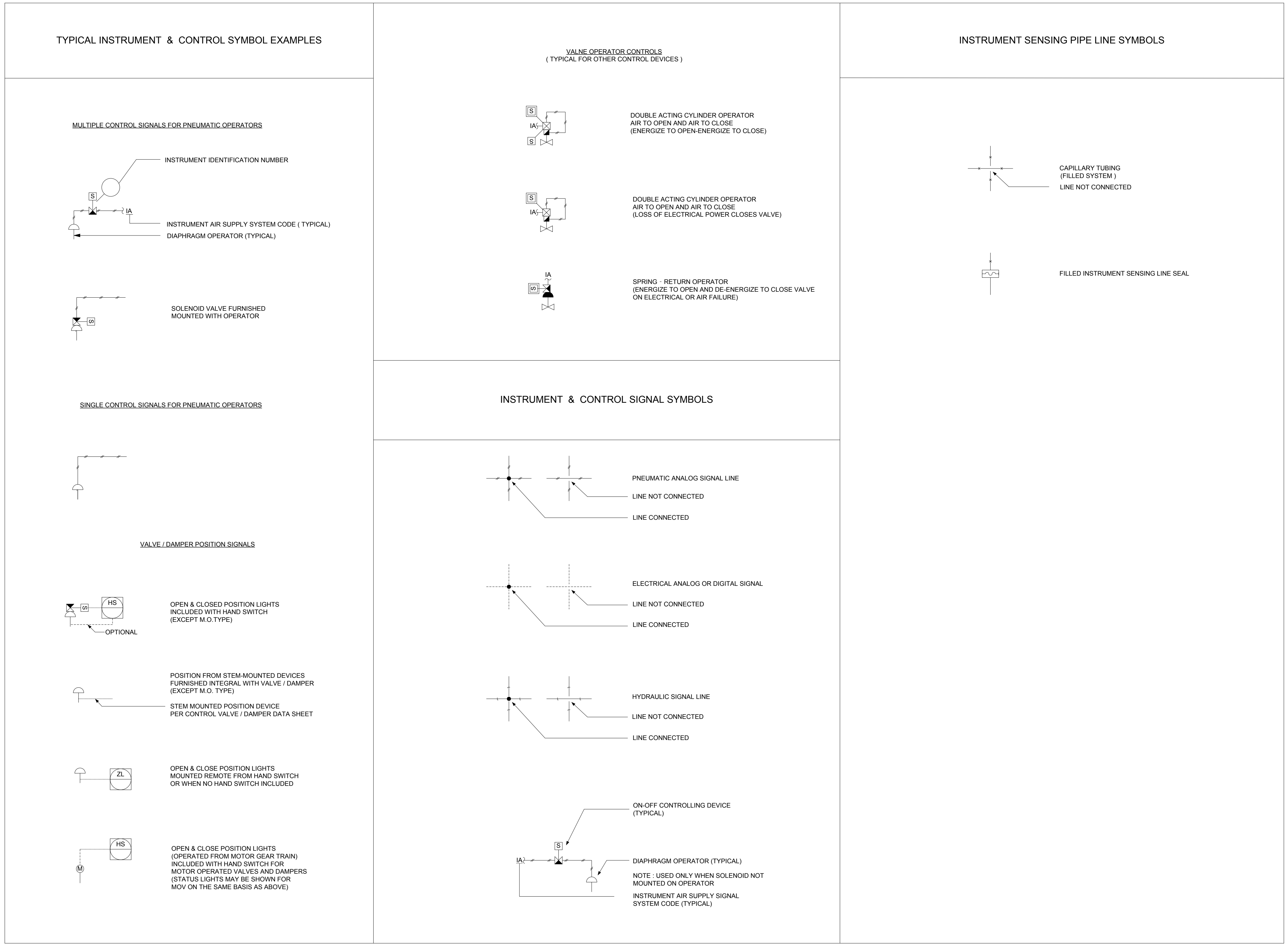

Figure 1.7-1 Flow Diagram Symbols and Legend (4 of 4)

1.8 Interfaces with Standard Designs

The APR1400 design includes an essentially complete nuclear plant, but does not include the structures, systems, and components (SSCs) that are part of the site-specific designs. Section 1.8 provides a list of the interface requirements for SSCs within the scope of the APR1400 that are required by 10 CFR 52.47(a) (Reference 1).

A standard site plot of the APR1400 is provided in Figure 1.2-1. The plot shows the scope of the design certification application. To provide reasonable assurance that the site-specific SSCs are compatible with the APR1400 design, interface requirements for site-specific SSCs to be satisfied by the combined license (COL) applicant are provided in the sections referenced in Table 1.8-1. In general, interface requirements for COL applicant-supplied SSCs that relate to a specific mechanical or electrical system are addressed in the appropriate chapter. The conceptual design information (CDI) for those portions of the plant for which the application does not seek certification is indicated by double brackets in the text and tables and cloud marks in the figures. Safety-significant interface requirements for site-specific SSCs are specified in Tier 1 of the DCD.

Table 1.8-1 is an index of all sections of this document that contain interface requirements. The COL applicant is to describe how the site-specific interface requirements are met.

Site characteristic assumptions on which the APR1400 design is based are presented in Chapter 2. The site characteristics are to be compatible with the APR1400 design envelopes but are not considered interface requirements as used in 10 CFR 52.47(a).

Table 1.8-2 presents the list of all COL information items. The COL applicant is to identify how each COL information item is addressed.

1.8.1 Combined License Information

COL 1.8(1) The COL applicant is to describe how site-specific interface requirements are met.

COL 1.8(2) The COL applicant is to identify how each COL information item is addressed.

1.8.2 References

1. 10 CFR 52.47(a), "Contents of Applications; Technical Information."

Table 1.8-1 (1 of 2)

Index of System, Structure, or Component Interface Requirements for APR1400

System, Structure, or Component	Interface Type	DCD Tier 2 Section
Structures		
Switchyard	COL	8.2
Emergency operations facility	COL	13.3.3.2
Ultimate heat sink, including ESWS intake/discharge	CDI	9.2.5.2
Domestic water and sanitary system structure	CDI	9.2.4
Circulating water pump house	CDI	10.4.5.2
Normal plant heat sink, including CW system intake/discharge	CDI	10.4.5
Systems		
Offsite power system, including switchyard	COL	8.2
Domestic water and sanitary systems, including sanitary water treatment facility	CDI	9.2.4
Security system	COL	13.6.1
Communication system (offsite)	COL	9.5.2.2.2
UHS	CDI	9.2.5

Table 1.8-1 (2 of 2)

Structure, System or Component	Interface Type	DCD Tier 2 Section
Components		
UHS cooling tower	CDI	9.2.5.2.2.1
UHS piping, valves, and fittings	CDI	9.2.5.2.2.2
UHS cooling tower basin	CDI	9.2.5.2.2.3
UHS cooling tower basin screens	CDI	9.2.5.2.2.4
Circulating water pumps	CDI	10.4.5.2.1
Cooling tower	CDI	10.4.5.2.3
Cooling tower basin	CDI	10.4.5.2.3
Cooling tower basin screen	CDI	10.4.5.2.3
Cooling tower makeup and blowdown pump	CDI	10.4.5.2.3
ESW blowdown piping	CDI	9.2.1.2.1
Condenser vacuum pressure of a high pressure alarm and turbine trip	CDI	10.4.1.5, 10.4.2.2.2
Cooling tower chemical injection system	CDI	10.4.5.2.3
Cation-bed ion exchanger vessels	CDI	Table 10.4.6-1
Mixed-bed ion exchanger vessels	CDI	Table 10.4.6-1
Spent resin holding tanks	CDI	Table 10.4.6-1
Resin holding tank	CDI	Table 10.4.6-1
Resin mixing and holding tank	CDI	Table 10.4.6-1
Resin traps	CDI	Table 10.4.6-1

Table 1.8-2 (1 of 29)

Combined License Information Items

Item No.	Description
COL 1.1(1)	The COL applicant that references the APR1400 is to identify the actual plant site location.
COL 1.1(2)	The COL applicant that references the APR1400 is to provide estimated schedules for the completion of construction and the start of commercial operation.
COL 1.2(1)	The COL applicant is to prepare a complete and detailed site plan.
COL 1.4(1)	The COL applicant that references the APR1400 design certification is to identify major agents, contractors, and participants for the construction and operation of the nuclear power plant.
COL 1.8(1)	The COL applicant is to describe how site-specific interface requirements are met.
COL 1.8(2)	The COL applicant is to identify how each COL information item is addressed.
COL 1.9(1)	The COL applicant is to provide an evaluation of the conformance with the regulatory criteria for the site-specific portions and operational aspects of the facility.
COL 2.0(1)	The COL applicant is to demonstrate that the APR1400 design meets the requirements imposed by the site-specific parameters and conforms with all design commitments and acceptance criteria if the characteristics of the site fall outside the assumed site parameters in Table 2.0-1.
COL 2.1(1)	The COL applicant is to provide site-specific information on the site location and description of the site, exclusion authority and control, and population distribution as stated in NRC RG 1.206.
COL 2.2(1)	The COL applicant is to provide site-specific information on nearby industrial, transportation, and military facilities as required in NRC RG 1.206.
COL 2.2(2)	The COL applicant is to identify the DBE caused by nearby industrial, transportation, and military facilities and determine its design parameters.
COL 2.3(1)	The COL applicant is to provide site-specific information on meteorology including regional climatology, local meteorology, onsite meteorological measurement program, estimated short-term atmospheric dispersion for accident release, and long-term atmospheric dispersion estimates for routine release as addressed in NRC RG 1.206.
COL 2.3(2)	The COL applicant is to perform the radiological consequence analysis and demonstrate that the related dose limits specified in 10 CFR 50.34 and 10 CFR Part 50 Appendix I are not exceeded, if the site-specific χ/Q values exceed the bounding values described in Tables 2.3-1 to 2.3-12.
COL 2.4(1)	The COL applicant is to provide the site-specific hydrologic information on probable maximum precipitation (PMP), probable maximum flood (PMF) on streams and rivers, potential dam failures, probable maximum surge and seiche flooding, probable maximum tsunami hazards, ice effects, cooling water canals and reservoirs, channel diversions, flood protection requirements, low water considerations, ground water, potential accidental release of liquid effluents in ground and surface water, and Technical Specifications and emergency operation requirements in accordance with NRC RG 1.206, NRC RG 1.59, and NRC JLD-ISG-2012-06.

Table 1.8-2 (2 of 29)

Item No.	Description
COL 2.5(1)	The COL applicant is to provide the site-specific information on geology, seismology, and geotechnical engineering as required in NRC RG 1.206.
COL 2.5(2)	The COL applicant is to confirm that the foundation input response spectra (FIRS) of the nuclear island are completely enveloped by the CSDRS-compatible free-field response motions at the bottom elevation of the nuclear island for a site with the low-strain shear wave velocity greater than 304.8 m/s (1,000 ft/s) at the finished grade in the free field. Alternately, the COL applicant is to confirm that FIRS of the nuclear island are completely enveloped by the CSDRS for a hard rock site with a low-strain shear wave velocity of supporting medium for the nuclear island greater than 2,804 m/s (9,200 ft/s).
COL 2.5(3)	The COL applicant is to confirm that the lower bound of the site-specific strain-compatible soil profile for a soil site is greater than the lower bound of the generic strain-compatible soil profiles used in the APR1400 seismic analyses.
COL 2.5(4)	The COL applicant is to confirm that the site-specific GMRS determined at the finished grade are completely enveloped by the hard rock high frequency (HRHF) response spectra for a site with a low-strain shear wave velocity of supporting medium for the nuclear island higher than 1,494 m/s (4,900 ft/s) overlaying a hard rock with a low-strain shear wave velocity greater than 2,804 m/s (9,200 ft/s).
COL 2.5(5)	The COL applicant is to perform a site-specific seismic analysis to generate in-structure response spectra at key locations using the procedure described in Appendix 3.7A if COL 2.5(2) and COL 2.5(3) above are not met. In addition, the COL applicant is to confirm that the site-specific in-structure response spectra so generated are enveloped by the corresponding in-structure response spectra provided in Appendix 3.7A.
COL 2.5(6)	The COL applicant is to perform a site-specific seismic response analysis using the procedure described in Appendix 3.7B and the EPRI White Paper, "Seismic Screening of Components Sensitive to High Frequency Vibratory Motions," if COL 2.5(4) is not met.
COL 2.5(7)	The COL applicant is to perform an evaluation of the subsurface conditions within the standard plant structure footprint based on the geologic investigation in accordance with NRC RG 1.132.
COL 2.5(8)	The COL applicant is to confirm that the dynamic properties of structural fill granular to be used in construction of the APR1400 seismic Category I structures satisfy the requirements of structural fill granular provided in Table 2.0-1.
COL 3.2(1)	The COL applicant is to identify the seismic classification of site-specific SSCs that should be designed to withstand the effects of the SSE.
COL 3.2(2)	The COL applicant is to identify the quality group classification of site-specific systems and components and their applicable codes and standards.
COL 3.3(1)	The COL applicant is to demonstrate that the site-specific design wind speed is bounded by the design wind speed of 64.8 m/s (145 mph).
COL 3.3(2)	The COL applicant is to demonstrate that the site-specific seismic Category II structures adjacent to the seismic Category I structures are designed to meet the provisions described in Subsection 3.3.1.2.
COL 3.3(3)	The COL applicant is to provide reasonable assurance that site-specific structures and components not designed for the extreme wind loads do not impact either the function or integrity of adjacent seismic Category I SSCs.

Table 1.8-2 (3 of 29)

Item No.	Description
COL 3.4(1)	The COL applicant is to provide site-specific information on protection measures for the design-basis flood, as required in Subsection 2.4.10.
COL 3.4(2)	The COL applicant is to provide flooding analysis with flood protection and mitigation features from internal flooding for the CCW Heat Exchanger Building and ESW Building.
COL 3.4(3)	The COL applicant is to confirm that the potential site-specific external flooding events are bounded by design-basis flood values or otherwise demonstrate that the design is acceptable.
COL 3.4(4)	The COL applicant is to identify any site-specific physical models that could be used to predict prototype performance of hydraulic structures and systems.
COL 3.5(1)	The COL applicant is to provide the procedure for heavy load transfer to strictly limit the transfer route inside and outside containment during plant maintenance and repair periods.
COL 3.5(2)	The COL applicant is to perform an assessment of the orientation of the turbine generator of this and other unit(s) at multi-unit sites for the probability of missile generation using the evaluation of Subsection 3.5.1.3.2 to verify that essential SSCs are outside the low-trajectory turbine missile strike zone.
COL 3.5(3)	The COL applicant is to evaluate site-specific hazards induced by external events that may produce more energetic missiles than tornado or hurricane missiles, and provide reasonable assurance that seismic Category I and II structures are designed to withstand these loads.
COL 3.5(4)	The COL applicant is to evaluate the potential for site proximity explosions and missiles due to train explosions (including rocket effects), truck explosions, ship or barge explosions, industrial facilities, pipeline explosions, or military facilities.
COL 3.5(5)	The COL applicant is to provide justification for the site-specific aircraft hazard and an aircraft hazard analysis in accordance with the requirements of NRC RG 1.206.
COL 3.6(1)	The COL applicant is to identify the site-specific SSCs that are safety related or required for safe shutdown that are located near high- and moderate-energy piping systems and that are susceptible to the consequences of piping failures.
COL 3.6(2)	The COL applicant is to provide a list of site-specific high- and moderate-energy piping systems including layout drawings and protection features and the failure modes and effects analysis for safe shutdown due to the postulated HELBs.
COL 3.6(3)	The COL applicant is to confirm that the bases for the LBB acceptance criteria are satisfied by the final as-built design and materials of the piping systems as site-specific evaluations, and is to provide the information including LBB evaluation report for the verification of LBB analyses.
COL 3.6(4)	The COL applicant is to provide the procedure for initial filling and venting to avoid the known causes for water hammer in DVI line.
COL 3.7(1)	The COL applicant is to determine the site-specific SSE and OBE that are applied to the seismic design of the site-specific seismic Category I and II SSCs and the basis for the plant shutdown. The COL applicant is also to verify the appropriateness of the site-specific SSE and OBE.
COL 3.7(2)	The COL applicant is to confirm that the horizontal components of the SSE site-specific ground motion in the free-field at the foundation level of the structure satisfy a peak ground acceleration of at least 0.1 g.

Table 1.8-2 (4 of 29)

Item No.	Description
COL 3.7(3)	The COL applicant is to provide the seismic design of the seismic Category I SSCs that are not part of the APR1400 standard plant design. The seismic Category I structures are as follows: a. Seismic Category I essential service water building b. Seismic Category I component cooling water heat exchanger building
COL 3.7(4)	The COL applicant is to confirm that the any site-specific non-seismic Category I SSCs are designed not to degrade the function of a seismic Category I SSC to an unacceptable safety level due to their structural failure or interaction.
COL 3.7(5)	The COL applicant is to perform any site-specific seismic design for dams that is required.
COL 3.7(6)	The COL applicant is to perform seismic analysis of buried seismic Category I piping, conduits, and tunnels.
COL 3.7(7)	The COL applicant is to perform seismic analysis for the seismic Category I above-ground tanks.
COL 3.7(8)	The COL applicant that references the APR1400 design certification will determine whether essentially the same seismic response from a given earthquake is expected at each unit in a multi-unit site or each unit is to be provided with a separate set of seismic instruments.
COL 3.7(9)	The COL applicant is to confirm details of the locations of the triaxial time-history accelerograph.
COL 3.7(10)	The COL applicant is to identify the implementation milestones for the seismic instrumentation implementation program based on the discussion in Subsections 3.7.4.1 through 3.7.4.5.
COL 3.7B(1)	The COL applicant is to evaluate the HRHF response spectra.
COL 3.7B(2)	The COL applicant is to evaluate the representative items listed in Table 3.7B-2.
COL 3.8(1)	The COL applicant is to provide the design of site-specific seismic Category I structures such as the essential service water supply structure and the component cooling water heat exchanger building.
COL 3.8(2)	The COL applicant is to identify any applicable site-specific loads such as site proximity explosions and missiles, potential aircraft crashes, and the effects of seiches, surges, waves, and tsunamis.
COL 3.8(3)	The COL applicant is to determine the environmental condition associated with the durability of concrete structures and provide the concrete mix design that prevents concrete degradation including the reactions of sulfate and other chemicals, corrosion of reinforcing bars, and influence of reactive aggregates.
COL 3.8(4)	The COL applicant is to determine construction techniques to minimize the effects of thermal expansion and contraction due to hydration heat, which could result in cracking.
COL 3.8(5)	The COL applicant is to monitor the safety and serviceability of seismic Category I structures during the operation of the plant and provide the appropriate maintenance.
COL 3.8(6)	The COL applicant is to provide reasonable assurance that the design criteria listed in Table 2.0-1 are met or exceeded.

Table 1.8-2 (5 of 29)

Item No.	Description
COL 3.8(7)	The COL applicant is to confirm that uneven settlement due to construction sequence of the NI basemat falls within the values specified in Table 2.0-1.
COL 3.8(8)	The COL applicant is to provide the necessary measures for foundation settlement monitoring considering site-specific conditions.
COL 3.8(9)	The COL applicant is to provide testing and inservice inspection program to examine inaccessible areas of the concrete structure for degradation and to monitor groundwater chemistry.
COL 3.8(10)	The COL application is to provide the following soil information for APR1400 site: 1) Elastic shear modulus and Poisson's ratio of the subsurface soil layers, 2) Consolidation properties including data from one-dimensional consolidation tests (initial void ratio, Cc, Ccr, OCR, and complete e-log p curves) and time-versus-consolidation plots, 3) Moisture content, Atterberg limits, grain size analyses, and soil classification, 4) Construction sequence and loading history, and 5) Excavation and dewatering programs.
COL 3.9(1)	The COL applicant is to provide the inspection results for the APR1400 reactor internals classified as non-prototype Category I in accordance with RG 1.20.
COL 3.9(2)	The COL applicant is to provide a summary of the maximum total stress, deformation, and cumulative usage factor values for each of the component operating conditions for ASME Code Class 1 components except for ASME Code Class 1 nine major components. For those values that differ from the allowable limits by less than 10 percent, the contribution of each loading category (e.g., seismic, deadweight, pressure, and thermal) to the total stress is provided for each maximum stress value identified in this range. The COL applicant is to also provide a summary of the maximum total stress and deformation values for each of the component operating conditions for Class 2 and 3 components required to shut down the reactor or mitigate consequences of a postulated piping failure without offsite power (with identification of those values that differ from the allowable limits by less than 10 percent).
COL 3.9(3)	The COL applicant is to identify the site-specific active pumps.
COL 3.9(4)	The COL applicant is to confirm the type of testing and frequency of site-specific pumps subject to IST in accordance with the ASME Code.
COL 3.9(5)	The COL applicant is to confirm the type of testing and frequency of site-specific valves subject to IST in accordance with the ASME Code.
COL 3.9(6)	The COL applicant is to provide a table listing all safety-related components that use snubbers in their support systems.

Table 1.8-2 (6 of 29)

Item No.	Description
COL 3.10(1)	The COL applicant is to provide documentation that the designs of seismic Category I SSCs are analyzed for OBE, if OBE is higher than 1/3 SSE.
COL 3.10(2)	The COL applicant is to investigate if site-specific spectra generated for the COLA exceed the APR1400 design spectra in the high-frequency range. Accordingly, the COL applicant is to provide reasonable assurance of the functional performance of vibration-sensitive components in the high-frequency range.
COL 3.10(3)	The COL applicant is to develop the equipment seismic qualification files that summarize the component's qualification, including a list of equipment classified as seismic Category I in Table 3.2-1 and seismic qualification summary data sheets (SQSDS) for each piece of safety-related seismic Category I equipment.
COL 3.10(4)	The COL applicant is to perform equipment seismic qualification for seismic Category I equipment and provide milestones and completion dates of equipment seismic qualification program.
COL 3.11(1)	The COL applicant is to identify and qualify the site-specific mechanical, electrical, I&C, and accident monitoring equipment specified in RG 1.97.
COL 3.11(2)	The COL applicant is to document the qualification test results and qualification status in an auditable file for each type of equipment in accordance with the requirements 10 CFR 50.49(j).
COL 3.11(3)	The COL applicant is to describe the EQP implementation milestones based on the APR1400 EQP.
COL 3.11(4)	The COL applicant is to identify the nonmetallic parts of mechanical equipment in procurement process.
COL 3.12(1)	The COL applicant is to prepare design reports for ASME Class 1, 2, and 3 piping system in accordance with ASME Section III.
COL 3.12(2)	The COL applicant is to design the piping exposed to wind and/or tornado, if any, to the plant design basis loads.
COL 3.12(3)	The COL applicant is to perform fatigue evaluations of ASME Class 1 piping.
COL 3.12(4)	The COL applicant is to perform stress evaluations for ASME Class 2 and 3 piping.
COL 3.12(5)	The COL applicant is to perform fatigue evaluations of environmental impact on ASME Class 1 piping, except for the RCS primary loop, using methods acceptable to the NRC at the time of evaluation.

Table 1.8-2 (7 of 29)

Item No.	Description
COL 3.12(6)	The COL applicant is to perform the piping stress analysis including thermal stratification effects on SCS suction line.
COL 3.12(7)	The COL applicant is to determine maximum radial thermal expansion at its design temperature.
COL 3.13(1)	The COL applicant is to maintain quality assurance records including CMTRs on ASME Section III Class 1, 2, and 3 component threaded fasteners in accordance with the requirements of 10 CFR 50.71.
COL 3.13(2)	The COL applicant is to submit the preservice and inservice inspection programs for ASME Section III Class 1, 2, and 3 component threaded fasteners to the NRC prior to performing the inspections.
COL 5.2(1)	The COL applicant is to address the addition of ASME Code cases that are approved in NRC RG 1.84.
COL 5.2(2)	The COL applicant is to address the ASME Code cases, which are invoked for the ISI program of specific plant.
COL 5.2(3)	The COL applicant is to address the Code cases invoked for operation and maintenance activities.
COL 5.2(4)	The COL applicant is to address the material specifications, which are not shown in Table 5.2-2, as necessary.
COL 5.2(5)	The COL applicant is to specify the version of EPRI's, "Primary Water Chemistry Guidelines," that will be implemented.
COL 5.2(6)	The COL applicant is to address the actual, as-procured, fracture toughness data of the RCPB materials to the staff at a predetermined time by an appropriate method.
COL 5.2(7)	The COL applicant is to submit the actual, as-procured yield strength of the austenitic stainless steel materials used in RCPB to the staff at a predetermined time agreed-upon by the regulatory body.

Table 1.8-2 (8 of 29)

Item No.	Description
COL 5.2(8)	The COL applicant is to provide and develop the implementation milestones for the inservice inspection and testing program for the RCPB, in accordance with ASME Code Section XI and 10 CFR 50.55a.
COL 5.2(9)	The COL applicant is to address the provisions to accessibility of Class 1 components for ISI if the design of the APR1400 Class 1 component is changed from the DCD design.
COL 5.2(10)	The COL applicant is to provide the list of Code exemptions in the ISI program of the specific plants, if it exists.
COL 5.2(11)	The COL applicant is to prepare and provide any requests for relief from the ASME Code requirements that are impracticable as a result of limitations of component design, geometry, or materials of construction for the specific plants, if necessary. The request will contain the information on applicable Code requirements, alternative ISI method, and justification.
COL 5.2(12)	The COL applicant may invoke ASME Code Cases listed in NRC RG 1.147 for the ISI program.
COL 5.2(13)	The COL applicant is to prepare and implement a boric acid corrosion (BAC) prevention program compliant with Generic Letter 88-05.
COL 5.2(14)	The COL applicant is to prepare the preservice inspection and testing program.
COL 5.2(15)	The COL applicant is to address and develop milestones for preparation and implementation of the procedure for operator responses to prolonged low level leakage.
COL 5.3(1)	The COL applicant is to provide a reactor vessel material surveillance program for a specific plant.
COL 5.3(2)	The COL applicant is to develop P-T limit curves based on plant-specific data.
COL 5.3(3)	The COL applicant is to verify the RT_{PTS} value and the USE at EOL based on plant-specific material property and neutron fluences.
COL 5.3(4)	The COL applicant is to provide and develop the inservice inspection and testing program for the RCPB, in accordance with ASME Section XI and 10 CFR 50.55a.
COL 5.4(1)	The COL applicant is to prepare operational procedures and maintenance programs as related to leak detection and contamination control of RCS.
COL 5.4(2)	The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations of RCS.
COL 5.4(3)	The COL applicant is to prepare operational procedures and maintenance programs as related to leak detection and contamination control of SCS.
COL 5.4(4)	The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations of SCS.
COL 5.4(5)	The COL applicant is to verify the as-built RV support material properties and 60-year neutron fluence.

Table 1.8-2 (9 of 29)

Item No.	Description
COL 6.1(1)	The COL applicant is to identify the implementation milestones for the coatings program.
COL 6.2(1)	The COL applicant is to identify the implementation milestone for the CILRT program.
COL 6.3(1)	The COL applicant is to prepare operational procedures and maintenance programs as related to leak detection and contamination control.
COL 6.3(2)	The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations.
COL 6.4(1)	The COL applicant is to provide automatic and manual operating procedures for the control room HVAC system, which are required in the event of a postulated toxic gas release.
COL 6.4(2)	The COL applicant is to provide the details of specific toxic chemicals of mobile and stationary sources and evaluate the MCR habitability based on the recommendations in NRC RG 1.78 to meet the requirements of TMI Action Plan Item III.D.3.4 and GDC 19.
COL 6.4(3)	The COL applicant is to identify and develop toxic gas detection requirements to protect the operators and provide reasonable assurance of the MCR habitability. The number, locations, sensitivity, range, type, and design of the toxic gas detectors are to be developed by the COL applicant.
COL 6.5(1)	The COL applicant is to provide the operational procedures and maintenance program as related to leak detection and contamination control.
COL 6.5(2)	The COL applicant is to maintain the complete documentation of system design, construction, design modifications, field changes, and operations.
COL 6.6(1)	The COL applicant is to identify the implementation milestones for ASME Section Xl inservice inspection program for ASME Code Section III Class 2 and 3 components.
COL 6.6(2)	The COL applicant is to identify the implementation milestone for the augmented inservice inspection program.
COL 6.8(1)	The COL applicant is to provide the operational procedures and maintenance program for leak detection and contamination control.
COL 6.8(2)	The COL applicant is to provide the preparation of cleanliness, housekeeping, and foreign materials exclusion program.
COL 6.8(3)	The COL applicant is to maintain the complete documentation of system design, construction, design modifications, field changes, and operations.
COL 6.8(4)	The COL applicant is responsible for the establishment and implementation of the Maintenance Rule program in accordance with 10 CFR 50.65.
COL 7.5(1)	The COL applicant is to provide a description of the site-specific AMI variables such as wind speed, and atmosphere stability temperature difference.
COL 7.5(2)	The COL applicant is to provide a description of the site-specific EOF.

Table 1.8-2 (10 of 29)

Item No.	Description
COL 8.2(1)	The COL applicant is to identify the circuits from the transmission network to the onsite electrical distribution system that are supplied by two physically independent circuits.
COL 8.2(2)	The COL applicant is to provide information on the location of rights-of-way, transmission towers, voltage level, and length of each transmission line from the site to the first major substation that connects the line to the transmission network.
COL 8.2(3)	The COL applicant is to describe the switchyard voltage related to the transmission system provider/operator (TSP/TSO) and the formal agreement between the nuclear power plant and the TSP/TSO. The COL applicant is to describe the capability and the analysis tool of the TSP. The COL applicant is also to describe the protocols for the plant to remain cognizant of grid vulnerabilities.
COL 8.2(4)	The COL applicant is to describe and provide layout drawings of the circuits connecting the onsite distribution system to the preferred power supply.
COL 8.2(5)	The COL applicant is to describe site-specific information for the protective devices, ac power, and dc power that control the switchyard equipment.
COL 8.2(6)	The COL applicant is to provide an FMEA for switchyard components. In addition, the COL applicant is to provide the results of grid stability analyses to demonstrate that the offsite power system does not degrade the normal and alternate preferred power sources to a level where the preferred power sources do not have the capacity or capability to support the onsite Class 1E electrical distribution system in performing its intended safety function.
COL 8.2(7)	The COL applicant is to design the offsite power system to detect, alarm, and automatically clear a single-phase open circuit condition.
COL 8.2(8)	The COL applicant is to describe how testing is performed on the offsite power system components.
COL 8.2(9)	The COL applicant is to provide the required number of immediate access circuits from the transmission network.

Table 1.8-2 (11 of 29)

Item No.	Description
COL 8.3(1)	The COL applicant is to provide and to design a mobile generator and its support equipment.
COL 8.3(2)	The COL applicant is to describe and provide detailed ground grid and lightning protection.
COL 8.3(3)	The COL applicant is to provide testing, inspection, and monitoring programs for detecting insulation degradation of underground and inaccessible power cables within the scope of 10 CFR 50.65.
COL 8.3(4)	The COL applicant is to provide protective device coordination.
COL 8.3(5)	The COL applicant is to provide insulation coordination of surge and lightning protection.
COL 8.3(6)	The COL applicant is to develop the maintenance program to optimize the life and performance of the batteries.
COL 8.3(7)	The COL applicant is to provide short circuit analysis of onsite dc power system with actual data.
COL 8.3(8)	The COL applicant is to describe any special features of the design that would permit online replacement of an individual cell, group of cells, or entire battery.
COL 8.4(1)	The COL applicant is to identify local power sources and transmission paths that could be made available to resupply power to the plant following the loss of a grid or the SBO.
COL 8.4(2)	The COL applicant is to develop detailed procedures for manually aligning the alternate AC power supply when two (Trains A and B) of the four diesel generators are unavailable during a loss of offsite power event.
COL 9.1(1)	The COL applicant is to provide operational procedures and maintenance program as related to leak detection and contamination control.
COL 9.1(2)	The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations.
COL 9.1(3)	The COL applicant is to address the load-handling procedures. Load-handling procedures are established for component handling procedures and plant operating procedures in accordance with ASME B30.2. ASME B30.2 requires establishing component handling procedures that include (1) a safe load path for lifting heavy loads to perform special handling component inspections, (2) acceptance criteria prior to lift, and (3) use of steps and proper sequence in handling the load. ASME B30.2 requires plant operating procedure guidelines that include appropriate crane operator training and crane inspections. ASME B30.2 also requires that the load-handling procedures include preparing operating procedures for preoperational load testing and checkouts of interlocks, brakes, hoisting cables, control circuitry, and lubrication of OHLHS equipment.

Table 1.8-2 (12 of 29)

Item No.	Description
COL 9.1(4)	The COL applicant is to provide plant procedures for preventing and mitigating inadvertent reactor cavity drain down events, maintenance procedures for the maintenance and inspection of refueling pool seal, and emergency response procedures for the proper measures during pool drain down events.
COL 9.1(5)	The COL applicant is to provide plant operating procedure guidelines for preoperational load testing and checkouts of interlocks, blocks, hoisting cables, control circuity and lubrication of fuel handling equipment.
COL 9.2(1)	The COL applicant is to develop procedures for system filling, venting, and operational procedures to minimize the potential for water hammer; to analyze the system for water hammer impacts; to design the piping system to withstand potential water hammer forces; and to analyze inadvertent water hammer events in accordance with NUREG-0927 in the ESWS.
COL 9.2(2)	The COL applicant is to develop layout of the site-specific portion of the system to minimize the potential for water hammer in the ESWS.
COL 9.2(3)	The COL applicant is to (1) to determine required pump design head, using pressure drop from the certified design portion of the plant and adding site-specific head requirements, (2) determine pump shutoff head to establish system design pressure, which is not to exceed APR1400 system design pressure, and (3) evaluate potential for vortex formation at the pump suction based on the most limiting applicable conditions in the ESWS.
COL 9.2(4)	The COL applicant is to determine the design details of the backwashing line, vent line, and their discharge locations in the ESWS.
COL 9.2(5)	The COL applicant is to provide the evaluation of the ESW pump at the high and low water levels of the UHS. In the event of approaching low UHS water level, the COL applicant is to develop a recovery procedure.
COL 9.2(6)	The COL applicant is to provide measures to prevent long-term corrosion and organic fouling that may degrade system performance in the ESWS.
COL 9.2(7)	The COL applicant is to evaluate the need and design and install freeze protection in the ESWS if required.
COL 9.2(8)	The COL applicant is to conduct periodic inspection, monitoring, maintenance, performance, and functional testing of the ESWS and UHS piping and components, including the heat transfer capability of the CCW heat exchangers based on GL 89-13 and GL 89-13 Supplement 1.

Table 1.8-2 (13 of 29)

Item No.	Description
COL 9.2(9)	The COL applicant is to develop procedures for water systems filling, venting, keeping the system full, and operation to minimize the potential for water hammer; to analyze the system for water hammer impacts; to design the piping system to withstand potential water hammer forces; and to analyze inadvertent water hammer events in accordance with NUREG-0927 in the CCWS.
COL 9.2(10)	The COL applicant is to provide operational procedures and maintenance programs as related to leak detection and contamination control in the CCWS.
COL 9.2(11)	The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations in the CCWS.
COL 9.2(12)	The COL applicant is to include a site-wide radiological environmental monitoring program to monitor environmental contamination in the CCWS.
COL 9.2(13)	The COL applicant is to determine all state and local departments of health and environmental protection standards to be applied and followed for the domestic water system.
COL 9.2(14)	The COL applicant is to determine the source of domestic water to the site and the necessary required treatment plant.
COL 9.2(15)	The COL applicant is to confirm the sizing of domestic water tanks and associated pumps, if used.
COL 9.2(16)	The COL applicant is to confirm whether the sanitary waste is sent to an onsite treatment facility or the city sewage system.
COL 9.2(17)	The COL applicant is to provide the UHS-related design information based on the site characteristics, including meteorological conditions.
COL 9.2(18)	The COL applicant is to provide the UHS-related systems such as blowdown, chemical injection, and makeup water system.
COL 9.2(19)	The COL applicant is to provide the location and design of the ESW building, and makeup water source.
COL 9.2(20)	The COL applicant is to provide isolation between the UHS and the non-safety-related systems.
COL 9.2(21)	The COL applicant is to provide the design of UHS cooling tower basin so the minimum water level will provide adequate NPSH to ESW pumps under accident conditions.
COL 9.2(22)	The COL applicant is to provide the non-safety-related makeup water source and capacity for normal operation loss and evaporation in the UHS.

Table 1.8-2 (14 of 29)

Item No.	Description
COL 9.2(23)	The COL applicant is to specify the following UHS chemistry requirements for bio-fouling and chemistry control: a. A chemical injection system to provide non-corrosive, non-scale-forming conditions to limit biological film formation b. The type of biocide, algaecide, pH adjuster, corrosion inhibitor, scale inhibitor, and silt dispersant, if necessary to maintain system performance based on site conditions.
COL 9.2(24)	The COL applicant is to verify the piping layout of the ESWS and UHS to prevent water hammer and develop operating procedures to provide reasonable assurance that the ESWS and UHS water pressure are above saturation conditions for all operating modes.
COL 9.2(25)	The COL applicant is to develop maintenance and testing procedures to monitor debris buildup and flush out and to remove the debris in the UHS.
COL 9.2(26)	The COL applicant is to evaluate the potential wind and recirculation effects of cooling towers based on meteorological condition.
COL 9.2(27)	The COL applicant is to provide the material specifications for piping, valves, and fittings of the UHS system based on site-specific conditions and meteorological conditions.
COL 9.2(28)	The COL applicant is to provide the evaluation of maximum evaporation and other losses based on the site-specific conditions and meteorological conditions in the UHS.
COL 9.2(29)	The COL applicant is to provide the detailed evaluation for UHS capability with consideration of site-specific conditions and meteorological data in the UHS.
COL 9.2(30)	The COL applicant is to provide chemical and blowdown to prevent biofouling and long-term corrosion, considering site water quality in the UHS.
COL 9.2(31)	The COL applicant is to provide the inspection and testing of the UHS to demonstrate that fouling and degradation mechanisms applicable to the site are effectively managed to maintain acceptable heat sink performance and integrity.
COL 9.2(32)	The COL applicant is to provide the alarms, instrumentation, and controls required for the safety-related functions of the UHS.
COL 9.2(33)	The COL applicant is to develop the following procedures for the water system: filling, venting, keeping it full, and operating it to minimize the potential for water hammer. The COL applicant is also to analyze the system for water hammer impacts, design the piping system to withstand potential water hammer forces, and analyze inadvertent water hammer events in the ECWS in accordance with NUREG-0927.
COL 9.2(34)	The COL applicant is either to prepare or to include operational procedures and maintenance programs.
COL 9.2(35)	The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations.
COL 9.2(36)	The COL applicant is to include a site-wide radiological environmental monitoring program to monitor both the horizontal and vertical variability of the onsite hydrogeology and the potential effects of the construction and operation of the plant.
COL 9.3(1)	The COL applicant is to provide operational procedures and maintenance programs as related to leak detection and contamination control.

Table 1.8-2 (15 of 29)

Item No.	Description
COL 9.3(2)	The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations.
COL 9.3(3)	The COL applicant is to prepare the site radiological environmental monitoring program.
COL 9.3(4)	The COL applicant is to provide the supply systems of the nitrogen gas subsystem, the hydrogen subsystem, the carbon dioxide subsystem, and the breathing air systems.
COL 9.4(1)	The COL applicant is to provide the capacities of heating coils in the safety-related air handling units and cooling and heating coils in the non safety-related air handling units affected by site-specific conditions.
COL 9.4(2)	The COL applicant is to provide the capacities of heating coils of electric duct heaters affected by site-specific conditions.
COL 9.4(3)	The COL applicant is to provide the system design information of ESW building and CCW heat exchanger building HVAC system including flow diagram, if the ESW building and CCW heat exchanger building require the HVAC system.
COL 9.4(4)	The COL applicant is to establish operational procedures and maintenance programs as related to leak detection and contamination control.
COL 9.5(1)	The COL applicant is to establish a fire protection program, including organization, training, and qualification of personnel, administrative controls of combustibles and ignition sources, firefighting procedures, and quality assurance.
COL 9.5(2)	The COL applicant is to address the design and fire protection aspects of the facilities, buildings and equipment, and a fire protection water supply system, which are site specific and/or are not a standard feature of the APR1400.
COL 9.5(3)	The COL applicant is to describe the provided apparatus for plant personnel and fire brigades such as portable fire extinguishers, self-contained breathing apparatus, and radio communication systems.
COL 9.5(4)	The COL applicant is to address the final FHA and FSSA based on the final plant design, including a detailed post-fire safe-shutdown circuit analysis.
COL 9.5(5)	The COL applicant is to provide a reliable starting method for the AAC GTG.
COL 9.5(6)	The COL applicant is to provide details of emergency response facilities and associated communication capabilities.

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Item No.	Description
COL 9.5(7)	The COL applicant is to provide the fire brigade radio systems.
COL 9.5(8)	The COL applicant is to provide the LAN and VPN system.
COL 9.5(9)	The COL applicant is to provide the emergency offsite communication system including dedication hotline, local law enforcement radio equipment, and wireless communication system.
COL 9.5(10)	The COL applicant is to specify that adequate and acceptable sources of fuel oil are available, including the means of transporting and recharging the fuel storage tank, following a design basis accident.
COL 9.5(11)	The COL applicant is to provide a description of the offsite communication system that interfaces with the onsite communication system, including type of connectivity, radio frequency, normal and backup power supplies, and plant security system interface.
COL 9.5(12)	The COL applicant is to provide the security radio system that consists of a base unit, mobile units, and portable units.
COL 9.5(13)	The COL applicant is to provide the local law enforcement communications including dedicated conventional telephone and radio-transmitted two-way communication system.
COL 9.5(14)	The COL applicant is to provide electric power for the security lighting system.
COL 9.5(15)	The COL applicant is to provide the system design information of AAC GTG building HVAC system including flow diagram, if the AAC GTG building requires the HVAC system.
COL 10.2(1)	The COL applicant is to identify the turbine vendor and model.
COL 10.2(2)	The COL applicant is to identify how the functional requirements for the overspeed protection system are met and provide a schematic of the TGCS and protection systems from sensors through valve actuators.
COL 10.2(3)	The COL applicant is to provide a description of how the turbine missile probability analysis conforms with Subsection 10.2.3.6 to ensure that requirements for protection against turbine missiles (e.g., applicable material properties, method of calculating the fracture toughness properties per SRP Section 10.2.3 Acceptance Criteria, preservice inspections) will be met.
COL 10.3(1)	The COL applicant is to provide operating and maintenance procedures including adequate precautions to prevent water (steam) hammer and relief valve discharge loads and water entrainment effects in accordance with NUREG–0927 and a milestone schedule for implementation of the procedure.
COL 10.3(2)	The COL applicant is to establish operational procedures and maintenance programs as related to leak detection and contamination control.
COL 10.3(3)	The COL applicant is to provide a description of the FAC monitoring program for carbon steel portions of the steam and power conversion systems that contain water or wet steam and are susceptible to erosion-corrosion damage. The description is to address consistency with GL 89-08 and NSAC-202L-R3 and provide a milestone schedule for implementation of the program.

Table 1.8-2 (17 of 29)

Item No.	Description
COL 10.4(1)	The COL applicant is to establish operational procedures and maintenance programs for leak detection and contamination control
COL 10.4(2)	The COL applicant is to maintain the complete documentation of system design, construction, design modifications, field changes, and operations
COL 10.4(3)	The COL applicant is to provide the location and design of the cooling tower, basin, and CW pump house
COL 10.4(4)	The COL applicant is to provide elevation drawings
COL 10.4(5)	The COL applicant is to address the design features for the prevention of contamination
COL 10.4(6)	The COL applicant is to provide operating and maintenance procedures for the following items in accordance with NUREG-0927 and a milestone schedule for implementation of the procedures.
COL 10.4(7)	The COL applicant is to describe the nitrogen or equivalent system design for SG drain
COL 10.4(8)	The COL applicant is to prepare the Site Radiological Environmental Monitoring Program
COL 10.4(9)	The COL applicant is to determine the wet bulb temperature correction factor to account for potential interference and recirculation effects
COL 11.2(1)	The COL applicant is to prepare the site-specific ODCM in accordance with NEI 07-09A.
COL 11.2(2)	The COL applicant is to prepare operational procedures and programs related to operations, inspection, calibration, and maintenance of the contamination control program.
COL 11.2(3)	The COL applicant is to determine whether contaminated laundry is sent to an offsite facility for cleaning or for disposal.
COL 11.2(4)	The COL applicant is to prepare and provide the P&IDs.
COL 11.2(5)	The COL applicant is to perform a site-specific cost-benefit analysis following the guidance in the regulatory requirements of NRC RG 1.110.
COL 11.2(6)	The COL applicant is to provide reasonable assurance that the mobile or temporary equipment and interconnections to plant systems conform with the regulatory requirements and guidance of 10 CFR 50.34a, 10 CFR 20.1406, NRC RG 1.143, and ANSI/ANS 40.37.
COL 11.2(7)	The COL applicant is to develop the procedure for the collection and shipment of mixed wastes, if and when they are generated, for offsite treatment. The generation of mixed liquid wastes is minimized by process control and the controlled use of hazardous chemicals.
COL 11.2(8)	The COL applicant is to develop the interface design and provide the site-specific information for the LWMS effluent discharge, including radioactive release points, effluent temperature, the design (type, shape, and size) of flow orifices, and the sampling requirements following the guidance of NRC RG 1.21 and RG 4.15 and the standards incorporated therein by reference.

Table 1.8-2 (18 of 29)

Item No.	Description
COL 11.2(9)	The COL applicant is to develop a plant-wide NRC RG 4.21 Program following the guidance in NEI 08-08A for contamination control.
COL 11.2(10)	The COL applicant is to maintain the complete documentation of system design, construction, design modifications, field changes, and operations and make them available for decommissioning planning and implementation.
COL 11.2(11)	The COL applicant is to prepare the site process control program and the site radiological environmental monitoring program.
COL 11.2(12)	The COL applicant is to confirm the assumed dilution flow rate provided by cooling tower blowdown, dilution pump, or other plant discharges at the discharge point based on site-specific parameters.
COL 11.2(13)	The COL applicant is to calculate dose to members of the public following the guidance of NRC RG 1.109 and NRC RG 1.113 using site-specific parameters and to compare the doses due to the liquid effluents with the numerical design objectives of Appendix I to 10 CFR 50, 10 CFR 20.1302, and 40 CFR 190.
COL 11.2(14)	The COL applicant is to perform an analysis to demonstrate that the potential groundwater or surface water contamination concentrations resulting from radioactive release from the liquid-containing tank failure, are in compliance with the limits in 10 CFR 20, Appendix B, Table 2.
COL 11.3(1)	The COL applicant is to prepare and implement the epoxy inspection and maintenance program in the GRS.
COL 11.3(2)	The COL applicant is to perform a site-specific cost-benefit analysis following the guidance in NRC RG 1.110 for conformance with 10 CFR 50 Appendix I.
COL 11.3(3)	The COL applicant is to prepare and provide the piping and instrumentation diagram (P&ID) for the combined operating license application.
COL 11.3(4)	The COL applicant is to prepare the operational procedures and maintenance programs related to leak detection and contamination control.
COL 11.3(5)	The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations for decommissioning planning.
COL 11.3(6)	The COL applicant is to prepare the site process control program and the site radiological environmental monitoring program.
COL 11.3(7)	The COL applicant is also to perform the dose calculation using the total gaseous effluents from the site for comparison with the requirements of 40 CFR 190.
COL 11.3(8)	The COL applicant is to perform an analysis using site-specific meteorological data to demonstrate that the potential airborne concentration resulting from GRS failure meets the requirements of 10 CFR 20, Appendix B, Table 2.
COL 11.3(9)	The COL applicant is to prepare an ODCM following the guidance in NEI 07-09A template.

Table 1.8-2 (19 of 29)

Item No.	Description
COL 11.4(1)	The COL applicant can incorporate an onsite laundry facility for processing of contaminated clothing.
COL 11.4(2)	The COL applicant is to perform a site-specific cost-benefit analysis following the guidance in NRC RG 1.110.
COL 11.4(3)	The COL applicant is to provide reasonable assurance that the provisions and requirements of ANSI/ANS-40.37-2009 are met. The COL applicant is to provide reasonable assurance that mobile and temporary solid radwaste processing and its interconnection to plant systems conform with regulatory requirements and guidance such as 10 CFR 50.34a, 10 CFR 20.1406, and NRC RG 1.143. The COL applicant is to prepare a plan to develop and use operating procedures so the guidance and information in IE Bulletin 80-10 are followed.
COL 11.4(4)	The COL applicant is to provide P&IDs.
COL 11.4(5)	The COL applicant is to prepare the site process control program and the site radiological environmental monitoring program.
COL 11.4(6)	The COL applicant is responsible for the collection, temporary storage, and shipment of mixed waste for offsite treatment and disposal.
COL 11.4(7)	The COL applicant is responsible for the provision of a site-wide IRSF for interim storage of radioactive wastes.
COL 11.4(8)	The COL applicant is to provide a mobile crane to retrieve a waste package that becomes stuck in the lifted condition or that is dropped.
COL 11.4(9)	The COL applicant is also to provide operational procedures to properly ship low-level wastes to external sites in accordance with US NRC and US Department of Transportation (DOT) regulations.
COL 11.4(10)	The COL applicant is to prepare the operational procedures and maintenance programs for the SWMS as related to leak detection and contamination control.
COL 11.4(11)	The COL applicant is to develop plant-wide RG 4.21 life-cycle planning for minimization of contamination program following the guidance in NEI 08-08A, in which the SWMS procedures and programs are to be integrated.
COL 11.4(12)	The COL applicant is to maintain the complete documentation of system design, construction, design modifications, field changes, and operations.
COL 11.5(1)	The COL applicant is to determine the WARN and ALARM setpoints of the PERMSS based on the site-specific conditions and operational requirements.
COL 11.5(2)	The COL applicant is to develop an annual report that specifies the quantity of each principal radionuclide released to unrestricted areas in liquid and gaseous effluents.
COL 11.5(3)	The COL applicant is to provide site-specific procedures that conform with the numerical guides of 10 CFR 50.34a and 10 CFR Part 50, Appendix I.

Table 1.8-2 (20 of 29)

Item No.	Description
COL 11.5(4)	The COL applicant is to prepare an ODCM that contains a description of the methodology and parameters for calculation of the offsite doses for the gaseous and liquid effluents. The COL applicant is to follow NEI 07-09A as an alternative to providing an offsite dose calculation manual.
COL 11.5(5)	The COL applicant is to provide analytical procedures and sensitivity for selected radioanalytical methods and types of sampling media for site-specific matter.
COL 11.5(6)	The COL applicant is to develop the calibration procedures in accordance with NRC RG 1.33 and 4.15.
COL 11.5(7)	The COL applicant is to develop detailed location and tubing installation and provide the sampling method including the sampling time to acquire representative sampling.
COL 11.5(8)	The COL applicant is to provide operational procedures and maintenance programs related to leak detection and contamination control.
COL 11.5(9)	The COL applicant is to develop a radiological and environmental monitoring program, taking into consideration local land use and census data in identifying all potential radiation exposure pathways. The COL applicant is to follow NEI 07-09A as an alternative to providing a radiological and environmental monitoring program.
COL 12.1(1)	The COL applicant is to provide the organizational structure to effectively implement the radiation protection policy, training, and reviews consistent with operational and maintenance requirements, while satisfying the applicable regulations and Regulatory Guides including NRC RGs 1.33, 1.8, 8.8, and 8.10.
COL 12.1(2)	The COL applicant is to describe the operational radiation protection program to provide reasonable assurance that occupational radiation exposures are ALARA.
COL 12.1(3)	The COL applicant is to describe how the plant follows the guidance provided in NRC RGs 8.2, 8.7, 8.9, 8.13, 8.15, 8.20, 8.25, 8.26, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, and 8.38.
COL 12.2(1)	The COL applicant is to provide any additional contained radiation sources, such as instrument calibration radiation sources, that are not identified in Subsection 12.2.1.
COL 12.3(1)	The COL applicant is to provide portable instruments and the associated training and procedures in accordance with 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737.
COL 12.3(2)	The COL applicant is to determine the WARN and ALARM setpoints of the ARMS based on the site-specific conditions and operational requirements

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Item No.	Description
COL 12.4(1)	The COL applicant is to estimate construction worker doses based on site-specific number of operating units, distances, meteorological conditions, and construction schedule.
COL 12.4(2)	The COL applicant is to provide operational procedures and programs, including the development of a site radiological environmental monitoring program, to implement the minimization of contamination approach.
COL 12.4(3)	The COL applicant is to implement concrete tunnels for piping of the systems that may include underground piping carrying contaminated or potentially contaminated fluid to minimize buried piping.
COL 12.5(1)	The COL applicant is to provide the operational radiation protection program, including the items described in Section 12.5.
COL 13.1(1)	The COL applicant is to provide a description of the corporate or home office organization, its functions and responsibilities, and the number and the qualifications of personnel. The COL applicant is to be directed to activities such as the facility design, design review, design approval, construction management, testing, and operation of the plant.
COL 13.1(2)	The COL applicant is to develop a description of experience in the design, construction, and operation of nuclear power plants and experience in activities of similar scope and complexity.
COL 13.1(3)	The COL applicant is to describe its management, engineering, and technical support organizations. The description includes organizational charts for the current headquarters and engineering structure and any planned modifications and additions to those organizations to reflect the added functional responsibilities with the nuclear power plant.
COL 13.1(4)	The COL applicant is to develop a description of the organizational arrangement. The description is to include organizational charts reflecting the current headquarters and engineering structure and any planned modifications and additions to reflect the added functional responsibilities associated with the addition of the nuclear plant to the applicant's power generation capacity. The description shows how these responsibilities are delegated and assigned or expected to be assigned to each of the working or performance-level organizational units identified to implement these responsibilities. The description includes organizational charts reflecting the current corporate structure and the working- or performance-level organizational units that provide technical support for the operation.
COL 13.1(5)	The COL applicant is to develop the description of the general qualifications in terms of educational background and experience for positions or classes of positions described in the organizational arrangement.
COL 13.1(6)	The COL applicant is to develop a description of the structure, functions, and responsibilities of the onsite organization established to operate and maintain the plant.
COL 13.1(7)	The COL applicant is to provide an organizational chart showing the title of each position, minimum number of persons to be assigned to duplicate positions, number of operating shift crews, and positions that require reactor operator and senior reactor operator licenses.

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Item No.	Description
COL 13.1(8)	The COL applicant is to provide organizational information such as the functions, responsibilities, and authorities of the plant position. The COL applicant is to develop a description of the line of succession of authority and responsibility for overall station operation in the event of unexpected temporary contingencies, and the delegation of authority.
COL 13.1(9)	The COL applicant is to develop a description of the position titles, applicable operator licensing requirements for each, and the minimum numbers of personnel planned for each shift for all combinations of units proposed to be at the station in either operating or cold shutdown mode. The COL applicant is also to develop the description of shift crew staffing plans unique to refueling operations.
COL 13.1(10)	The COL applicant is to provide a description of the education, training, and experience requirements for each management, operating, technical, and maintenance position in the operating organization.
COL 13.1(11)	The COL applicant is to provide the qualification requirements of the initial appointees to plant positions for key plant managerial and supervisory personnel through the shift supervisory level.
COL 13.2(1)	The COL applicant is to develop the description and schedule of the training program for licensed reactor operators and non-licensed plant staff.
COL 13.2(2)	The COL applicant is to develop the site-specific training program by using NEI 06-13A as the template for the basic structure and content.
COL 13.2(3)	The COL applicant is to provide a licensed plant staff training program in accordance with NUREG-0800, Subsection 13.2.1.I.3.
COL 13.2(4)	The COL applicant is to provide a non-licensed plant staff training program in accordance with NUREG-0800, Subsection 13.2.2.I.3.
COL 13.2(5)	The COL applicant is to develop training programs. The programs are to include a chart that shows the schedule of each part of the training program for each functional group of employees in the organization in relation to the schedule for preoperational testing, expected fuel loading, and expected time for examinations prior to plant criticality for licensed operators.
COL 13.2(6)	The COL applicant is to determine the extent of the NRC guidance that is applicable to the facility training program or the justification of exceptions.
COL 13.3(1)	The COL applicant is to develop the interfaces of design features with site-specific designs and site parameters.
COL 13.3(2)	The COL applicant is to develop a comprehensive emergency plan. The plan is developed as a physically separate document and includes copies of letters of agreement (or other certifications) from state and local governmental agencies with emergency planning responsibilities.

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Item No.	Description
COL 13.3(3)	The COL applicant is to address an emergency classification and action level scheme as required by 10 CFR 50.47(b)(4).
COL 13.3(4)	The COL applicant is to develop the security-related aspects of an emergency plan.
COL 13.3(5)	The COL applicant is to develop a multi-unit site interface plan depending on the location of the new reactor on or near an operating reactor site with an existing emergency plan.
COL 13.3(6)	The COL applicant is to develop emergency planning inspections, tests, analyses, and acceptance criteria.
COL 13.4(1)	The COL applicant is to develop operational programs and provide schedules for implementation of the programs, as defined in SECY-05-0197. The COL applicant is to provide commitments for the implementation of operational programs that are required by regulation. In some instances, the programs may be implemented in phases, where practical, and the applicant is to include the phased implementation milestones.
COL 13.4(2)	The COL applicant is responsible for developing a leakage monitoring and prevention program for the systems, as specified in Subsection 5.5.2 in Chapter 16, Technical Specifications. The leakage monitoring and prevention program is to provide suitable methods and acceptance criteria as defined in NUREG-0737, Item III.D.1.1.
COL 13.5(1)	The COL applicant is to describe the administrative and operating procedures that the operating organization (plant staff) use to provide reasonable assurance that routine operating, off-normal, and emergency activities are conducted in a safe manner. The COL applicant is to provide a brief description of the nature and content of the procedures and a schedule for the preparation of appropriate written administrative procedures.
COL 13.5(2)	The COL applicant is to develop a description of administrative procedures that provide administrative control over activities that are important to safety for operation of the facility. NRC RG 1.33 contains guidance on facility administrative policies and procedures. The COL applicant is to determine whether the portions of NRC RG 1.33 applicable to plant procedures are followed. If the guidance is not followed, the COL applicant is to develop a description of alternative methods that will be used and the manner of implementing them.
COL 13.5(3)	The COL applicant is to describe the different classifications of procedures the operators use in the MCR and locally in the plant for plant operations. The COL applicant is to identify the group within the operating organization responsible for maintaining the procedures and describe the general format and content of the different classifications.
COL 13.5(4)	The COL applicant is to provide a program for developing operating procedures.
COL 13.5(5)	The COL applicant is to provide a program for developing and implementing emergency operating procedures.

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Item No.	Description
COL 13.5(6)	The COL applicant is to describe how other operating and maintenance procedures are classified, which group or groups within the operating organization have the responsibility for following each class of procedures, and the general objectives and character of each class and subclass.
COL 13.5(7)	The COL applicant is to provide a program for developing shutdown procedure.
COL 13.6(1)	The COL applicant is to develop a physical security plan, training and qualification plan, and safeguards contingency plan. The COL applicant is to address site-specific information related to the physical security, contingency, and guard training and qualification plans. These documents are categorized as SGI and are withheld from public disclosure pursuant to 10 CFR 73.21. The COL applicant is to address site-specific physical security ITAACs as applicable.
COL 13.6(2)	The COL applicant is to develop an access authorization program that meets the requirements of 10 CFR 73.56, and conformance with the requirement is to be specified in the physical security plan.
COL 13.6(3)	The COL applicant is to develop a cyber security plan and implementation program in accordance with 10 CFR 73.54. The plan document is categorized as SGI and is to be withheld from public disclosure pursuant to 10 CFR 2.390(d)(1).
COL 13.7(1)	The COL applicant is to develop the description of the fitness-for-duty programs during construction and for the operating plant.
COL 14.2(1)	The COL applicant is to develop the site-specific organization and staffing level appropriate for its facility.
COL 14.2(2)	The COL applicant is to prepare the site-specific test procedures and/or guidelines that are to be used for the conduct of the plant startup program.
COL 14.2(3)	The COL applicant is to prepare a startup administrative manual and also provide preoperational and startup test summaries that contain testing objectives and acceptance criteria applicable for its scope of the plant design. Testing performed at other than design operating conditions for systems is to be reconciled either through the test acceptance criteria or post-test data analysis.
COL 14.2(4)	The COL applicant is to perform review and evaluation of individual test results.
COL 14.2(5)	The COL applicant is to develop the detailed description of test and acceptance criteria for the Security System.
COL 14.2(6)	The COL applicant is to develop a schedule for the development of the plant operating and emergency procedures that should allow sufficient time for trial use of these procedures during the initial test program. The schedule for plant startup is to be developed by the COL applicant to allow sufficient time to systematically perform the required testing in each phase.
COL 14.2(7)	The COL applicant is to describe its program for reviewing available information on reactor operating and testing experiences and discusses how it used this information in developing the initial test program. The description is to include the sources and types of information reviewed, the conclusions or findings, and the effect of the review on the initial test program.

Table 1.8-2 (25 of 29)

Item No.	Description
COL 14.2(8)	The COL applicant that references the APR1400 design certification is to identify the specific operator training to be conducted as part of the low-power testing program related to the resolution of TMI Action Plan Item I.G.1, as described in (1) NUREG-0660, “NRC Action Plans Developed as a Result of the TMI-2 Accident,” Revision 1, August 1980 and (2) NUREG-0737, “Clarification of TMI Action Plan Requirements.”
COL 14.2(9)	The COL applicant is to prepare the pre-operational test of cooling tower and associated auxiliaries, and raw water and service water cooling systems.
COL 14.2(10)	The COL applicant is to develop the test program of personnel monitors and radiation survey instruments.
COL 14.2(11)	The COL applicant is to develop the test procedure of the communication system.
COL 14.3(1)	The COL applicant is to provide the ITAAC for the site-specific portion of the plant systems specified in Subsection 14.3.3.
COL 14.3(2)	The COL applicant is to provide the proposed ITAAC for the facility’s emergency planning addressed in Subsection 14.3.2.10.
COL 14.3(3)	The COL applicant is to provide the proposed ITAAC for the facility’s physical security hardware addressed in Subsection 14.3.2.12.
COL 14.3(4)	The COL applicant is to provide a DAC closure schedule for implementing the piping DAC.
COL 15.0(1)	The COL applicant is to perform the radiological consequence analysis using site-specific χ/Q values, unless the χ/Q values used in the DCD envelop the site-specific short-term or long-term χ/Q values of the DCD, and to show that the resultant doses are within the guideline values of 10 CFR 50.34 for EAB and LPZ and that of 10 CFR Part 50, Appendix A, GDC 19 for the MCR and TSC.
COL 17.4(1)	The COL applicant is to develop and implement Phases 2 and 3 of the design RAP, including QA requirements. In Phase 2, the plant’s site-specific information is to be subjected to the design RAP process, and the site-specific risk-significant SSCs are combined with the APR1400 design risk-significant SSCs into one list for the plant. Phase 2 is to be performed during the COL application phase and updated/maintained during the COL license holder phase. In Phase 3, procurement, fabrication, construction, and test specifications for the SSCs within the scope of the RAP provide reasonable assurance that key assumptions, such as equipment reliability, are realistic and achievable. The QA requirements are implemented during the procurement, fabrication, construction, and pre-operational testing of the SSCs within the scope of the RAP. Phase 3 is to be performed during the COL license holder phase and prior to initial fuel loading. The COL applicant is to propose a method for incorporating the objectives of the reliability assurance program into other programs for design or operational errors that degrade non-safety-related, risk-significant SSCs.

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Item No.	Description
COL 17.4(2)	The COL applicant is to develop and implement the RAP in the operational phase in which RAP activities are integrated into the existing operational program (e.g., Maintenance Rule, surveillance testing, in-service inspection, in-service testing, QA). The RAP in the operational phase also includes the process for providing corrective actions for design and operational errors that degrade non-safety-related SSCs within the scope of the RAP. A description of the proposed method for developing /integrating the operational RAP into operating plant programs (e.g., Maintenance Rule, quality assurance) is to be performed during the COL application phase. The development/integration of the O-RAP is performed during the COL license holder phase and prior to initial fuel loading. All SSCs identified as risk-significant within the scope of the design RAP are categorized as high-safety-significant (HSS) within the scope of the initial Maintenance Rule. Integration of reliability assurance activities into existing operational programs also addresses the establishment of: a. Reliability performance goals for risk-significant SSCs consistent with the existing maintenance and quality assurance processes on the basis of information from the design RAP (for example, implementation of the Maintenance Rule following the guidance contained in NRC RG 1.160 is one acceptable method for establishing performance goals if SSCs are categorized as HSS within the scope of the Maintenance Rule program). b. Performance and condition monitoring requirements to provide reasonable assurance that risk-significant SSCs do not degrade to an unacceptable level during plant operations.
COL 17.5(1)	The COL applicant is to establish and implement a QA program that is applicable to site-specific design activities related to the plant construction and operation phases.
COL 17.6(1)	The COL applicant is to provide in its Final Safety Analysis Report a description of the Maintenance Rule program and a plan for implementing it to meet the requirements of 10 CFR 50.65.
COL 19.0(1)	The COL applicant is either to confirm that the PRA in the design certification bounds the site-specific design information and any design changes or departures, or to update the PRA to reflect the site-specific design information and any design changes or departures.

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Item No.	Description
COL 19.1(1)	The COL applicant is to describe the uses of PRA in support of licensee programs, and to identify and describe risk-informed applications being implemented during the combined license application phase.
COL 19.1(2)	The COL applicant is to describe the uses of PRA in support of licensee programs, and identify and describe risk-informed applications being implemented during the construction phase.
COL 19.1(3)	The COL applicant is to describe the uses of PRA in support of licensee programs, and identify and describe risk-informed applications being implemented during the operational phase.
COL 19.1(4)	The COL applicant is to review as-designed and as-built information and conduct walkdowns as necessary to confirm that the assumptions used in the PRA (including PRA inputs to RAP and SAMDA) remain valid with respect to internal events, internal flood and fire events (routings and locations of pipe, cable, and conduit), and HRA analyses (development of operating procedures, emergency operating procedures, and severe accident management guidelines and training), external events including PRA-based seismic margins and HCLPF fragilities, and LPSD procedures.
COL 19.1(5)	The COL applicant is to conduct a peer review of the PRA relative to the industry PRA Standard prior to use of the PRA to support risk-informed applications, as applicable.
COL 19.1(6)	The COL applicant is to describe the PRA maintenance and upgrade program.
COL 19.1(7)	The COL applicant is to confirm that the PRA-based seismic margin assessment is bounding for the selected site, and to update the assessment to include site-specific SSC and soil effects (including sliding, overturning liquefaction, and slope failure). The COL applicant is to confirm that the as-built plant has adequate seismic margin.
COL 19.1(8)	The COL applicant is address following issues with a site-specific risk assessment, as applicable: dam failure, external flooding, extreme winds and tornadoes, industrial or military facility, pipeline accident, release of chemicals from onsite storage, river diversion, sandstorm, toxic gas, and transportation accidents.
COL 19.1(9)	The COL applicant is to describe the uses of PRA in support of licensee programs such as Maintenance Rule implementation during the operational phase.
COL 19.1(10)	The COL applicant is to describe the uses of PRA in support of licensee programs such as the reactor oversight process during the operational phase.

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Item No.	Description
COL 19.1(11)	The COL applicant is to develop the fire barrier management procedures that direct the appropriate use of a fire watch and use of the isolation devices with a quick-disconnect mechanism for hose and cables that bleach a fire barrier.
COL 19.1(12)	The COL applicant is to develop procedures and operator training for reliance (during fire response) on undamaged instrumentation (when the location of the fire is known).
COL 19.1(13)	The COL applicant is to develop procedures specifying that a fire watch be present when hot work is being performed.
COL 19.1(14)	The COL applicant is to establish procedures for closing the containment hatch (after being opened during during LPSD operations) to promptly re-establish the containment as a barrier to fission product release. This guidance must include steps that allow for sealing of the hatch with four bolts (versus the 40 bolts used to secure the hatch during at-power operation); four bolts are sufficient to secure the hatch so that no visible gap can be seen between the seals and the sealing surface.
COL 19.1(15)	The COL applicant is to develop a configuration control program requiring that, during Modes 4, 5, and 6, the watertight flood doors and fire doors be maintained closed in at least one quadrant. Furthermore, the COL applicant is to incorporate, as part of the aforementioned configuration control program, a provision that if the flood or fire doors to this designated quadrant must be opened for reasons other than normal ingress/egress, a flood or fire watch must be established for the affected doors.
	The COL applicant is to develop outage management procedures that limit planned maintenance that can potentially impair one or both SC trains during the shutdown modes.
COL 19.1(16)	The COL applicant is to develop procedures and a configuration management strategy to address the period of time when one SC train is unexpectedly unavailable (including the termination of any testing or maintenance that can affect the remaining train and restoration of all equipment to its nominal availability).
COL 19.2(1)	The COL applicant is to perform and submit site-specific equipment survivability assessment in accordance with 10 CFR 50.34(f) and 10 CFR 50.44.
COL 19.2(2)	The COL applicant is to develop and submit an accident management plan.

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Item No.	Description
COL 19.3(1)	The COL applicant is to perform site-specific seismic hazard evaluation and seismic risk evaluation as applicable in accordance with NTTF Recommendation 2.1 as outlined in the NRC RFI.
COL 19.3(2)	The COL applicant is to address the flood requirements for wet sites
COL 19.3(3)	The COL applicant is to develop the details for offsite resources.
COL 19.3(4)	The COL applicant is to address the details of storage location for FLEX equipment.
COL 19.3(5)	The COL applicant is to address site-specific strategies to mitigate BDBEEs as specified in the NRC Order EA-12-049.
COL 19.3(6)	The COL applicant is to address SFP level instrumentation maintenance procedure development and perform training as specified in NRC Order EA-12
COL 19.3(7)	The COL applicant is to address development of EOPs, SAMGs, and EDMGs that incorporate lessons learned from TEPCO's Fukushima Dai-Ichi nuclear power plant accident as addressed in SECY-12-0025.
COL 19.3(8)	The COL applicant is to address enhancement of the offsite communication system as specified in the NRC Request for Information pertaining to NTTF Recommendation 9.3.
COL 19.3(9)	The COL applicant is to address staffing for large-scale natural events as specified in the NRC RFI pertaining to NTTF Recommendation 9.3.

1.9 Conformance with Regulatory Criteria

The conformance of the APR1400 design with U.S. Nuclear Regulatory Commission (NRC) regulatory criteria is documented in this section. Regulatory criteria include NRC Regulatory Guides (RGs), Standard Review Plans (SRPs), generic issues including Three Mile Island (TMI) requirements, operational experience (generic communications), and advanced and evolutionary light-water reactor design issues per Subsections C.I.1.9.1 through C.I.1.9.5 of NRC RG 1.206 (Reference 1).

In addition, the conformance with post-Fukushima NRC recommendations and requirements are addressed in section 1.9.6.

The combined license (COL) applicant is to address an evaluation of the conformance with regulatory criteria for the site-specific portion and operational aspects of the facility.

1.9.1 Conformance with Regulatory Guides

This section provides an evaluation of conformance with the following groups of NRC RGs:

a. Division 1, Power Reactors

b. Division 4, Environmental and Siting

c. Division 5, Materials and Plant Protection

d. Division 8, Occupational Health

Conformance with applicable active guides is summarized in Table 1.9-1. The evaluation includes an identification and description of deviations from the guidance in the NRC RGs, as well as suitable justifications for exceptions or any alternative approaches. For NRC RGs not applicable to design, the reason for non-applicability is specified in the "Conformance or Summary Description of Deviation" column of Table 1.9-1. Table 1.9-1 is also cross-referenced to applicable sections of the design control document (DCD). NRC RGs that have been withdrawn or are not publicly available are excluded from Table 1.9-1.

1.9.2 Conformance with Standard Review Plan

In accordance with 10 CFR 52.47(a)(9) (Reference 2), this subsection provides the APR1400 conformance with the acceptance criteria for each section of the SRP in effect 6 months before the docket date of the application. The evaluation results are presented in Table 1.9-2. The evaluation includes the identification and description of deviations from the SRP. Where differences exist, specific sections are identified and further details relevant to each SRP deviation are addressed.

1.9.3 Generic Issues

In accordance with 10 CFR 52.47(a)(21), this subsection addresses the proposed technical resolution for all unresolved safety issues (USIs) and medium-and high-priority generic safety issues (GSIs) that are identified in the version of NUREG-0933 (Reference 3), current as of the date 6 months before the docket date of the application, and that are technically relevant to the design. USIs and GSIs applicable to the APR1400 design were identified using the criteria given in Regulatory Position C.IV.8 of NRC RG 1.206. Appendix B to NUREG-0933 (Rev. 25) (Reference 4), issued on September 2011, which is current on the date up to 6 months before the docket date of application, was used to identify new generic issues applicable to the APR1400 design certification. The evaluation results for each issue with the cross-reference to related sections are provided in Table 1.9-3.

In accordance with the requirements of 10 CFR 52.47(a)(8), the evaluations were also performed to assess conformance with any technically relevant portions of the TMI requirements set forth in 10 CFR 50.34(f) (Reference 5), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v). Table 1.9-4 addresses the assessment results of conformance with the TMI requirements and is cross-referenced to related DCD sections. TMI requirements applicable to the reactor types other than pressurized water reactor (PWR) or a specific vendor design are excluded from Table 1.9-4.

1.9.4 Operational Experience (Generic Communications)

The requirements of 10 CFR 52.47(a)(22) and Regulatory Position C.I.1.9.3 of NRC RG 1.206 specify that design certification applicants are to provide information necessary to demonstrate how the plant design incorporates operating experience insights from Generic Letters (GLs) and bulletins issued after the most recent revision of the applicable SRP and 6 months before the docket date of the application or demonstrate comparable international operating experience.

The APR1400 design is an evolutionary plant design that has been developed based on Combustion Engineering's System 80+ plant.

GLs and bulletins issued after the March 2007 revision of the SRP have been assessed to address how the applicable operating experience has been incorporated into the APR1400 design. Table 1.9-5 provides the applicability of the generic communications to the APR1400 design and the results of the assessment.

1.9.5 Advanced and Evolutionary Light-Water Reactor Design Issues

Section C.I.1.9.5 of NRC RG 1.206 specifies that the applicant is to address the licensing and policy issues developed by the NRC and documented in the Office of the Secretary of the Commission (SECY) documents and the associated Staff Requirements Memoranda (SRM) for advanced and evolutionary light water reactor (LWR) designs that apply to the proposed facility design.

Table 1.9-6 provides a discussion of the applicability of the SECYs listed in Section C.I.1.9.5 of NRC RG 1.206 to the APR1400 design. Table 1.9-7 provides a discussion of individual issues specified in SECY-93-087 (Reference 6) and provides cross references to relevant to the DCD chapters, sections, or subsections.

1.9.6 Conformance with Post-Fukushima NRC Recommendations and Requirements

As a result of the Fukushima Dai-Ichi event, additional requirements have been established to manage and mitigate external events that are beyond the design basis of the plant. This section addresses the APR1400 conformance with SECY-12-0025 including the requirements contained in NRC Orders EA-12-049 and EA 12-051 and the related request for information. The specific details of addressing the Tier 1, 2 and 3 NTTF items are discussed in DCD 19.3.

Table 1.9-8 titled, "APR1400 Strategy for Addressing Tier 1, 2 and 3 NTTF Recommendations" provides response summary to SECY-11-0093, SECY-11-0137, SECY-11-0025, NRC Orders EA-12-049 and EA-12-051. Additionally, this table also provides the reference section(s) in the APR1400 design (Technical Report No. APR1400-E-P-NR-14005-P), applicable DCD section and COL action required to specifically address the NTTF recommendations and requirements.

1.9.7 Part 21 Notification of Failure to Comply or Existence of a Defect and Its Evaluation

Conformance with 10 CFR Part 21 is a necessary requirement in the APR1400 design control process. Assessment and management of the design or other issues against the reporting requirements is required during both the development and implementation of the design. At of the time of the ARP1400 design certification application submission, no issues related to the APR1400 design that meet the reporting criteria of 10 CFR Part 21 (Reference 7) had been identified.

1.9.8 Combined License Information

COL 1.9(1) The COL applicant is to provide an evaluation of the conformance with the regulatory criteria for the site-specific portions and operational aspects of the facility.

1.9.9 References

1. Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," U.S. Nuclear Regulatory Commission, June 2007.

2. 10 CFR 52.47, "Contents of Applications; Technical Information," U.S. Nuclear Regulatory Commission

3. NUREG-0933, "Resolution of Generic safety Issues," Rev. 34, U.S. Nuclear Regulatory Commission, September 2011, (includes Supplements 1-34).

4. NUREG-0933, Appendix B, "Applicability of NUREG-0933 Issues to Operating and Future Reactor Plants," Rev. 25, U.S. Nuclear Regulatory Commission, September 2011.

5. 10 CFR 50.34(f), "Additional TMI-related Requirements," U.S. Nuclear Regulatory Commission

6. SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” U.S. Nuclear Regulatory Commission, July 1993

7. 10 CFR Part 21, "Reporting of Defects and Noncompliance," U.S. Nuclear Regulatory Commission.

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APR1400 Conformance with Regulatory Guides

NRC Regulatory Guide	Revision/ Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.1 Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps	11/1970	The APR1400 conforms with the regulatory position with the following exception: • Calculations of available NPSH for the emergency core cooling and containment heat removal pumps were performed assuming that the containment pressure during post-accident conditions is equal to the vapor pressure of the liquid in the containment. This assumption provides reasonable assurance that the actual available NPSH is always greater than the calculated available NPSH, which meets the intent of the regulatory position.	Table 6.2.2-1
1.3 Assumptions Used for Evaluating the Potential Radiological Consequences of a LOCA for Boiling Water Reactors	Rev. 2 06/1974	Not applicable (BWR)	N/A
1.4 Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors	Rev. 2 06/1974	The APR1400 applies NRC RG 1.183 instead of this NRC RG.	N/A
1.5 Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors	03/1971	Not applicable (BWR)	N/A
1.6 Independence Between Redundant Standby (On-Site) Power Sources and Between Their Distribution Systems	03/1971	The APR1400 conforms with this NRC RG.	8.1.3.3, 8.3.1.2.2, 8.3.2.2.2
1.7 Control of Combustible Gas Concentration in Containment	Rev. 3 03/2007	The APR1400 conforms with this NRC RG.	6.1.1.1, 6.2.5.1

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NRC Regulatory Guide		Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.8	Qualification & Training of Personnel for Nuclear Power Plants	Rev. 3 05/2000	Not applicable (COL)	N/A
1.9	Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants	Rev. 4 03/2007	The APR1400 conforms with this NRC RG.	8.1.3.3, 8.3.1.2.2
1.11	Instrument Lines Penetrating Primary Reactor Containment	Rev. 1 03/2010	The APR1400 conforms with this NRC RG.	3.6.2.1.4.2, 6.2.4.1
1.12	Nuclear Power Plant Instrumentation for Earthquakes	Rev. 2 03/1997	The APR1400 conforms with this NRC RG.	3.7.4.1
1.13	Spent Fuel Storage Facility Design Basis	Rev. 2 03/2007	The APR1400 conforms with this NRC RG.	9.1.1.1, 9.1.1.3, 9.1.2.1, 9.1.3.3.3, 9.1.4.3, 9.1.5.2.1, 9.1.5.3, 9.4.2.1
1.14	Reactor Coolant Pump Flywheel Integrity	Rev. 1 08/1975	The APR1400 conforms with this NRC RG.	5.4.1.1
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	Rev. 3 03/2007	The APR1400 conforms with this NRC RG with the following exception: • Startup testing with measurement of SG internals	3.9.2.4, 14.2.7.1.6
1.21	Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste	Rev. 2 06/2009	The APR1400 conforms with this NRC RG.	11.5, 12.3.4, TS Part 3, 5.0
1.22	Periodic Testing of Protection System Actuation Functions	02/1972	The APR1400 conforms with this NRC RG.	7.1.2.38, Table 7.1-1, 7.2.2.5, 7.2.3.3, 7.3.2.5, 7.3.3.5, 8.1.3.3
1.23	Meteorological Monitoring Programs for Nuclear Power Plants	Rev. 1 03/2007	Not applicable (COL)	N/A

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NRC Regulatory Guide		Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure	03/1972	Not applicable	N/A
1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors	03/1972	The APR1400 applies NRC RG 1.183 instead of this NRC RG.	N/A
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	Rev. 4 03/2007	The APR1400 conforms with this NRC RG.	3.2.2, 3.2.3, 5.2.1.1, 6.2.4.1.2, 6.6.1, 9.2.1.2.2.1, 9.5.4.1, 9.5.5.1, 9.5.5.3, 9.5.6, 9.5.6.3, 9.5.6, 9.5.6.3, 9.5.7, 9.5.7.3, 9.5.8, 10.3.6.2, 10.4.2.1, 10.4.3
1.27	Ultimate Heat Sink for Nuclear Power Plants	Rev. 2 01/1976	The APR1400 conforms with this NRC RG with the following exception: • Design of the UHS is site-specific and will be the responsibility of the COL applicant.	9.2.1.1.1, 9.2.5, 9.2.5.1, 9.2.5.3
1.28	Quality Assurance Program Requirements (Design and Construction)	Rev. 4 06/2010	The APR1400 conforms with this NRC RG.	10.3.6.2, 10.4.2.1, 10.4.9.1.2, 14.2.6, 17.5

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.29 Seismic Design Classification	Rev. 4 03/2007	The APR1400 conforms with this NRC RG.	3.2.1, 5.2.5, 6.2.4.1.2, 9.1.2.1, 9.1.2.2.3, 9.1.4.3, 9.1.5.2.1, 9.1.5.2.2, 9.1.5.2.3, 9.2.1.1.1, 9.2.2.1.1, 9.2.5.1, 9.4.3.1, 9.4.5.1.2, 9.4.5.1.1, 9.5.4.1, 9.5.5.1, 9.5.5.3, 9.5.6.1, 9.5.6.3, 9.5.7.1, 9.5.7.3, 9.5.8.1, 10.3.1, 10.4.8,10.4.9.3, 11.2, 11.3, 11.4
1.30 Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment	08/1972	The APR1400 conforms with this NRC RG.	17.5
1.31 Control of Ferrite Content in Stainless Steel Weld Metal	Rev. 4 10/2013	The APR1400 conforms with this NRC RG	4.5.2.2, 4.5.2.4, 5.2.3.4.4, 5.3.1.4, 5.4.2.1.4, 6.1.1.1, 6.1.1.2.2
1.32 Criteria for Power Systems for Nuclear Power Plants	Rev. 3 03/2004	The APR1400 conforms with this NRC RG.	8.1.3.3, 8.2.2.2, 8.3.1.2.2, 8.3.2.2.2, 9.5.4.1
1.33 Quality Assurance Program Requirements (Operation)	Rev. 3 06/2013	Not applicable (COL)	N/A

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NRC Regulatory Guide		Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.34	Control of Electroslag Weld Properties	Rev. 1 03/2011	The APR1400 conforms with this NRC RG except that the electroslag process is not used during fabrication of any reactor coolant pressure boundary components.	5.2.3.3, 5.2.3.4.4, 5.3.1.4, 5.4.2.1.4
1.35	Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment	Rev. 3 07/1990	The APR1400 conforms with this NRC RG.	3.8.1.2.2, 3.8.1.7.2.3
1.35.1	Determining Prestressing Forces for Inspection of Prestressed Concrete Containments	07/1990	The APR1400 conforms with this NRC RG.	3.8.1.2.2, 3.8.1.5.1.2, 3.8.1.5.2.2, 3.8.1.7.2.2, 3.8A.1.4.1.3.3, 3.8.1.2.2, 3.8.1.5.1.2, 3.8.1.5.2.2
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel	02/1973	The APR1400 conforms with this NRC RG.	5.2.3.2.3, 5.4.2.1.4, 6.1.1.1, 6.1.1.2.2, 6.1.1.2.3
1.40	Qualification of Continuous-Duty Safety-Related Motors for Nuclear Power Plants	Rev. 1 02/2010	The APR1400 conforms with this NRC RG.	N/A
1.41	Preoperational Testing of Redundant On-site Electric Power Systems to Verify Proper Load Group Assignments	03/1973	The APR1400 conforms with this NRC RG.	14.2.12, 8.1.3.3
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components	Rev. 1 03/2011	The APR1400 conforms with this NRC RG.	5.2.3.3, 5.3.1.4, 5.4.2.1.3

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NRC Regulatory Guide		Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.44	Control of the Processing and Use of Stainless Steel	Rev. 1 03/2011	The APR1400 conforms with this NRC RG.	4.5.1.2, 4.5.2.4, 5.2.3.2.2, 5.2.3.4.1, 5.3.1.4, 5.4.2.1.4, 6.1.1.1, 6.1.1.2.2
1.45	Guidance on Monitoring and Responding to Reactor Coolant System Leakage	Rev. 1 05/2008	The APR1400 conforms with this NRC RG.	3.6.3.5.1, 5.2.5, 5.2.5.1.1.2, 5.2.5.1.2.2, 5.2.5.1.2.4, 5.2.5.1.3, 5.2.5.2, 5.2.5.5, 9.3.3.1.2, 11.5.1.2
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	Rev. 1 02/2010	The APR1400 conforms with this NRC RG.	7.1.2.39, Table 7.1-1, 7.2.2.5, 7.3.2.5, 7.5.1.3, 7.5.2.3, 7.7.1.3, 8.3.1.2.2, 8.1.3.3, 8.3.2.2.2
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel	Rev. 1 03/2011	The APR1400 conforms with this NRC RG.	5.2.3.3, 5.3.1.2, 5.3.1.4, 5.4.2.1.3, 6.1.1.1, 6.1.1.2.2, 10.3.6.2
1.52	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants	Rev. 4 09/2012	The APR1400 conforms with this NRC RG.	6.4.2.2, 6.4.6, 6.5.1.1, 6.5.1.1, 6.5.1.2.1, 6.5.1.4.1, 6.5.1.4.2, 6.5.1.5, 6.5.1.5.4, 6.5.1.6, 7.3.1.9, 9.4.1.1, 9.4.1.4, 9.4.2.1, 9.4.5.1.3, 9.4.5.4.3, 11.3.4, 14.2.12.1.98

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NRC Regulatory Guide		Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	Rev. 2 11/2003	The APR1400 conforms with this NRC RG.	7.1.2.40, Table 7.1-1, 8.1.3.3, 8.3.1.2.2, 8.3.2.2.2
1.54	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants	Rev. 2 10/2010	The APR1400 conforms with this NRC RG.	6.1.2, 11.2.1.2, 6.8.4.5
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components	Rev. 2 05/2013	The APR1400 conforms with this NRC RG.	3.8.2.2.2
1.59	Design Basis Floods for Nuclear Power Plants	Rev. 2 08/1977	The APR1400 conforms with this NRC RG except for the actual site-related flooding assessment (COL).	2.4, 3.4.1.1
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants	Rev. 2 07/2014	The APR1400 conforms with this NRC RG.	3.7.1.1.1, 3.7.1.1.2, 3.7B-1
1.61	Damping Values for Seismic Design of Nuclear Power Plants	Rev. 1 03/2007	The APR1400 conforms with this NRC RG.	3.7.1.2, 3.7.3.9, 3.9.2.2.13, 3.9.3.3.2.2, 3.10.2, 3.10.2.1, 3.10.2.2, 3.12.3.2.1, 3.12.3.2.3, 3.12.3.3, 3.12.5.4, 3.12.6.8, App. 3.9B.4
1.62	Manual Initiation of Protective Actions	Rev. 1 06/2010	The APR1400 conforms with this NRC RG.	7.1.2.41, Table 7.1-1, 10.4.9.1.2, 8.1.3.3

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.63 Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants	Rev. 3 02/1987	The APR1400 conforms with this NRC RG.	3.11.2, 8.1.3.3, Table 8.1-2, 8.3.1.1.9, 8.3.1.2.1, 8.3.1.2.2, 8.3.2.2.1, 8.3.2.2.2
1.65 Materials and Inspections for Reactor Vessel Closure Studs	Rev. 1 04/2010	The APR1400 conforms with this NRC RG.	5.2.3.6, 5.3.1.7, 5.3.3.8
1.68 Initial Test Programs for Water-Cooled Nuclear Power Plants	Rev. 4 06/2013	The APR1400 conforms with this NRC RG.	6.3.4.2, 7.4.2, 10.4.4.4, 14.2.7, 14.2.7.1, Table 14.2-7, 14.3.2.14
1.68.2 Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants	Rev. 2 04/2010	The APR1400 conforms with this NRC RG.	7.4.2, 14.2.12.4.7
1.68.3 Preoperational Testing of Instrument and Control Air Systems	Rev. 1 09/2012	The APR1400 conforms with this NRC RG.	14.2.7.2, 14.2.12.1.125
1.69 Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants	Rev. 1 05/2009	The APR1400 conforms with this NRC RG.	12.3.2.2
1.70 Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants	Rev. 3 11/1978	Not applicable (refer to NRC RG 1.206)	N/A
1.71 Welder Qualification for Areas of Limited Accessibility	Rev. 1 03/2007	The APR1400 conforms with this NRC RG.	4.5.2.4, 5.2.3.3, 5.2.3.4.4, 5.3.1.4, 5.4.2.1.3, 5.4.2.1.4, 6.1.1.1, 6.1.1.2.2, 10.3.6.2

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NRC Regulatory Guide		Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.72	Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin	Rev. 2 11/1978	Not applicable The APR1400 design does not use spray pond piping made from fiberglass-reinforced thermosetting resin.	N/A
1.73	Qualification Tests for Safety-Related Actuators in Nuclear Power Plants	Rev. 1 10/2013	The APR1400 conforms with this NRC RG.	3.9.3.3.1.3, 3.9.3.3.1.3.1, 3.11.2, 8.1.3.3
1.75	Criteria for Independence of Electrical Safety Systems	Rev. 3 02/2005	The APR1400 conforms with this NRC RG except the following. • Two CEA position inputs instead of four CEA position input described in Subsection 7.1.2.3.	7.1.2.42, Table 7.1-1, 7.2.2.3, 7.3.2.3, 7.9.2.7, 8.1.3.3, 8.3.1.1.2.3, 8.3.1.1.9, 8.3.1.1.10, 8.3.1.2.2, 8.3.2.1.2.2, 8.3.2.1.2.4, 8.3.2.1.2.5, 8.3.2.2.2
1.76	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants	Rev. 1 03/2007	The APR1400 conforms with this NRC RG.	Table 2.0-1, 3.3.2.1, 3.5.1.4
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors	05/1974	The APR1400 conforms with this NRC RG. Refer to Subsection 15.4.8 for further information. Note: SRP Section 4.2 Appendix B will be used in conjunction with the requirements of NRC RG 1.77 for the APR1400.	15.4.8
1.78	Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	Rev. 1 12/2001	The APR1400 conforms with this NRC RG except for the following: • Full conformance by the COL applicant with site-specific consequence data.	6.4.4.2, 6.4.7

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NRC Regulatory Guide		Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Systems	Rev. 2 10/2013	The APR1400 conforms with this NRC RG.	6.3.4.1, 14.2.7.3
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants	Rev. 1 01/1975	Not applicable. The APR1400 is a single unit plant; therefore, this NRC RG is not applicable to the APR1400.	N/A
1.82	Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident	Rev. 4 03/2012	The APR1400 conforms with this NRC RG.	6.2.2.2.5, 6.2.2.3, 6.3.1.3, 6.3.2.2.3, 6.8.2.2.1, 6.8.4.5, Table 15.0-12, Table 15.0-13
1.84	Design, Fabrication, and Materials Code Case Acceptability, ASME Section III	Rev. 36 08/2014	The APR1400 conforms with this NRC RG.	3.12.2.2, 3.13.1.1, 4.5.1.1, 4.5.2.1, 5.2.3.1, 5.2.6, 6.0, 10.3.6.2
1.86	Termination of Operating Licenses for Nuclear Reactors	06/1974	Not applicable (COL)	N/A
1.87	Guidance for Construction of Class 1 Components in Elevated Temperature Reactors	06/1975	Not applicable	N/A
1.89	Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants	Rev. 1 06/1984	The APR1400 conforms with this NRC RG except for the following. • IEEE Standard 323-2003 is applied instead of IEEE Standard 323-1974 because NRC RG 1.209 endorses the current national qualification standard (IEEE Standard 323-2003).	3.9.3.3.1.2, 3.9.3.3.1.3.1, 3.11.2, 3.11.5.2, Table 6.5-2, Table 7.1-1

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NRC Regulatory Guide		Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.90	Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons	Rev. 2 11/2012	Not applicable The APR1400 adopted prestressed concrete containment with ungrouted tendon.	N/A
1.91	Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants	Rev. 2 04/2013	Not applicable (COL)	N/A
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis	Rev. 3 10/2012	The APR1400 conforms with this NRC RG.	3.7.2.6, 3.7.2.7, 3.7.3.5, 3.9.2.2.5, 3.9.2.2.6, 3.12.3.2.4, 3.12.3.2.5, 3.12.5.5, App. 3.9B.4
1.93	Availability of Electric Power Sources	Rev.1 03/2012	The APR1400 conforms with this NRC RG.	8.1.3.3, 16.3.8
1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants	Rev. 1 06/1976	Not applicable (BWR)	N/A

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.97 Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants.	Rev. 4 06/2006	The APR1400 conforms with this NRC RG.	1.2.6.3, 3.10, 3.11.1.1, 3.11.2, 3.11.6, 7.1.1.5, 7.1.2.43, Table 7.1-1, 7.5.1.1, 7.5.2.1, 7.7.1.2, 10.4.9.5.3, 10.4.9.5, 11.5.1, 11.5.2, 11.5.2.1, 11.5.3, 12.3.1.7, 12.3.4.1.1, 12.3.4.1.5, 14.3.2.7
1.98 Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor	03/1976	Not applicable (BWR)	N/A
1.99 Radiation Embrittlement of Reactor Vessel Materials	Rev. 2 05/1988	The APR1400 conforms with this NRC RG.	5.2.3.1, 5.3.1.4, 5.3.1.6.7, 5.3.2, 5.3.2.1.1, 5.3.2.1.2, 5.3.2.4

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.100 Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants	Rev. 3 09/2009	The APR1400 conforms with this NRC RG.	3.9.2.2.1, 3.9.3.3.1.1, 3.9.3.3.1.2, 3.9.3.3.1.3, 3.9.3.3.1.3.1, 3.9.3.3.2.2, 3.9.6.1, 3.10.1.1, 3.10.1, 3.10.2, 3.10.2.1, 3.10.2.2, 3.10.2.3, 3.11.2, 5.2.2.1.1, 5.4.12.2.1, 5.4.12.2.2, Table 6.5-2, 8.3.2.2.2
1.101 Emergency Planning and Preparedness for Nuclear Power Reactors	Rev. 5 06/2005	Not applicable (COL)	N/A
1.102 Flood Protection for Nuclear Power Plants	Rev. 1 09/1976	The APR1400 conforms with this NRC RG.	3.4.1.1, 3.4.1.2
1.105 Setpoints for Safety-Related Instrumentation	Rev. 3 12/1999	The APR1400 conforms with this NRC RG.	7.1.2.44, Table 7.1-1, 7.2.2.7, 7.3.2.7, 15.0.0.9

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.106 Thermal Overload Protection for Electric Motors on Motor-Operated Valves	Rev. 2 02/2012	The APR1400 conforms with this NRC RG.	Table 8.1-2, 8.3.1.1.3.12, 8.3.1.2.2, 8.3.2.2.2
1.107 Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures	Rev. 2 06/2011	Not applicable The APR1400 adopted prestressed concrete containment with ungrouted tendon.	N/A
1.109 Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I	Rev. 1 10/1977	The APR1400 conforms with this NRC RG.	11.2.5, Table 11.2-4, 11.3.3.1, 11.3.7, Table 11.3-5, 11.4.4, 11.5.1.2
1.110 Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors	Rev. 1 10/2013	The APR1400 conforms with this NRC RG. The cost-benefit analysis approach stipulated by 10 CFR Part 50, Appendix I, Section II, Paragraph D requires that a population dose analysis be performed to demonstrate that the radwaste system is designed consistent with the as low as reasonably achievable criterion. Due to the extreme site-specific nature of population dose analyses, the cost-benefit analysis is deferred to site-specific environmental reports.	11.2.1.5, 11.3.1.6,

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.111 Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases From Light-Water-Cooled Reactors	Rev. 1 07/1977	Not applicable (COL)	N/A
1.112 Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors	Rev. 1 03/2007	The APR1400 conforms with this NRC RG.	11.1, 11.2.2, 11.2.3.1, 11.3.1.2
1.113 Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I	Rev. 1 04/1977	The APR1400 conforms with this NRC RG.	11.2.5, 11.4.4, 11.5.1.2
1.114 Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Plant	Rev. 3 10/2008	Not applicable (COL)	N/A
1.115 Protection Against Turbine Missiles	Rev. 2 01/2012	The APR1400 conforms with this NRC RG.	3.5.1.3, 9.1.2, 10.1.1, 10.2.1, 10.3.1
1.117 Tornado Design Classification	Rev. 1 04/1978	The APR1400 conforms with this NRC RG.	3.3.2, 9.1.2.1, 10.3.1
1.118 Periodic Testing of Electric Power and Protection Systems	Rev. 3 04/1995	The APR1400 conforms with this NRC RG.	Table 6.5-2, 7.1.2.45, Table 7.1-1, 7.2.3.3, 7.3.2.5, 7.5.2.1, 7.6.2.3, Table 8.1-2, 8.1.3.3, 8.3.1.1.6, 8.3.1.2.2, 8.3.2.2.2
1.121 Bases for Plugging Degraded PWR Steam Generator Tubes	08/1976	The APR1400 conforms with this NRC RG.	5.4.2, 5.4.2.2.2.12

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.122 Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components	Rev. 1 02/1978	The APR1400 conforms with this NRC RG.	3.7.2.5, 3.7.2.9, 3.7A.3.3, 3.12.3.2.2
1.124 Service Limits and Loading Combinations for Class 1 Linear Type Component Supports	Rev. 3 07/2013	The APR1400 conforms with this NRC RG.	3.9.3.4 Table 3.9-3
1.125 Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants	Rev. 2 03/2009	Not applicable (COL)	N/A
1.126 An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification	Rev. 2 03/2010	The APR1400 conforms with this NRC RG.	4.2.1.2.3
1.127 Inspection of Water-Control Structures Associated with Nuclear Power Plants	Rev. 1 03/1978	Not applicable (COL)	N/A
1.128 Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	Rev. 2 02/2007	The APR1400 conforms with this NRC RG.	8.1.3.3, Table 8.1-2, 8.3.2.2.2, 9.4.4.2.2, 9.4.5.2.2.3, 9.4.5.2.2.4, 9.4.7.2.1
1.129 Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	Rev. 3 09/2013	The APR1400 conforms with this NRC RG.	8.1.3.3, Table 8.1-2, 8.3.2.2.2
1.130 Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Supports	Rev. 3 07/2013	The APR1400 conforms with this NRC RG.	3.9.3.4 Table 3.9-3

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.132 Site Investigations for Foundations of Nuclear Power Plants	Rev. 2 10/2003	Not applicable (COL)	N/A
1.133 Loose-Part Detection Program for the Primary Systems of Light-Water-Cooled Reactors	Rev. 1 05/1981	The APR1400 conforms with this NRC RG.	7.7.1.5
1.134 Medical Evaluation of Licensed Personnel at Nuclear Power Plants	Rev. 3 03/1998	Not applicable (COL)	N/A
1.136 Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments	Rev. 3 03/2007	The APR1400 conforms with this NRC RG.	3.8.1.2.2, 3.8.1.3, 3.8.1.3.2, 3.8.1.4.7, 3.8.1.6, 3.8.1.6.3, 3.8A.1.3.1, 3.8A.1.3.2
1.137 Fuel Oil Systems for Emergency Power Supplies	Rev.2 06/2013	The APR1400 conforms with this NRC RG.	8.1.3.3, 9.5.4.1
1.138 Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants	Rev. 2 12/2003	Not applicable (COL)	N/A
1.140 Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants	Rev. 2 06/2001	The APR1400 conforms with this NRC RG.	9.4.2.1, 9.4.2.4, 9.4.5.1.3, 9.4.5.4.3, 9.4.6.4.2, 9.4.7.4, 11.3.4, 14.2.12.1.94, 14.2.12.1.98, 14.2.12.1.99

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.141 Containment Isolation Provisions for Fluid Systems	Rev. 1 07/2010	The APR1400 conforms with this NRC RG.	6.2.4
1.142 Safety-Related Concrete Structures for Nuclear Power Plants	Rev. 2 11/2001	The APR1400 conforms with this NRC RG.	3.5.3.2, 3.8.3.3, 3.8.4.4, 3.8.4.4.2.1, 3.8.4.5, 3.8.5.4, 3.8A.2.4.2
1.143 Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants	Rev. 2 11/2001	The APR1400 conforms with this NRC RG.	3.2.1, 3.2.2, 10.4.8.1.2, 11.2, 11.2.1.2, 11.2.2.3, 11.2.5, 11.3, 11.3.1.3, 11.3.5, Table 11.3-2, 11.4.1.2, 11.4.1.3, 11.4.1.7, 11.4.2.2.1, 11.4.2.3, 11.4.9, 11.5.1.2
1.145 Atmospheric Dispersion Models for Potential Accident Consequence Assessment at Nuclear Power Plants	Rev. 1 02/1983	Not applicable (COL)	N/A
1.147 In-service Inspection Code Case Acceptability, ASME Section XI, Division 1	Rev. 17 08/2014	Not applicable (COL)	5.2.1.2, 5.2.4.1.9, 5.2.4.2, 6.6.1, 6.6.3
1.149 Nuclear Power Plant Simulation Facilities for Use in Operator Training, License Examinations, and Applicant Experience Requirements	Rev. 4 04/2011	The APR1400 conforms with this NRC RG.	N/A

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.151 Instrument Sensing Lines	Rev. 1 07/2010	The APR1400 conforms with this NRC RG.	3.2.1, 7.1.2.46, Table 7.1-1, 7.2.2.3, 7.3.2.3
1.152 Criteria for Digital Computer in Safety Systems of Nuclear Power Plants	Rev. 3 07/2011	The APR1400 conforms with this NRC RG.	1.5.4, 7.1.2.47, 7.1.2.71, Table 7.1-1, 7.9.1.2
1.153 Criteria for Safety Systems	Rev. 1 06/1996	The APR1400 conforms with this NRC RG.	8.1.3.3, Table 8.1-2, 8.3.1.1.2.2, 8.3.1.2.2, 8.3.2.1.2.4, 8.3.2.2.2
1.155 Station Blackout	08/1988	The APR1400 conforms with this NRC RG.	8.1.2, 8.1.3.2, 8.1.3.3 Table 8.1-2, 8.2.2.2, 8.3.1.1.3, 8.3.1.2.2, 8.3.2.2.2, 8.4.1.1, 8.4.1.2, 8.4.1.3, 8.4.1.6, 8.4.2.2, 9.4.1.1, 9.5.9, 9.5.9.1, 9.5.9.2, 10.3.1, 10.4.9.1.2
1.156 Qualification of Connection Assemblies for Nuclear Power Plants	Rev. 1 07/2011	The APR1400 conforms with this NRC RG.	3.11.2, 8.1.3.3
1.157 Best-Estimate Calculations of Emergency Core Cooling System Performance	05/1989	The APR1400 conforms with this NRC RG. SBLOCA analyses performed for the APR1400 are based on the more stringent requirements of Appendix K to 10 CFR Part 50.	6.2.1.5.1, 15.6.5.3.1

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.158 Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants	02/1989	The APR1400 conforms with this NRC RG except for the following. • IEEE Standard 535-2006 is applied instead of IEEE Standard 535-1986 because NRC RG 1.212 endorses the current national qualification standard (IEEE Standard 535-2006).	3.11.2, 8.1.3.3
1.159 Assuring the Availability of Funds for Decommissioning Nuclear Reactors	Rev. 2 10/2011	Not applicable (COL)	N/A
1.160 Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	Rev. 3 05/2012	Not applicable (COL)	N/A
1.161 Evaluation of Reactor Pressure Vessels with Charpy Upper Shelf Energy Less Than 50 Ft-Lb	06/1995	Not applicable	N/A
1.162 Format and Contents of Report for Thermal Annealing of Reactor Pressure Vessel	02/1996	Not applicable (COL)	N/A
1.163 Performance-Based Containment Leak-Test Program	09/1995	The APR1400 conforms with this NRC RG.	6.2.1.6, 6.2.6, 6.2.6.1, 6.2.6.4
1.166 Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Action	03/1997	Not applicable (COL)	N/A
1.167 Restart of Nuclear Power Plant Shut Down by a Seismic Event	03/1997	Not applicable (COL)	N/A
1.168 Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety System of Nuclear Power Plants	Rev. 2 07/2013	The APR1400 conforms with this NRC RG.	7.1.2.48, Table 7.1-1

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.169 Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Rev. 1 07/2013	The APR1400 conforms with this NRC RG except the position as follows; • Regulatory Position - Conformance with IEEE 828-2005 • Position of the APR1400 - Conformance with IEEE 828-1998 • Summary Description of Deviation - There are no deviations in the body between IEEE 828-1998 and IEEE 828-2005 except for the document numbering and adding Appendix B in IEEE 828-1998. Therefore, conformance with IEEE 828-2005 is met by conforming with IEEE 828-1998.	7.1.2.49, Table 7.1-1
1.170 Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Rev. 1 07/2013	The APR1400 conforms with this NRC RG.	7.1.2.50, Table 7.1-1
1.171 Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Rev. 1 07/2013	The APR1400 conforms with this NRC RG.	7.1.2.51, Table 7.1-1
1.172 Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Rev. 1 07/2013	The APR1400 conforms with this NRC RG.	7.1.2.52, Table 7.1-1
1.173 Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Rev. 1 07/2013	The APR1400 conforms with this NRC RG.	7.1.2.53, Table 7.1-1

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.174 An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis	Rev. 2 05/2011	Not applicable	N/A
1.175 An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing	08/1998	Not applicable	N/A
1.177 An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications	Rev.1 05/2011	Not applicable	N/A
1.178 An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping	Rev. 1 09/2003	Not applicable	N/A
1.179 Standard Format and Content of License Termination Plans for Nuclear Power Reactors	Rev. 1 06/2011	Not applicable (COL)	N/A
1.180 Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems	Rev. 1 10/2003	The APR1400 conforms with this NRC RG.	3.11.2, 7.1.2.54, Table 7.1-1, 7.2.2.8, 7.3.2.8, 7.9.2.11, 8.1.3.3
1.181 Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)	09/1999	Not applicable (COL)	N/A

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.183 Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors	07/2000	The APR1400 conforms with this NRC RG.	3.11.2, 3.11.5.2, 6.4.2.5, Table 6.5-2, 12.2.3, 12.3.1.8, 12.4.1.2.7, 15.0.3.2, 15.0.3.3, 15.0.3.4, 15.0.3.6, 15.0.3.7, 15.1.5.5, 15.1.5.5.2, 15.2.8.5.2, 15.3.3.5.1, 15.3.3.5.2, 15.4.8.5.1, 15.4.8.5.2, 15.6.2.5, 15.6.3.2.5, 15.6.3.2.5.2, 15.6.5.5, 15.6.5.5.1.1, 15.6.5.5.1.2, 15.6.5.5.1.3, 15.7.4.1, 15.7.4.2, 15A.1.2.2, 15A.2.1, 15A.5.2.1, 15A.5.2.2, 15A.5.3
1.184 Decommissioning of Nuclear Power Reactors	Rev. 1 10/2013	Not applicable (COL)	N/A
1.185 Standard Format and Content for Post-Shutdown Decommissioning Activities Report	Rev. 1 06/2013	Not applicable (COL)	N/A

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.186 Guidance and Examples for Identifying 10 CFR 50.2 Design Bases	12/2000	Not applicable	N/A
1.187 Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments	11/2000	Not applicable (COL)	N/A
1.188 Standard Format and Content for Applications To Renew Nuclear Power Plant Operating Licenses	Rev. 1 09/2005	Not applicable (COL)	N/A
1.189 Fire Protection for Nuclear Power Plants	Rev. 2 10/2009	Conformance with exceptions. Refer to Table 9.5.1-1 for a point-by-point discussion of conformance with this guide.	3.2.1, 7.1.2.55, Table 7.1-1, 7.4.2, 8.1.3.3, 9.5.1.1, 9.5.1.2, 9.5.1.2.1, 9.5.1.2.1, 9.5.1.2.3, 9.5.1.2.4, 9.5.1.2.5, 9.5.1.2.6, 9.5.1.3.2, 9.5.2.2.1.8, 9.5.3.1, Table 9.5.1-1, Table 9.5.1-2, 9.5A.1, 9.5A.2.1, 9.5A.2.5, 9.5A.2.5.1, 9.5A.3.1, 11.3.1.4
1.190 Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence	03/2001	The APR1400 conforms with this NRC RG.	N/A
1.191 Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown	05/2001	Not applicable (COL)	N/A

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.192 Operation and Maintenance Code Case Acceptability, ASME OM Code	Rev. 1 08/2014	The APR1400 conforms with this NRC RG.	5.2.1.2
1.193 ASME Code Cases Not Approved for Use	Rev. 4 08/2014	The APR1400 conforms with this NRC RG.	N/A
1.194 Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants	06/2003	The APR1400 conforms with this NRC RG.	2.3.4
1.195 Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors	05/2003	The APR1400 applies to NRC RG 1.183 instead of this NRC RG.	N/A
1.196 Control Room Habitability at Light-Water Nuclear Power Reactors	Rev. 1 01/2007	The APR1400 conforms with this NRC RG.	6.4
1.197 Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors	05/2003	The APR1400 conforms with this NRC RG.	6.4.5, 9.4.1.4
1.198 Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites	11/2003	Not applicable (COL)	N/A
1.199 Anchoring Components and Structural Supports in Concrete	11/2003	The APR1400 conforms with this NRC RG.	3.8.4, 4.2.1, 3.8.4.4.2.6, 3.12.6.4
1.200 An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities	Rev. 2 03/2009	The APR1400 conforms with this NRC RG.	Table 7.1-1, 19.0, 19.1

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.201 Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance	Rev. 1 05/2006	Not applicable. This NRC RG was written to address PRAs performed in support of changes proposed for existing, already-licensed plants.	N/A
1.202 Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors	02/2005	Not applicable (COL)	N/A
1.203 Transient and Accident Analysis Methods	12/2005	Not applicable. APR1400 conforms with the NRC RG 1.157.	N/A
1.204 Guidelines for Lightning Protection of Nuclear Power Plants	11/2005	The APR1400 conforms with this NRC RG.	7.1.2.56, Table 7.1-1, Table 8.1-2, 8.1.3.3, 8.2.2.2, 8.3.1.1.8, 8.3.1.2.2. 15.6.5
1.205 Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants	Rev. 1 12/2009	Not applicable. This NRC RG is allowed to the nuclear power reactor licensees to permit reactor licensees to adopt risk-informed, performance-based approach as an alternative to the existing deterministic fire protection requirement. APR 1400 fire protection is designed to the requirements of NRC RG 1.189, Rev. 2, which provides deterministic fire protection guidance; it is not necessary to incorporate this NRC RG.	N/A
1.206 Combined License Applications for Nuclear Power Plants (LWR Edition)	06/2007	The APR1400 conforms with exception. Section C.II.2, C.III, and C.IV are guidance for COL application referencing a certified design and/or an early site permit (ESP). These sections conform in the COL application. The APR1400 is not a passive-ALWR-plant. Therefore, Section C.IV.9 is not applicable to the APR1400.	All

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.207 Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors	03/2007	The APR1400 conforms with this NRC RG.	3.9.1.1, 3.9.1.2.1.12, 3.9.3.1, 3.12.5.7, 3.12.5.19
1.208 A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion	03/2007	Not applicable (COL)	N/A
1.209 Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants	03/2007	The APR1400 conforms with this NRC RG.	7.2.2.8
1.210 Qualification of Safety-Related Battery Chargers and Inverters for Nuclear Power Plants	06/2008	The APR1400 conforms with this NRC RG.	3.11
1.211 Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants	04/2009	The APR1400 conforms with this NRC RG.	3.11.2, 8.1.3.3
1.212 Sizing of Large Lead-Acid Storage Batteries	11/2008	The APR1400 conforms with this NRC RG except for the following. • IEEE Standard 485-2010 is applied instead of IEEE Standard 485-1997 because NRC RG 1.129 endorses the current standard (IEEE Standard 485-2010).	8.1.3.3, 8.3.2.2.2
1.213 Qualification of Safety-Related Motor Control Centers for Nuclear Power Plants	05/2009	The APR1400 conforms with this NRC RG.	3.11
1.215 Guidance for ITAAC Closure Under 10 CFR 52	Rev. 1 05/2012	The APR1400 conforms with this NRC RG.	14.3.2.3, 14.3.5
1.216 Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design-Basis Pressure	08/2010	The APR1400 conforms with this NRC RG.	19.2.1, 19.2.4.2.2

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NRC Regulatory Guide		Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.217	Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts	08/2011	The APR1400 conforms with this NRC RG.	19.5
1.218	Condition-Monitoring Techniques for Electric Cables Used in Nuclear Power Plants	04/2012	Not applicable (COL)	N/A
1.219	Guidance on Making Changes to Emergency Plans for Nuclear Power Reactors	11/2011	Not applicable (COL)	N/A
1.221	Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants	10/2011	The APR1400 conforms with this NRC RG.	Table 2.0-1, 3.3.2.1, 3.5.1.4
4.1	Radiological Environmental Monitoring for Nuclear Power Plants	Rev. 2 06/2009	Not applicable (COL)	N/A
4.2	Preparation of Environmental Reports for Nuclear Power Stations	Rev. 2 07/1976	Not applicable (COL)	N/A
4.2	Supplement 1 – Preparation of Supplemental Environmental Reports for Applications To Renew Nuclear Power Plant Operating Licenses	09/2000	Not applicable (COL)	N/A
4.7	General Site Suitability Criteria for Nuclear Power Stations	Rev. 3 03/2014	Not applicable (COL)	N/A
4.9	Preparation of Environmental Reports for Commercial Uranium Enrichment Facilities	Rev. 1 10/1975	Not applicable (COL)	N/A
4.11	Terrestrial Environmental Studies for Nuclear Power Stations	Rev. 2 03/2012	Not applicable (COL)	N/A
4.13	Performance, Testing, and Procedural Specifications for Thermo luminescence Dosimetry: Environmental Applications	Rev. 1 07/1977	Not applicable (COL)	N/A
4.14	Radiological Effluent and Environmental Monitoring at Uranium Mills	Rev. 1 04/1980	Not applicable	N/A

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NRC Regulatory Guide		Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
4.15	Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) – Effluent Streams and the Environment	Rev. 2 07/2007	Not applicable (COL)	N/A
4.16	Monitoring and Reporting Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Cycle Facilities	Rev. 2 12/2010	Not applicable	N/A
4.17	Standard Format and Content of Site Characterization Plans for High-Level-Waste Geologic Repositories	Rev. 1 03/1987	Not applicable	N/A
4.18	Standard Format and Content of Environmental Reports for Near-Surface Disposal of Radioactive Waste	06/1983	Not applicable	N/A
4.19	Guidance for Selecting Sites for Near-Surface Disposal of Low-Level Radioactive Waste	08/1988	Not applicable	N/A
4.20	Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors	Rev.1 04/2012	Not applicable	N/A

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NRC Regulatory Guide		Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
4.21	Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning	06/2008	The APR1400 conforms with this NRC RG	5.4.3.5, 5.4.7.4.4, 5.4.12.2.3, 6.2.5.2.3, 6.3.6, 6.5.2.2.1, Table 6.5-2, 6.8.2.1.3, 9.1.2.2.2, 9.1.3.2.3, 9.2.1.2.4, 9.2.2.2.5, 9.3.2.2.4, 9.3.3.2.6, 9.3.4.2.10, 9.4.6.1.1, 9.4.8, 10.3.2.4, 10.4.2.2.3, 10.4.6.2.4, 10.4.7.2.4, 10.4.8.2.4, 10.4.9.2.5, 11.2.1.1, 11.2.1.2, 11.2.2.4.1, 11.3.1.2, 11.3.2.2.2, 11.4, 11.4.2.5.1, 11.5.2.4, 12.1.3.1, 12.1.3.4, 12.3.1.1, 12.3.1.4, 12.3.1.10, 12.3.1.10, 12.3.1.10.1, 12.3.1.10.2, 12.3.1.10.3, Table 12.3-6, Table 12.3-7
4.22	Decommissioning Planning During Operations	12/2012	Not applicable	N/A
5.3	Statistical Terminology and Notation for Special Nuclear Materials Control and Accountability	02/1973	Not applicable	N/A
5.4	Standard Analytical Methods for the Measurement of Uranium Tetrafluoride (UF4) and Uranium Hexafluoride (UF6)	02/1973	Not applicable	N/A

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NRC Regulatory Guide		Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
5.5	Standard Methods for Chemical, Mass Spectrometric, and Spectrochemical Analysis of Nuclear-Grade Uranium Dioxide Powders and Pellets	02/1973	Not applicable	N/A
5.7	Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas	Rev. 1 05/1980	Not applicable (COL)	N/A
5.8	Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Drying and Fluidized Bed Operations	Rev. 1 05/1974	Not applicable	N/A
5.9	Guidelines for Germanium Spectroscopy Systems for Measurement of Special Nuclear Material	Rev. 2 12/1983	Not applicable	N/A
5.11	Nondestructive Assay of Special Nuclear Material Contained in Scrap and Waste	Rev. 1 04/1984	Not applicable	N/A
5.12	General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials	11/1973	Not applicable	N/A
5.13	Conduct of Nuclear Material Physical Inventories	11/1973	Not applicable	N/A
5.18	Limit of Error Concepts and Principles of Calculation in Nuclear Materials Control	01/1974	Not applicable	N/A
5.20	Training, Equipping, and Qualifying of Guards and Watchmen	01/1974	Not applicable	N/A
5.21	Nondestructive Uranium-235 Enrichment Assay by Gamma Ray Spectrometry	Rev. 1 12/1983	Not applicable	N/A
5.22	Assessment of the Assumption of Normality (Employing Individual Observed Values)	04/1974	Not applicable	N/A

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
5.23 In Situ Assay of Plutonium Residual Holdup	Rev. 1 02/1984	Not applicable	N/A
5.25 Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Equipment for Wet Process Operations	06/1974	Not applicable	N/A
5.26 Selection of Material Balance Areas and Item Control Areas	Rev. 1 04/1975	Not applicable	N/A
5.27 Special Nuclear Material Doorway Monitors	06/1974	Not applicable	N/A
5.28 Evaluation of Shipper-Receiver Differences in the Transfer of Special Nuclear Materials	06/1974	Not applicable	N/A
5.31 Specially Designed Vehicle with Armed Guards for Road Shipment of Special Nuclear Material	Rev. 1 04/1975	Not applicable	N/A
5.33 Statistical Evaluation of Material Unaccounted For	06/1974	Not applicable	N/A
5.34 Nondestructive Assay for Plutonium in Scrap Material by Spontaneous Fission Detection	Rev. 1 05/1984	Not applicable	N/A
5.36 Recommended Practice for Dealing with Outlying Observations	06/1974	Not applicable	N/A
5.37 In Situ Assay of Enriched Uranium Residual Holdup	Rev. 1 10/1983	Not applicable	N/A
5.38 Nondestructive Assay of High-Enrichment Uranium Fuel Plates by Gamma Ray Spectrometry	Rev. 1 10/1983	Not applicable	N/A
5.39 General Methods for the Analysis of Uranyl Nitrate Solutions for Assay, Isotopic Distribution, and Impurity Determinations	12/1974	Not applicable	N/A

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NRC Regulatory Guide		Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
5.42	Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Equipment for Dry Process Operations	01/1975	Not applicable	N/A
5.43	Plant Security Force Duties	01/1975	Not applicable	N/A
5.44	Perimeter Intrusion Alarm Systems	Rev. 3 10/1997	Not applicable	N/A
5.48	Design Considerations – Systems for Measuring the Mass of Liquids	02/1975	Not applicable	N/A
5.49	Internal Transfers of Special Nuclear Material	03/1975	Not applicable	N/A
5.51	Management Review of Nuclear Material Control and Accounting Systems	06/1975	Not applicable	N/A
5.52	Standard Format and Content of a Licensee Physical Protection Plan for Strategic Special Nuclear Material at Fixed Sites (Other than Nuclear Power Plants)	Rev. 3 12/1994	Not applicable	N/A
5.53	Qualification, Calibration, and Error Estimation Methods for Nondestructive Assay	Rev. 1 02/1984	Not applicable	N/A
5.55	Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities	03/1978	Not applicable	N/A
5.56	Standard Format and Content of Safeguards Contingency Plans for Transportation	03/1978	Not applicable	N/A
5.57	Shipping and Receiving Control of Strategic Special Nuclear Material	Rev. 1 06/1980	Not applicable	N/A
5.58	Considerations for Establishing Traceability of Special Nuclear Material Accounting Measurements	Rev. 1 02/1980	Not applicable	N/A

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
5.59 Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance	Rev. 1 02/1983	Not applicable	N/A
5.60 Standard Format and Content of a Licensee Physical Protection Plan for Strategic Special Nuclear Material in Transit	04/1980	Not applicable	N/A
5.61 Intent and Scope of the Physical Protection Upgrade Rule Requirements for Fixed Sites	06/1980	Not applicable	N/A
5.62 Reporting of Safeguards Events	Rev. 1 11/1987	Not applicable	N/A
5.63 Physical Protection for Transient Shipments	07/1982	Not applicable	N/A
5.65 Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls	09/1986	Not applicable (COL)	N/A
5.66 Access Authorization Program for Nuclear Power Plants	Rev. 2 10/2011	Not applicable (COL)	N/A
5.68 Protection Against Malevolent Use of Vehicles at Nuclear Power Plants	08/1994	Not applicable (COL)	N/A
5.71 Cyber Security Programs for Nuclear Facilities	01/2010	Not applicable (COL)	N/A
5.73 Fatigue Management for Nuclear Power Plant Personnel	03/2009	Not applicable (COL)	N/A
5.74 Managing the Safety/Security Interface	06/2009	Not applicable (COL)	N/A
5.75 Training and Qualification of Security Personnel at Nuclear Power Reactor Facilities	07/2009	Not applicable (COL)	N/A

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NRC Regulatory Guide		Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
5.79	Protection of Safeguards Information	04/2011	Not applicable (COL)	N/A
5.80	Pressure-Sensitive and Tamper-Indicating Device Seals for Material Control and Accounting of Special Nuclear Material	12/2010	Not applicable (COL)	N/A
8.2	Administrative Practices in Radiation Surveys and Monitoring	Rev. 1 05/2011	Not applicable (COL)	N/A
8.4	Personal Monitoring Device-Direct Reading Pocket Dosimeters	Rev. 1 06/2011	Not applicable (COL)	N/A
8.7	Instructions for Recording and Reporting Occupational Radiation Exposure Data	Rev. 2 11/2005	Not applicable (COL)	N/A
8.8	Information Relevant to Ensuring the Occupational Radiation Exposures at Nuclear Power Stations will be ALARA	Rev. 3 06/1978	The APR1400 conforms with this NRC RG except for the following. • Nickel-based alloy is used for SG tubes based on industry experience for similar applications in Korean domestic plants as described in Subsection 12.3.1.3.	Table 6.5-2, 10.4.6.1, 10.4.8.1.2, 11.2.1.2, 11.3.1.2, 11.3.1.3, 11.3.1.4, 11.4.1.2, 11.4.1.3, 11.4.2.4, 11.5.2.1, 11.5.3, 12.1.1.1, 12.1.2.1, 12.1.2.2, 12.1.3.2, 12.2.4, 12.3.1.1, 12.3.1.2, 12.3.1.3, 12.3.1.4, 12.3.1.5, 12.3.2.1, 12.3.2.3, 12.3.4
8.9	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program	Rev. 1 07/1993	Not applicable (COL)	N/A
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable	Rev. 1-R 05/1977	Not applicable (COL)	N/A

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
8.11 Applications of Bioassay for Uranium	06/1974	Not applicable	N/A
8.13 Instruction Concerning Prenatal Radiation Exposure	Rev. 3 06/1999	Not applicable (COL)	N/A
8.15 Acceptable Programs for Respiratory Protection	Rev. 1 10/1999	Not applicable (COL)	N/A
8.18 Information Relevant to Ensuring that Occupational Radiation Exposures at Medical Institutions Will Be as Low as Reasonably Achievable	Rev. 2 04/2011	Not applicable (COL)	N/A
8.19 Occupational Radiation Dose Assessment in Light-Water Reactor Plants-Design Stage Man-Rem Estimates	Rev. 1 06/1979	The APR1400 conforms with this NRC RG.	12.4.1.2
8.20 Applications of Bioassay for I-125 and I-131	Rev. 1 09/1979	Not applicable (COL)	N/A
8.21 Health Physics Surveys for Byproduct Material at NRC-Licensed Processing and Manufacturing Plants	Rev. 1 10/1979	Not applicable	N/A
8.22 Bioassay at Uranium Mills	Rev. 2 05/2014	Not applicable	N/A
8.23 Radiation Safety Surveys at Medical Institutions	Rev. 1 01/1981	Not applicable	N/A
8.24 Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication	Rev. 2 06/2012	Not applicable	N/A
8.25 Air Sampling in the Workplace	Rev. 1 06/1992	The APR1400 conforms with this NRC RG	12.3.1.4, 12.3.4

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NRC Regulatory Guide		Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
8.26	Applications of Bioassay for Fission and Activation Products	09/1980	Not applicable	N/A
8.27	Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants	03/1981	Not applicable	N/A
8.28	Audible-Alarm Dosimeters	08/1981	Not applicable	N/A
8.29	Instruction Concerning Risks from Occupational Radiation Exposure	Rev. 1 02/1996	Not applicable	N/A
8.30	Health Physics Surveys in Uranium Recovery Facilities	Rev. 1 05/2002	Not applicable	N/A
8.31	Information Relevant to Ensuring that Occupational Radiation Exposures at Uranium Recovery Facilities Will Be as Low as Is Reasonably Achievable	Rev. 1 05/2002	Not applicable	N/A
8.32	Criteria for Establishing a Tritium Bioassay Program	07/1988	Not applicable (COL)	N/A
8.34	Monitoring Criteria and Methods To Calculate Occupational Radiation Doses	07/1992	Not applicable (COL)	N/A
8.35	Planned Special Exposures	Rev. 1 08/2010	Not applicable (COL)	N/A
8.36	Radiation Dose to the Embryo/Fetus	07/1992	Not applicable (COL)	N/A
8.37	ALARA Levels for Effluents from Materials Facilities	07/1993	Not applicable (COL)	N/A

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NRC Regulatory Guide	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
8.38 Control of Access to High and Very High Radiation Areas of Nuclear Plants	Rev. 1 05/2006	The APR1400 conforms with this NRC RG.	Ch. 12
8.39 Release of Patients Administered Radioactive Materials	04/1997	Not applicable	N/A
8.40 Methods for Measuring Effective Dose Equivalent from External Exposure	07/2010	Not applicable (COL)	N/A

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APR1400 Conformance with the Standard Review Plan

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
1.0 – Introduction and Interfaces	Rev. 2 12/2011	The APR1400 conforms with this SRP.	Ch. 1
2.0 – Site Characteristics and Site Parameters	03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the combined license application (COLA).	Ch. 2
2.1.1 – Site Location and Description	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.1.1
2.1.2 – Exclusion Area Authority and Control	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.1.2
2.1.3 – Population Distribution	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.1.3
2.2.1 ~ 2.2.2 – Identification of Potential Hazards in Site Vicinity	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.2.1~2.2.2
2.2.3 – Evaluation of Potential Accidents	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.2.3
2.3.1 – Regional Climatology	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.3.1
2.3.2 – Local Meteorology	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.3.2
2.3.3 – Onsite Meteorological Measurements Program	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.3.3
2.3.4 – Short-Term Atmospheric Dispersion Estimates for Accident Releases	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.3.4

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SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
2.3.5 – Long-Term Atmospheric Dispersion Estimates for Routine Releases	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.3.5
2.4.1 – Hydrologic Description	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.4.1
2.4.2 – Floods	Rev. 4 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.4.2
2.4.3 – Probable Maximum Flood (PMF) on Streams and Rivers	Rev. 4 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.4.3
2.4.4 – Potential Dam Failures	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.4.4
2.4.5 – Probable Maximum Surge and Seiche Flooding	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.4.5
2.4.6 – Probable Maximum Tsunami Hazards	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.4.6
2.4.7 – Ice Effects	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.4.7
2.4.8 – Cooling Water Canals and Reservoirs	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.4.8
2.4.9 – Channel Diversions	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.4.9
2.4.10 – Flooding Protection Requirements	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.4.10

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SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
2.4.11 – Low Water Considerations	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.4.11
2.4.12 – Groundwater	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.4.12
2.4.13 – Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.4.13
2.4.14 – Technical Specifications and Emergency Operation Requirements	Rev. 3 03/2007	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.4.14
2.5.1 – Basic Geologic and seismic Information	Rev. 5 07/2014	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.5.1
2.5.2 – Vibratory Ground Motion	Rev. 5 07/2014	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.5.2
2.5.3 – Surface Faulting	Rev. 5 07/2014	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.5.3
2.5.4 – Stability of Subsurface Materials and Foundations	Rev. 5 07/2014	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.5.4
2.5.5 – Stability of Slopes	Rev. 5 07/2014	The APR1400 conforms with this SRP. The site-specific data will be addressed in the COLA.	2.5.5
3.2.1 – Seismic Classification	Rev. 2 03/2007	The APR1400 conforms with this SRP.	3.2.1
3.2.2 – System Quality Group Classification	Rev. 2 03/2007	The APR1400 conforms with this SRP.	3.2.2
3.3.1 – Wind Loading	Rev. 3 03/2007	The APR1400 conforms with this SRP.	3.3.1

Table 1.9-2 (4 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
3.3.2 – Tornado Loads	Rev. 3 03/2007	The APR1400 conforms with this SRP.	3.3.2
3.4.1 – Internal Flood Protection for Onsite Equipment Failures	Rev. 3 03/2007	The APR1400 conforms with this SRP.	3.4.1
3.4.2 – Analysis Procedures	Rev. 3 03/2007	The APR1400 conforms with this SRP.	3.4.2
3.5.1.1 – Internally Generated Missiles (Outside Containment)	Rev. 3 03/2007	The APR1400 conforms with this SRP.	3.5.1.1
3.5.1.2 – Internally-Generated Missiles (Inside Containment)	Rev. 3 03/2007	The APR1400 conforms with this SRP.	3.5.12
3.5.1.3 – Turbine Missiles	Rev. 3 03/2007	The APR1400 conforms with this SRP.	3.5.13
3.5.1.4 – Missiles Generated by Tornadoes and Extreme Winds	Rev.4 (Draft) 07/2013	The APR1400 conforms with this SRP.	3.5.1.4
3.5.1.5 – Site Proximity Missiles (Except Aircraft)	Rev. 4 03/2007	Not applicable (COL)	N/A
3.5.1.6 – Aircraft Hazards	Rev. 4 03/2010	Not applicable (COL)	N/A
3.5.2 – Structures, Systems, and Components to be Protected from Externally-Generated Missiles	Rev. 3 03/2007	The APR1400 conforms with this SRP.	3.5.2
3.5.3 – Barrier Design Procedures	Rev. 3 03/2007	The APR1400 conforms with this SRP.	3.5.3
3.6.1 – Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	Rev. 3 03/2007	The APR1400 conforms with this SRP.	3.6.1

Table 1.9-2 (5 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
3.6.2 – Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	Rev. 2 03/2007	The APR1400 conforms with this SRP.	3.6.2
3.6.3 – Leak-Before-Break Evaluation Procedures	Rev. 1 03/2007	The APR1400 conforms with this SRP.	3.6.3
3.7.1 – Seismic Design Parameters	Rev.4 (Draft) 12/2012	The APR1400 conforms with this SRP.	3.7.1
3.7.2 – Seismic System Analysis	Rev. 4 09/2013	The APR1400 conforms with this SRP except for limits on response reduction due to incoherent seismic input motion.	3.7.2
3.7.3 – Seismic Subsystem Analysis	Rev. 4 09/2013	Alternate analysis methods are employed for piping systems. No explicit range of the fundamental frequencies of components and equipment with respect to the dominant frequencies of the support structure is made.	3.7.3
3.7.4 – Seismic Instrumentation	Rev. 2 03/2007	The APR1400 conforms with this SRP.	3.7.4
3.8.1 – Concrete Containment	Rev. 4 09/2013	The APR1400 conforms with this SRP.	3.8.1
3.8.2 – Steel Containment	Rev. 3 05/2010	The APR1400 conforms with this SRP for the areas relating to Class MC steel portions of concrete containment.	3.8.2
3.8.3 – Concrete and Steel Internal Structures of Steel or Concrete Containments	Rev. 4 09/2013	The APR1400 conforms with this SRP.	3.8.3
3.8.4 – Other seismic Category I Structures	Rev. 4 09/2013	The APR1400 conforms with this SRP.	3.8.4

Table 1.9-2 (6 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
3.8.5 – Foundations	Rev. 4 09/2013	The APR1400 conforms with this SRP.	3.8.5
3.9.1 – Special Topics for Mechanical Components	Rev. 3 03/2007	The APR1400 conforms with this SRP.	3.9.1
3.9.2 – Dynamic Testing and Analysis of Systems, Structures, and Components	Rev. 3 03/2007	The APR1400 conforms with this SRP with the following exception: • Startup testing with measurement of SG internals	3.9.2, App. 3.9B
3.9.3 – ASME Code Class 1, 2, and 3 Components and Component Supports, and Core Support Structures	Rev. 3 04/2014	The APR1400 conforms with this SRP.	3.9.3
3.9.4 – Control Rod Drive Systems	Rev. 3 03/2007	The APR1400 conforms with this SRP.	3.9.4
3.9.5 – Reactor Pressure Vessel Internals	Rev. 3 03/2007	The APR1400 conforms with this SRP.	3.9.5
3.9.6 – Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	Rev. 3 03/2007	The APR1400 conforms with this SRP.	3.9.6
3.9.7 – Risk-Informed Inservice Testing	08/1998	The APR1400 conforms with this SRP.	3.9.7
3.9.8 – Risk-Informed Inservice Inspection of Piping	09/2003	The APR1400 conforms with this SRP.	3.9.8
3.10 – Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	Rev. 3 03/2007	The APR1400 conforms with this SRP.	3.10
3.11 – Environmental Qualification of Mechanical and Electrical Equipment	Rev. 3 03/2007	The APR1400 conforms with this SRP.	3.11
3.12 – ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports	Rev. 1 04/2014	The APR1400 conforms with this SRP.	3.12
3.13 – Threaded Fasteners-ASME Code Class 1, 2, and 3	03/2007	The APR1400 conforms with this SRP.	3.13

Table 1.9-2 (7 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
BTP 3-1 – Classification of Main Steam Components Other Than the Reactor Coolant Pressure Boundary for BWR Plants	Rev. 2 03/2007	Not applicable (BWRs only)	N/A
BTP 3-2 – Classification of BWR/6 Main Steam and Feedwater Components Other Than the Reactor Coolant Pressure Boundary	Rev. 2 03/2007	Not applicable (BWRs only)	N/A
BTP 3-3 – Protection against Postulated Piping Failures in Fluid Systems Outside Containment	Rev. 3 03/2007	The APR1400 conforms with this BTP.	3.6.1, 10.4.4.3
BTP 3-4 – Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment	Rev. 2 03/2007	The APR1400 conforms with this BTP.	3.6.2, 10.4.4.3
4.2 – Fuel System Design	Rev. 3 03/2007	The APR1400 conforms with this SRP.	4.2
4.3 – Nuclear Design	Rev. 3 03/2007	The APR1400 conforms with this SRP.	4.3
4.4 – Thermal and Hydraulic Design	Rev. 2 03/2007	The APR1400 conforms with this SRP.	4.4
4.5.1 – Control Rod Drive Structural Materials	Rev. 3 03/2007	The APR1400 conforms with this SRP except for the following: • The usage of control drive structural material with a yield strength greater than 90 ksi is limited to the steel ball in the vent valve on the top of the CEDMs, bearing inserts, and alignment tab in the motor assembly.	4.5.1
4.5.2 – Reactor Internal and Core Support Structure Materials	Rev. 3 03/2007	The APR1400 conforms with this SRP.	4.5.2

Table 1.9-2 (8 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
4.6 – Functional Design of Control Rod Drive System	Rev. 2 03/2007	The APR1400 conforms with this SRP.	3.6, 3.9.4, 4.6, 6.3, 7.2, 7.7, 9.3.4, 9.4, Chapter 14, Chapter 15
BTP 4-1 – Westinghouse Constant Axial Offset Control (CAOC)	Rev. 3 03/2007	Not applicable	N/A
5.2.1.1 – Compliance with the Codes and Standards Rule, 10 CFR 50.55a	Rev. 3 03/2007	The APR1400 conforms with this SRP.	5.2.1.1
5.2.1.2 – Applicable Code Cases	Rev. 3 03/2007	The APR1400 conforms with this SRP.	5.2.1.2
5.2.2 – Overpressure Protection	Rev. 3 03/2007	The APR1400 conforms with this SRP.	5.2.2
5.2.3 – Reactor Coolant Pressure Boundary Materials	Rev. 3 03/2007	The APR1400 conforms with this SRP except for the following: The electroslag weld process is not used in the fabrication of any RCPB components.	5.2.3
5.2.4 – Reactor Coolant Pressure Boundary Inservice Inspection and Testing	Rev. 2 03/2007	The APR1400 conforms with this SRP.	5.2.4
5.2.5 – Reactor Coolant Pressure Boundary Leakage Detection	Rev. 2 03/2007	The APR1400 conforms with this SRP.	5.2.5
5.3.1 – Reactor Vessel Materials	Rev. 2 03/2007	The APR1400 conforms with this SRP except for the following: Actual reactor vessel materials are tested at the time of material procurement. Test requirements are described in Subsection 5.3.1.5.	5.3.1
5.3.2 – Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	Rev. 2 03/2007	The APR1400 conforms with this SRP.	5.3.2

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SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
5.3.3 – Reactor Vessel Integrity	Rev. 2 03/2007	The APR1400 conforms with this SRP.	5.3.3
5.4 – Reactor Coolant System Component and Subsystem Design	Rev. 2 03/2007	The APR1400 conforms with this SRP.	5.4
5.4.1.1 – Pump Flywheel Integrity (PWR)	Rev. 3 05/2010	The APR1400 conforms with this SRP with the following exception: • Design stress criteria.	5.4.1.1
5.4.2.1 – Steam Generator Materials	Rev. 3 03/2007	The APR1400 conforms with this SRP.	5.4.2.1
5.4.2.2 – Steam Generator Program	Rev. 2 03/2007	The APR1400 conforms with this SRP.	5.4.2.6
5.4.6 – Reactor Core Isolation Cooling System (BWR)	Rev. 4 03/2007	Not applicable (BWRs only)	N/A
5.4.7 – Residual Heat Removal (RHR) System	Rev. 5 05/2010	The APR1400 conforms with this SRP.	5.4.7
5.4.8 – Reactor Water Cleanup System (BWR)	Rev. 3 03/2007	Not applicable (BWRs only)	N/A
5.4.11 – Pressurizer Relief Tank	Rev. 4 05/2010	The APR1400 conforms with this SRP.	5.4.11
5.4.12 – Reactor Coolant System High Point Vents	Rev. 1 03/2007	The APR1400 conforms with this SRP.	5.4.12
5.4.13 – Isolation Condenser System (BWR)	03/2007	Not applicable (BWRs only)	N/A
BTP 5-1 – Monitoring of Secondary Side Water Chemistry in PWR Steam Generators	Rev. 3 03/2007	The APR1400 conforms with this BTP.	10.4.8.3

Table 1.9-2 (10 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
BTP 5-2 – Overpressurization Protection of Pressurized-Water Reactors While Operating at Low Temperatures	Rev. 3 03/2007	The APR1400 conforms with this BTP.	5.2.2.1.1 5.2.2.1.2
BTP 5-3 – Fracture Toughness Requirements	Rev. 2 03/2007	The APR1400 conforms with this BTP.	5.2.3.3, 5.3.1.1, 5.3.1.5, 5.3.1.6, 5.3.2, 5.3.2.3, 5.3.2.4
BTP 5-4 – Design Requirements of the Residual Heat Removal System	Rev. 4 03/2007	The APR1400 conforms with this BTP.	5.4.7.1.2, 5.2.2.1.1
6.1.1 – Engineered Safety Features Materials	Rev. 2 03/2007	The APR1400 conforms with this SRP.	6.1.1
6.1.2 – Protective Coating Systems (Paints)-Organic Materials	Rev. 3 03/2007	The APR1400 conforms with this SRP.	6.1.2
6.2.1 – Containment Functional Design	Rev. 3 03/2007	The APR1400 conforms with this SRP.	6.2.1
6.2.1.1.A – PWR Dry Containments, Including Subatmospheric Containments	Rev. 3 03/2007	The APR1400 Conforms with this SRP. As for Criterion 9, the structural design pressure of each subcompartment is determined based on the design experience.	6.2.1
6.2.1.1.B – Ice Condenser Containments	Rev. 2 07/1981	Not applicable	N/A
6.2.1.1.C – Pressure-Suppression Type BWR Containments	Rev. 7 03/2007	Not applicable (BWR)	N/A
6.2.1.2 – Subcompartment Analysis	Rev. 3 03/2007	The APR1400 conforms with this SRP.	6.2.1.2

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SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
6.2.1.3 – Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)	Rev. 3 03/2007	The APR1400 conforms with this SRP except for the following: Metal-water reaction energy is not included in the mass/energy source terms since this energy has been shown to have a small effect on the containment pressure.	6.2.1.3
6.2.1.4 – Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	Rev. 2 03/2007	The APR1400 conforms with this SRP.	6.2.1.4
6.2.1.5 – Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	Rev. 3 03/2007	The APR1400 conforms with this SRP.	6.2.1.5
6.2.2 – Containment Heat Removal Systems	Rev. 5 03/2007	Conformance with exceptions. Criterion 4 is not applied to APR1400, because the APR1400 does not have the fan cooler system for containment heat removal following the design base accident.	6.2.2
6.2.3 – Secondary Containment Functional Design	Rev. 3 03/2007	Not applicable The APR1400 does not have a secondary containment.	N/A
6.2.4 – Containment Isolation System	Rev. 3 03/2007	The APR1400 conforms with this SRP. The Chapter 15 dose analysis showed the acceptability of 30-second closure times for the purge valves.	6.2.4
6.2.5 –Combustible Gas Control in Containment	Rev. 3 03/2007	The APR1400 conforms with this SRP.	6.2.5
6.2.6 – Containment Leakage Testing	Rev. 3 03/2007	The APR1400 conforms with this SRP.	6.2.6
6.2.7 – Fracture Prevention of Containment Pressure Boundary	Rev. 1 03/2007	The APR1400 conforms with this SRP.	6.2.7

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SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
6.3 – Emergency Core Cooling System	Rev. 3 03/2007	The APR1400 conforms with exceptions relevant to criteria applied to BWRs. BTP 6-5 Item E is applied to traditional PWR with a switchover from the injection mode to the recirculation cooling mode.	6.3
6.4 – Control Room Habitability System	Rev. 3 03/2007	The APR1400 conforms with this SRP except for the following. The control room habitability during a postulated hazardous chemical release is addressed in COLA.	6.4
6.5.1 – Engineered Safety Features (ESF) Atmosphere Cleanup Systems	Rev. 4 05/2010	The APR1400 conforms with this SRP.	6.5.1
6.5.2 – Containment Spray as a Fission Product Cleanup System	Rev. 4 03/2007	The APR1400 conforms with this SRP. Conformance with exceptions. Criterion 3B is not applied to APR1400, because the APR1400 does not have the containment spray chemical additive tanks.	6.5.2
6.5.3 – Fission Product Control Systems and Structures	Rev. 3 03/2007	The APR1400 analysis assumes more than 50% mixing.	6.5.3
6.5.4 – Ice Condenser as a Fission Product Cleanup System	Rev. 3 12/1988	Not applicable	N/A
6.5.5 – Pressure Suppression Pool as a Fission Product Cleanup System	Rev. 1 03/2007	Not applicable (BWRs only) The APR1400 has adopted the Containment Spray System as Fission Product Cleanup System to control the fission products released during postulated reactor accidents	N/A
6.6 – Inservice Inspection and Testing of Class 2 and 3 Components	Rev. 2 03/2007	The APR1400 conforms with this SRP.	6.6
6.7 – Main Steam Isolation Valve Leakage Control System (BWR)	Rev. 2 07/1981	Not applicable (BWR)	N/A
BTP 6-1 – pH for Emergency Coolant Water for Pressurized Water Reactors	03/2007	The APR1400 conforms with this BTP.	6.5.2

Table 1.9-2 (13 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
BTP 6-2 – Minimum Containment Pressure model for PWR ECCS Performance Evaluation	Rev. 3 03/2007	The APR1400 conforms with this BTP.	6.2
BTP 6-3 – Determination of Bypass Leakage Paths in Dual Containment Plants	Rev. 3 03/2007	Not applicable The APR1400 does not have a dual containment.	N/A
BTP 6-4 – Containment Purging during Normal Plant Operations	Rev. 3 03/2007	The APR1400 conforms with this BTP.	9.4.6.2
BTP 6-5 – Currently the Responsibility of Reactor Systems Piping from the RWST (or BWST) and Containment Sump(s) to the Safety Injection Pumps	Rev. 3 03/2007	The APR1400 design conforms with this BTP except for the following: The APR1400 adopted the IRWST (In-containment Refueling Water Storage Tank) design feature. Therefore, containment sumps and the recirculation mode concept are not applied in the APR1400 design.	6.3.2.5.2
7.0 – Instrumentation and Controls – Overview of Review Process	Rev. 6 05/2010	The APR1400 conforms with this SRP.	7.1
App. 7.0-A – Review Process for Digital Instrumentation and Control Systems	Rev. 5 03/2007	The APR1400 conforms with this SRP.	7.1
7.1 – Instrumentation and Controls – Introduction	Rev. 5 03/2007	The APR1400 conforms with this SRP.	7.1
7.1-T – Table 7-1 Regulatory Requirements, Acceptance Criteria, and Guidelines for Instrumentation and Control Systems Important to Safety	Rev. 5 03/2007	The APR1400 conforms with this SRP.	7.1
App. 7.1-A – Acceptance Criteria and Guidelines for Instrumentation and Control Systems Important to Safety	Rev. 5 03/2007	The APR1400 conforms with this SRP.	7.1
App. 7.1-B – Guidance for Evaluation of Conformance to IEEE Std. 279	Rev. 5 03/2007	Not applicable	N/A

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SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
App. 7.1-C – Guidance for Evaluation of Conformance to IEEE Std. 603	Rev. 5 03/2007	The APR1400 conforms with this SRP.	7.1
App. 7.1-D – Guidance for Evaluation of the Application of IEEE Std. 7-4.3.2	03/2007	The APR1400 conforms with this SRP.	7.1
7.2 – Reactor Trip System	Rev. 5 03/2007	The APR1400 conforms with this SRP.	7.2
7.3 – Engineered Safety Features Systems	Rev. 5 03/2007	The APR1400 conforms with this SRP.	7.3
7.4 – Safe Shutdown Systems	Rev. 5 03/2007	The APR1400 conforms with this SRP.	7.4
7.5 – Information Systems Important to Safety	Rev. 5 03/2007	The APR1400 conforms with this SRP.	7.5
7.6 – Interlock Systems Important to Safety	Rev. 5 03/2007	The APR1400 conforms with this SRP except for the following: Interlocks for shutdown cooling system (SCS) suction isolation valves are not diverse.	7.6
7.7 – Control Systems	Rev. 5 03/2007	The APR1400 conforms with this SRP.	7.7
7.8 – Diverse Instrumentation and Control Systems	Rev. 5 03/2007	The APR1400 conforms with this SRP.	7.8
7.9 – Data Communication Systems	Rev. 5 03/2007	The APR1400 conforms with this SRP.	7.9

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SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
BTP 7-1 – Guidance on Isolation of Low-Pressure Systems from the High-Pressure Reactor Coolant System	Rev. 5 03/2007	The APR1400 conforms with this BTP.	7.1.2.58, Table 7.1-1
BTP 7-2 – Guidance on Requirements of Motor-Operated Valves in the Emergency Core Cooling System Accumulator Lines	Rev. 5 03/2007	The APR1400 conforms with this BTP.	7.1.2.59, Table 7.1-1
BTP 7-3 – Guidance on Protection System Trip Point Changes for Operation With Reactor Coolant Pumps Out of Service	Rev. 5 03/2007	The APR1400 conforms with this BTP.	N/A
BTP 7-4 – Guidance on Design Criteria for Auxiliary Feedwater Systems	Rev. 5 03/2007	The APR1400 conforms with this BTP.	7.1.2.61, Table 7.1-1
BTP 7-5 – Guidance on Spurious Withdrawals of Single Control Rods in Pressurized Water Reactors	Rev. 5 03/2007	The APR1400 conforms with this BTP.	7.1.2.62, Table 7.1-1
BTP 7-6 – Guidance on Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recirculation Mode	Rev. 5 03/2007	The APR1400 adopted the in-containment refueling water storage tank (IRWST). Therefore, containment sumps and the recirculation mode concept are not applied in the APR1400.	N/A
BTP 7-8 – Guidance for Application of Regulatory Guide 1.22	Rev. 5 03/2007	The APR1400 conforms with this BTP.	7.1.2.64, Table 7.1-1

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SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
BTP 7-9 – Guidance on Requirements for Reactor Protection System Anticipatory Trips	Rev. 5 03/2007	The APR1400 conforms with this BTP.	7.1.2.65, Table 7.1-1
BTP 7-10 – Guidance on Application of Regulatory Guide 1.97	Rev. 5 03/2007	The APR1400 conforms with this BTP.	7.1.2.66, Table 7.1-1
BTP 7-11 – Guidance on Application and Qualification of Isolation Devices	Rev. 5 03/2007	The APR1400 conforms with this BTP.	7.1.2.67, Table 7.1-1
BTP 7-12 – Guidance on Establishing and Maintaining Instrument Setpoints	Rev. 5 03/2007	The APR1400 conforms with this BTP except for acceptance Criterion 4. For the APR1400, surveillance and calibration interval is set by 18-months as considering overhaul interval.	7.1.2.68, Table 7.1-1
BTP 7-13 – Guidance on Cross-Calibration of Protection System Resistance Temperature Detectors	Rev. 5 03/2007	The APR1400 conforms with this BTP.	7.1.2.69, Table 7.1-1
BTP 7-14 – Guidance on Software Reviews for Digital Computer-Based Instrumentation and Controls Systems	Rev. 5 03/2007	The APR1400 conforms with this BTP.	7.1.2.70, Table 7.1-1
BTP 7-17 – Guidance on Self-Test and Surveillance Test Provisions	Rev. 5 03/2007	The APR1400 conforms with this BTP.	7.1.2.71, Table 7.1-1
BTP 7-18 – Guidance on the Use of Programmable Logic Controllers in Digital Computer-Based Instrumentation and Control Systems	Rev. 5 03/2007	The APR1400 conforms with this BTP.	7.1.2.72, Table 7.1-1
BTP 7-19 – Guidance for Evaluation of Diversity and Defense-In-Depth in Digital Computer-Based Instrumentation and Control Systems	Rev. 6 07/2012	The APR1400 conforms with this BTP.	7.1.2.73, Table 7.1-1, 7.3.2.4, 7.8.2.1 7.8.2.2, 7.8.2.3, 7.8.3.1
BTP 7-21 – Guidance on Digital Computer Real-Time Performance	Rev. 5 03/2007	The APR1400 conforms with this BTP.	7.1.2.74, Table 7.1-1

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SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
8.1 – Electric Power – Introduction	Rev. 4 02/2012	The APR1400 conforms with this SRP.	8.1
8.2 – Offsite Power System	Rev. 5 05/2010	The APR1400 conforms with this SRP.	8.2
8.3.1 – AC Power Systems (Onsite)	Rev. 4 05/2010	The APR1400 conforms with this SRP.	8.3.1
8.3.2 – DC Power Systems (Onsite)	Rev. 4 05/2010	The APR1400 conforms with this SRP.	8.3.2
8.4 – Station Blackout	Rev. 1 05/2010	The APR1400 conforms with this SRP.	8.4
App. 8-A – General Agenda, Station Site Visits	Rev. 1 03/2007	Not applicable (COL)	N/A
BTP 8-1 – Requirements on Motor-Operated Valves in the ECCS Accumulator Lines	Rev. 3 03/2007	The APR1400 conforms with this BTP.	8.1.3.3, Table 8.1-2
BTP 8-2 – Use of Diesel Generator Sets for Peaking	Rev. 3 03/2007	The emergency diesel generator (EDG) provides backup power to the safety-related loads for safety shutdown during a loss of offsite power (LOOP). However, the EDG is not used for peaking service for offsite power system. The APR1400 conforms with this BTP.	8.1.3.3, Table 8.1-2
BTP 8-3 – Stability of Offsite Power Systems	Rev. 3 03/2007	Not applicable (COL)	N/A
BTP 8-4 – Application of the Single Failure Criterion to Manually Controlled Electrically Operated Valves	Rev. 3 03/2007	The APR1400 conforms with this BTP.	8.1.3.3, Table 8.1-2

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SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
BTP 8-5 – Supplemental Guidance for Bypass and Inoperable Status Indication for Engineered Safety Features Systems	Rev. 3 03/2007	The APR1400 conforms with this BTP.	8.1.3.3, 8.3.1.2.2, 8.3.2.2.2 Table 8.1-2
BTP 8-6 – Adequacy of Station Electric Distribution System Voltages	Rev. 3 03/2007	The APR1400 conforms with this BTP with the exception of B.1. The Class 1E distribution system is separated from the offsite power system by the secondary undervoltage relay regardless of the occurrence of an SIAS.	8.1.3.3, 8.2.2.3, 8.3.1.1.2.3, 8.3.1.1.3.12, Table 8.1-2
BTP 8-7 – Criteria for Alarms and Indications Associated with Diesel-Generator Unit Bypassed and Inoperable Status	Rev. 3 03/2007	The APR1400 conforms with this BTP.	8.1.3.3, 8.3.1.1.3 Table 8.1-2
BTP 8-8 – Onsite (Emergency Diesel Generators) and Offsite Power Sources Allowed Outage Time Extensions	02/2012	Not applicable	N/A
9.1.1 – Criticality Safety of Fresh and Spent Fuel Storage and Handling	Rev. 3 03/2007	The APR1400 conforms with this SRP.	9.1.1
9.1.2 – New and Spent Fuel Storage	Rev. 4 03/2007	The APR1400 conforms with this SRP.	9.1.2
9.1.3 – Spent Fuel Pool Cooling and Cleanup System	Rev. 2 03/2007	The APR1400 conforms with this SRP.	9.1.3
9.1.4 – Light Load Handling System and Refueling Cavity	Rev. 4 07/2014	The APR1400 conformance with acceptance criteria 5 is not applicable for the APR1400 design certification. (APR1400 is a single unit.)	9.1.4
9.1.5 – Overhead Heavy Load Handling Systems	Rev. 1 03/2007	The APR1400 conformance with exceptions. Criterion 5 is not applicable for the APR1400 design certification. (APR1400 is a single unit.)	9.1.5

Table 1.9-2 (19 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
9.2.1 – Station Service Water System	Rev. 5 03/2007	The APR1400 conforms with this SRP.	9.2.1
9.2.2 – Reactor Auxiliary Cooling Water System	Rev. 4 03/2007	The APR1400 conforms with this SRP.	9.2.2
9.2.4 – Potable and Sanitary Water Systems	Rev. 3 03/2007	The APR1400 conforms with this SRP.	9.2.4
9.2.5 – Ultimate Heat Sink	Rev. 3 03/2007	The APR1400 conforms with this SRP.	9.2.5
9.2.6 – Condensate Storage Facilities	Rev. 3 03/2007	Not applicable. Condensate storage facilities have no safety-related functions and handle nonradioactive fluid. The APR1400 is not multi-unit.	9.2.6
9.3.1 – Compressed Air System	Rev. 2 03/2007	The APR1400 conforms with this SRP.	9.3.1
9.3.2 – Process and Post-Accident Sampling Systems	Rev. 3 03/2007	The APR1400 conforms with this SRP.	9.3.2
9.3.3 – Equipment and Floor Drainage System	Rev. 3 03/2007	The APR1400 conforms with this SRP.	9.3.3
9.3.4 – Chemical and Volume Control System (PWR) (Including Boron Recovery System)	Rev. 3 03/2007	The APR1400 conforms with this SRP.	9.3.4
9.3.5 – Standby Liquid Control System (BWR)	Rev. 3 03/2007	Not applicable (BWR)	N/A
9.4.1 – Control Room Area Ventilation System	Rev. 3 03/2007	The APR1400 conforms with this SRP.	9.4.1

Table 1.9-2 (20 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
9.4.2 – Spent Fuel Pool Area Ventilation System	Rev. 3 03/2007	The APR1400 conformance with exceptions. Criterion 2 is not applicable (Not multiple unit plants)	9.4.2
9.4.3 – Auxiliary and Radwaste Area Ventilation System	Rev. 3 03/2007	The APR1400 conforms with this SRP.	9.4.3, 9.4.7
9.4.4 – Turbine Area Ventilation System	Rev. 3 03/2007	The APR1400 conforms with this SRP.	9.4.4
9.4.5 – Engineered Safety Feature Ventilation System	Rev. 3 03/2007	The APR1400 conformance with exceptions. Criterion 3: Not multiple unit plants. Criterion 5: Air cleanup function is provided for aux. building controlled area HVAC system only.	9.4.5
9.5.1.1 – Fire Protection Program	02/2009	The APR1400 conforms with this SRP.	9.5.1
9.5.1.2 – Risk-Informed (RI), Performance-Based (PB) Fire Protection Program (FPP)	12/2009	Not applicable. This SRP is allowed to the operating nuclear power reactor licensees to adopt risk-informed, performance-based approach as an alternative to the existing deterministic fire protection requirement. APR 1400 fire protection is designed to the requirements of SRP 9.5.1.1 which provides deterministic fire protection guidance, it is not necessary to incorporate this SRP.	N/A
9.5.2 – Communications Systems	Rev. 3 03/2007	The APR1400 conforms with exceptions. Acceptance criteria 1, 2, 3, 12, 13, and 14 refer to site-specific emergency response and security requirements that are the responsibility of the COL applicant.	9.5.2
9.5.3 – Lighting Systems	Rev. 3 03/2007	The APR1400 conforms with this SRP.	9.5.3
9.5.4 – Emergency Diesel Engine Fuel Oil Storage and Transfer System	Rev. 3 03/2007	The APR1400 conforms with this SRP.	9.5.4

Table 1.9-2 (21 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
9.5.5 – Emergency Diesel Engine Cooling Water System	Rev. 3 03/2007	The APR1400 conforms with this SRP.	9.5.5
9.5.6 – Emergency Diesel Engine Starting System	Rev. 3 03/2007	The APR1400 conforms with this SRP.	9.5.6
9.5.7 – Emergency Diesel Engine Lubrication System	Rev. 3 03/2007	The APR1400 conforms with this SRP.	9.5.7
9.5.8 – Emergency Diesel Engine Combustion Air Intake and Exhaust System	Rev. 3 03/2007	The APR1400 conforms with this SRP.	9.5.8
10.2 – Turbine Generator	Rev. 3 03/2007	The APR1400 conformance with exceptions. SRP 10.2 II. Acceptance Criteria 3 is not applicable because there is no safety-related equipment in the turbine room.	10.2
10.2.3 – Turbine Rotor Integrity	Rev. 2 03/2007	The APR1400 conforms with this SRP except FATT and Charpy V-notch energy in the material selection. The values specified in SRP 10.2.3.II.1 are based on material acceptance data taken from specimens at the surface of a shrunk-on wheel forgings. The values in the APR1400 DCD Tier 2 are specified that FATT is no higher than –1 °C (30 °F) and Cv energy is at least 6.22 kg-m (45 ft-lbs). These values are based on deep-seated specimens near the center of the integral rotor forging. (The material testing has shown that FATT increases from the outer surface to the deep-seated region of the forging as a result of variation in the cooling rate during the quenching process.)	10.2.3

Table 1.9-2 (22 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
10.3 – Main Steam Supply System	Rev. 4 03/2007	The APR1400 conforms with this SRP.	10.3
10.3.6 – Steam and Feedwater System Materials	Rev. 3 03/2007	The APR1400 conforms with this SRP.	10.3.6
10.4.1 – Main Condensers	Rev. 3 03/2007	The APR1400 conforms with this SRP.	10.4.1
10.4.2 – Main Condenser Evacuation System	Rev. 3 03/2007	The APR1400 conformance with exceptions. Criterion 1 refers to a potential for explosive mixtures and the APR1400 has no potential for explosive mixtures.	10.4.2
10.4.3 – Turbine Gland Sealing System	Rev. 3 03/2007	The APR1400 conforms with this SRP.	10.4.3
10.4.4 – Turbine Bypass System	Rev. 3 03/2007	The APR1400 conforms with this SRP.	10.4.4
10.4.5 – Circulating Water System	Rev. 3 03/2007	System is site-specific and is addressed with interface requirements.	10.4.5
10.4.6 – Condensate Cleanup System	Rev. 3 03/2007	The APR1400 conforms with this SRP.	10.4.6
10.4.7 – Condensate and Feedwater System	Rev. 4 03/2007	The APR1400 conformance with exceptions. Criterion 3 refers to shared systems and the APR1400 is a single unit design. Criterion 7 is defined as COL item in DCD subsection 10.3.6. Criterion 8 is for BWR.	10.4.7
10.4.8 – Steam Generator Blowdown System (PWR)	Rev. 3 03/2007	The APR1400 conforms with this SRP.	10.4.8

Table 1.9-2 (23 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
10.4.9 – Auxiliary Feedwater System (PWR)	Rev. 3 03/2007	The APR1400 conforms with this SRP.	10.4.9
BTP 10-1 – Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants	Rev. 3 03/2007	The APR1400 conforms with this BTP.	10.4.9
BTP 10-2 – Design Guidelines for Avoiding Water Hammers in Steam Generators	Rev. 4 03/2007	The APR1400 conforms with this BTP.	10.4.7.6, 10.4.9.1.2, 10.4.9.3
11.1 – Source Terms	Rev. 3 03/2007	The APR1400 conforms with this SRP.	11.1
11.2 – Liquid Waste Management Systems	Rev. 4 05/2010	The APR1400 conforms with this SRP. However, cost-benefit analysis for liquid waste management systems is deferred to the site-specific application due to the site-specific nature of population dose analyses. The plant transients that might occur less frequently than once per fuel cycle are not taken into account for the design of waste collection tanks and waste sample tanks.	11.2
11.3 – Gaseous Waste Management System	Rev. 3 03/2007	The APR1400 conforms with this SRP. However, cost-benefit analysis for gaseous waste management systems is deferred to the site-specific application.	11.3
11.4 – Solid Waste Management System	Rev. 3 03/2007	The APR1400 conforms with this SRP. However, cost-benefit analysis for gaseous waste management systems is deferred to the site-specific application.	11.4
11.5 – Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	Rev. 5 05/2010	The APR1400 conforms with this SRP.	11.5

Table 1.9-2 (24 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
BTP 11-3 – Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants	Rev. 3 03/2007	The APR1400 conforms with this BTP.	11.4
BTP 11-5 – Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure	Rev. 3 03/2007	The APR1400 conforms with this BTP.	11.3.3.2
BTP 11-6 – Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	Rev. 3 03/2007	The APR1400 conforms with this BTP except for the site-specific features. The COL applicant is to provide the site-specific geology and hydrology.	11.2.3.2
12.1 – Assuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable	Rev. 4 09/2013	The APR1400 conforms with this SRP except for the acceptance Criteria 3 and 4. These criteria are addressed in the COLA.	12.1
12.2 – Radiation Sources	Rev. 4 09/2013	The APR1400 conforms with this SRP.	12.2
12.3-12.4 – Radiation Protection Design Features	Rev. 5 09/2013	The APR1400 conforms with this SRP.	12.3, 12.4
12.5 – Operational Radiation Protection Program	Rev. 5 09/2013	Not applicable (COL)	N/A
13.1.1 – Management and Technical Support Organization	Rev. 5 03/2007	Not applicable (COL)	N/A
13.1.2-13.1.3 – Operating Organization	Rev. 6 03/2007	Not applicable (COL)	N/A

Table 1.9-2 (25 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
13.2.1 – Reactor Operator Requalification Program; Reactor Operator Training	Rev. 3 03/2007	Not applicable (COL)	N/A
13.2.2 – Non-Licensed Plant Staff Training	Rev. 3 03/2007	Not applicable (COL)	N/A
13.3 – Emergency Planning	Rev. 3 03/2007	Conformance with an exception. Under COL applicant's responsibility. The design feature, facilities, functions, and equipment necessary for emergency planning are included in DCD Tier 2.	13.3
13.4 – Operational Programs	Rev. 3 03/2007	Not applicable (COL)	N/A
13.5.1.1 – Administrative Procedures-General	Rev.1 12/2011	Not applicable (COL)	N/A
13.5.2.1 – Operating and Emergency Operating Procedures	Rev. 2 03/2007	Not applicable (COL)	N/A
13.6 – Physical Security	Rev. 3 03/2007	The APR1400 conforms with this SRP.	13.6
13.6.1 – Physical Security – Combined License and Operating Reactors	Rev. 1 10/2010	Not applicable (COL)	N/A
13.6.2 – Physical Security – Design Certification	Rev. 1 10/2010	The APR1400 conforms with this SRP.	13.6.2
13.6.3 – Physical Security – Early Site Permit	Rev. 1 10/2010	Not applicable (COL)	N/A
13.6.6 – Cyber Security Plan	11/2010	Not applicable (COL)	N/A

Table 1.9-2 (26 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
14.2 – Initial Plant Test Program – Design Certification and New License Applicants	Rev. 3 03/2007	The APR1400 conforms with this SRP.	14.2
14.2.1 – Generic Guidelines for Extended Power Uprate Testing Programs	08/2006	Not applicable	N/A
14.3 – Inspections, Tests, Analyses, and Acceptance Criteria	03/2007	The APR1400 conforms with this SRP.	14.3
14.3.2 – Structural and Systems Engineering – Inspections, Tests, Analyses, and Acceptance Criteria	03/2007	The APR1400 conforms with this SRP.	14.3.2.2
14.3.3 – Piping Systems and Components – Inspections, Tests, Analyses, and Acceptance Criteria	03/2007	The APR1400 conforms with this SRP.	14.3.2.3
14.3.4 – Reactor Systems – Inspections, Tests, Analyses, and Acceptance Criteria	03/2007	The APR1400 conforms with this SRP.	14.3.2.4
14.3.5 – Instrumentation and Controls – Inspections, Tests, Analyses, and Acceptance Criteria	03/2007	The APR1400 conforms with this SRP.	14.3.2.5
14.3.6 – Electrical Systems – Inspections, Tests, Analyses, and Acceptance Criteria	03/2007	The APR1400 conforms with this SRP.	14.3.2.6
14.3.7 – Plant Systems – Inspections, Tests, Analyses, and Acceptance Criteria	03/2007	The APR1400 conforms with this SRP.	14.3.2.7
14.3.8 – Radiation Protection – Inspections, Tests, Analyses, and Acceptance Criteria	03/2007	The APR1400 conforms with this SRP.	14.3.2.8

Table 1.9 – 2 (27 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
14.3.9 – Human Factors Engineering – Inspections, Tests, Analyses, and Acceptance Criteria	03/2007	The APR1400 conforms with this SRP.	14.3.2.9 18.4 ~ 18.12
14.3.10 – Emergency Planning – Inspections, Tests, Analyses, and Acceptance Criteria	03/2007	The APR1400 conforms with this SRP.	14.3.2.10 18.4 ~ 18.12
14.3.11 – Containment Systems – Inspections, Tests, Analyses, and Acceptance Criteria	03/2007	The APR1400 conforms with this SRP.	14.3.2.11
14.3.12 – Physical Security Hardware – Inspections, Tests, Analyses, and Acceptance Criteria	Rev. 1 05/2010	The APR1400 conforms with this SRP.	14.3.2.12
15.0 – Introduction – Transient and Accident Analyses	Rev. 3 03/2007	The APR1400 conforms with this SRP.	15.0
15.0.1 – Radiological Consequence Analyses Using Alternative Source Terms	07/2000	Not applicable. This SRP applies to operating plants adopting alternative source term inputs.	N/A
15.0.2 – Review of Transient and Accident Analysis Method	03/2007	The APR1400 conforms with this SRP.	15.02
15.0.3 – Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors	03/2007	The APR1400 conforms with this SRP.	15.0.2
15.1.1-15.1.4 – Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	Rev. 2 03/2007	The APR1400 conforms with this SRP.	15.1.1 ~ 15.1.4
15.1.5 – Steam System Piping Failures Inside and Outside of Containment (PWR)	Rev. 3 03/2007	The APR1400 conforms with this SRP.	15.1.5

Table 1.9-2 (28 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
15.1.5.A – Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR	Rev. 2 07/1981	Not applicable. SRP 15.0.3, “Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors” is applied instead of SRP 15.1.5.A.	15.0.3
15.2.1–15.2.5 – Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	Rev. 2 03/2007	The APR1400 conforms with this SRP.	15.2.1 ~ 15.2.5
15.2.6 – Loss of Nonemergency AC Power to the Station Auxiliaries	Rev. 2 03/2007	The APR1400 conforms with this SRP.	15.2.6
15.2.7 – Loss of Normal Feedwater Flow	Rev. 2 03/2007	The APR1400 conforms with this SRP.	15.2.7
15.2.8 – Feedwater System Pipe Break Inside and Outside Containment (PWR)	Rev. 2 03/2007	The APR1400 conforms with this SRP.	15.2.8
15.3.1-15.3.2 – Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	Rev. 2 03/2007	The APR1400 conforms with this SRP.	15.3.1, 15.3.2
15.3.3-15.3.4 – Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	Rev. 3 03/2007	The APR1400 conforms with this SRP.	15.3.3, 15.3.4

Table 1.9-2 (29 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
15.4.1 – Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	Rev. 3 03/2007	The APR1400 conforms with this SRP.	15.4.1
15.4.2 – Uncontrolled Control Rod Assembly Withdrawal at Power	Rev. 3 03/2007	The APR1400 conforms with this SRP.	15.4.2
15.4.3 – Control Rod Misoperation (System Malfunction or Operator Error)	Rev. 3 03/2007	The APR1400 conforms with this SRP.	15.4.3
15.4.4–15.4.5 – Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction causing an Increase in BWR Core Flow Rate	Rev. 2 03/2007	The APR1400 conforms with this SRP.	15.4.4, 15.4.5
15.4.6 – Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR)	Rev. 2 03/2007	The APR1400 conforms with this SRP.	15.4.6
15.4.7 – Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	Rev. 2 03/2007	The APR1400 conforms with this SRP.	15.4.7
15.4.8 – Spectrum of Rod Ejection Accidents (PWR)	Rev. 3 03/2007	The APR1400 conforms with this SRP.	15.4.8
15.4.8.A – Radiological Consequences of a Control Rod Ejection Accident (PWR)	Rev. 1 07/1981	Not applicable. SRP 15.0.3, “Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors,” is applied instead of SRP 15.4.8.A.	15.0.3

Table 1.9-2 (30 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
15.4.9 – Spectrum of Rod Drop Accidents (BWR)	Rev. 3 03/2007	Not applicable (BWR)	N/A
15.4.9.A – Radiological Consequences of Control Rod Drop Accident (BWR)	Rev. 2 07/1981	Not applicable (BWR)	N/A
15.5.1-15.5.2 – Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	Rev. 2 03/2007	The APR1400 conforms with this SRP.	15.5.1, 15.5.2
15.6.1 – Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve	Rev. 2 03/2007	The APR1400 conforms with this SRP.	15.6.1
15.6.2 – Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	Rev. 2 07/1981	Not applicable. SRP 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors," is applied instead of SRP 15.6.2.	15.0.3
15.6.3 – Radiological Consequences of Steam Generator Tube Failure	Rev. 2 07/1981	Not applicable. SRP 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors," is applied instead of SRP 15.6.3.	15.0.3
15.6.4 – Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)	Rev. 2 07/1981	Not applicable (BWR)	N/A

Table 1.9-2 (31 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
15.6.5 – Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	Rev. 3 03/2007	The APR1400 conforms with this SRP	15.6.5
15.6.5.A – Radiological Consequences of a Design Basis Loss-of-Coolant Accident Including Containment Leakage Contribution	Rev. 1 07/1981	Not applicable. SRP 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors," is applied instead of SRP 15.6.5.A.	15.0.3
15.6.5.B – Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Engineered Safety Feature Components Outside Containment	Rev. 1 07/1981	Not applicable. SRP 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors," is applied instead of SRP 15.6.5.B.	15.0.3
15.6.5.D – Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Main Steam Isolation Valve Leakage Control System (BWR)	Rev. 1 07/1981	Not applicable (BWR)	N/A
15.7.3 – Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	Rev. 2 07/1981	Not applicable. SRP 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors," is applied instead of SRP 15.7.3.	15.0.3
15.7.4 – Radiological Consequences of Fuel Handling Accidents	Rev. 1 07/1981	Not applicable. SRP 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors," is applied instead of SRP 15.7.4.	15.0.3
15.7.5 – Spent Fuel Cask Drop Accidents	Rev. 2 07/1981	Not applicable. SRP 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors," is applied instead of SRP 15.7.5.	15.0.3

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SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
15.8 – Anticipated Transients Without Scram	Rev. 2 03/2007	The APR1400 conforms with this SRP.	15.8
15.9 – Boiling Water Reactor Stability	03/2007	Not applicable (BWR)	N/A
16.0 – Technical Specifications	Rev. 3 03/2010	The APR1400 conforms with this SRP.	16.1, 16.2, 16.3, 16.4, 16.5
16.1 – Risk-Informed Decision Making: Technical Specifications	Rev. 1 03/2007	N/A. The APR1400 does not apply Risk-Information Technical Specifications.	N/A
17.1 – Quality Assurance During the Design and Construction Phases	Rev. 2 07/1981	The APR1400 conforms with this SRP with the following exceptions: 17.1.8, 17.1.9, 17.1.13, and 17.1.14 are not applied in the DC phase.	17.1, 17.5
17.2 – Quality Assurance During the Operations Phase	Rev. 2 07/1981	Not applicable. The COL applicant is responsible for conforming with this SRP.	N/A
17.3 – Quality Assurance Program Description	08/1990	The APR1400 conforms with this SRP with exceptions. B-6, 7, 10, and 11 are not applied in DC phase.	17.3
17.4 – Reliability Assurance Program (RAP)	Rev.1 05/2014	The APR1400 conforms with this SRP.	17.4
17.5 – Quality Assurance Program Description-Design Certification, Early Site Permit and New License Applicants	03/2007	The APR1400 conforms with this SRP.	17.5
17.6 – Maintenance Rule	Rev. 1 08/2007	Not applicable (COL)	N/A

Table 1.9-2 (33 of 33)

SRP Section/Title	Revision / Issue Date	Conformance or Summary Description of Deviation	DCD Tier 2 Section
18.0 – Human Factors Engineering	Rev. 2 03/2007	The APR1400 conforms with this SRP.	Ch. 18
18-A –Guidance for Crediting Manual Operator Actions in Diversity and Defense-in-Depth (D3) Analyses	04/2014	The APR1400 conforms with this SRP.	18.6
19.0 – Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors	Rev. 2 06/2007	The APR1400 conforms with this SRP with exceptions. Note: SRP Acceptance Criteria for AP600 are out of the APR1400 scope.	19.1, 19.2
19.1 – Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests after Initial Fuel Load	Rev. 3 09/2012	The APR1400 conforms with this SRP.	19.1
19.2 – Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance	06/2007	Not applicable. This SRP section was written to address PRAs performed in support of changes proposed for existing, already-licensed plants.	N/A
19.4 – Strategies and Guidance to Address Loss-of-Large Areas of the Plant Due to Explosions and Fires	06/2014	The APR1400 conforms with this SRP.	19.4
19.5 – Adequacy of Design Features and Functional Capabilities Identified and Described for Withstanding Aircraft Impacts	04/2013	The APR1400 conforms with this SRP.	19.5

Table 1.9-3 (1 of 4)

APR1400 Conformance with Generic Issues (NUREG-0933)

Issue No.	Title	Discussion	DCD Tier 2 Section
89	Stiff Pipe Clamps	The stiff pipe clamps described in the generic issue, which are preloaded to prevent themselves from lifting of the piping under dynamic loading conditions, are not used for the APR1400 piping design.	N/A
186	Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants	For the APR1400, design for the containment polar and refueling cranes, spent fuel handling crane, and auxiliary building crane preclude the dropping of heavy loads. A critical load is defined in ASME NOG-1-2004 as any lifted load whose uncontrolled movement or release could adversely affect a nuclear safety-related (SC-1) SSC in terms of its ability to perform a required safety function, or when uncontrolled movement or release could result in potential offsite exposure in excess of 10 CFR limits. Cranes that may be used to handle critical loads over SC-I SSCs are classified as Type I cranes as defined in ASME NOG-1-2004 and conform with the applicable requirements of that standard as well as the Crane Manufacturers Association of America (CMAA) Specification No. 70-00. Type I cranes are designed to remain in place and support the critical load during and after, a seismic event, and are equipped with single failure-proof features in conformance with the requirements of ASME NOG-1-2004, to prevent load drops. The APR1400 cranes that do not handle critical loads over SC-I SSCs are not required to have single failure-proof features; however, any such cranes that may travel over SC-I SSCs are designed to remain in place during a seismic event.	9.1.5

Table 1.9-3 (2 of 4)

Issue No.	Title	Discussion	DCD Tier 2 Section
186	Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants (cont.)	Cranes that handle critical loads as well as non-critical loads conform with the applicable requirements of ASME NOG-1-2004 and CMAA Specification No. 70-00 or CMAA Specification No. 74-04 for their applicable lifts. Further, cranes are designed according to the crane structural standard and so structured as to prevent diversion and derailment. In addition, in the measures against earthquake, drop prevention design is employed based on earthquake design criteria. Therefore, load drops and derailment of cranes do not represent credible sources of missiles that would jeopardize safety-related SSCs, and load drop missiles are not postulated. The significance of crane operation and restricted load movement around the reactor vessel is stressed to those involved with heavy load lifts. Anticipated heavy load movements are analyzed as required by NUREG-0612 and safe load paths defined. However, all specific loads and load paths cannot be defined prior to the operations. For these cases, it is anticipated that safe load path considerations are based on comparison with analyzed cases, previously defined safe movement areas, and previously defined restricted areas and reviewed by the COL applicant's plant review board. Load handling procedures – Movements of heavy loads are controlled to protect safety-related SSCs. Load handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment will be controlled by written procedures.	

Table 1.9-3 (3 of 4)

Issue No.	Title	Discussion	DCD Tier 2 Section
186	Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants (cont.)	As a minimum, procedures are used for handling loads with spent fuel cask bridge crane and polar crane, and for the loads listed in Table 3-1 of NUREG-0612. It is anticipated that each procedure will address the following: • Equipment required to handle load (e.g., special lifting device, slings, shackles, turnbuckles, clevises, load cell) • Requirements for crane operator and rigger qualification • Requirements for inspection prior to load movement and acceptance criteria for inspection • Defined safe load path and provisions to provide visual reference to the crane operator and/or signal person of the safe load path envelope • Specific steps and proper sequence to be followed for handling load • Precautions, limitations, prerequisites, and/or initial conditions associated with movement of the load • Slings and other equipment used to make a complete lifting device specified in the load handling procedures, which conform with NUREG-0612 • Equipment layout drawings showing the safe load path, which are used to define safe load paths in load handling procedures; deviations from defined safe load paths require a written alternative procedure approved by the COL applicant's plant review board These considerations and commitments, as well as the other design and operational material presented above, are intended to prevent the types of events that are the subject of this generic issue. The APR1400 design is attentive to any new NRC operational guidance related to this issue.	

Table 1.9-3 (4 of 4)

Issue No.	Title	Discussion	DCD Tier 2 Section
189	Susceptibility of Ice Condenser Containments to Early Failure from Hydrogen Combustion During Severe Accident	Not applicable (applicable to ice condenser containment only)	N/A
191	Assessment of Debris Accumulation on PWR Sump Performance	The APR1400 is designed in accordance with NRC RG 1.82, Revision 4, the methodology of NEI 04-07, and the NRC's Safety Evaluation Report (SER) for NEI 04-07. Four redundant passive-type strainers are installed in the IRWST, which has a broad footprint for sufficient surface area. Insulation and coating debris is estimated by the NEI 04-07 methodology, and 200 pounds of latent debris is assumed to reach each strainer location. Trisodium phosphate (TSP) is selected as the agent for pH control in the recirculation water inside the holdup volume tank (HVT), to mitigate the chemical effect that might be caused during long-term cooling.	6.8.4.5,
193	BWR ECCS Suction Concerns	Not applicable (BWR)	N/A
199	Implications of Updated Probabilistic seismic Hazard Estimates in Central and Eastern United States	Not applicable (COL)	2.5

Table 1.9-4 (1 of 11)

APR1400 Conformance with Additional TMI-Related Requirements (10 CFR 50.34(f))

10 CFR 50.34(f) Item / Issue No.	Requirements	Conformance Discussion	DCD Tier 2 Section
(1)(i) / II.B.8	Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant.	The APR1400 conforms with this TMI-related requirement.	19.1
(1)(ii) / II.E.1.1	Perform an evaluation of the proposed auxiliary feedwater system (AFWS), to include (PWRs only): a. A simplified AFWS reliability analysis using event-tree and fault-tree logic techniques b. A design review of AFWS c. An evaluation of AFWS flow design bases and criteria	a. The APR1400 conforms with this TMI-related requirement b. Not applicable c. Not applicable	Table 15.0-11, 19.1 N/A N/A

Table 1.9-4 (2 of 11)

10 CFR 50.34(f) Item / Issue No.	Requirements	Conformance Discussion	DCD Tier 2 Section
(1)(iii) / II.K.2.16, II.K.3.25	Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss-of-coolant accident with subsequent reactor coolant pump seal damage.	RCP seal integrity can be maintained by either of two independent sources of cooling water: the seal injection flow from the chemical and volume control system (CVCS) or component cooling water (CCW). In the event of a loss of offsite AC power, the seal injection can be restored by aligning the emergency diesel generators (EDGs) power to the charging pumps or auxiliary charging pump (ACP). The CCW pumps restart in accordance with the EDG load sequencing to provide seal cooling. During a complete loss of AC power (loss of offsite power with loss of the EDGs), power can be supplied to the ACP from the onsite AAC power source to provide the RCP seal injection. Therefore, The APR1400 conforms with this TMI-related requirement.	9.2.2.2.4.5, Table 15.0-11
(1)(iv) / II.K.3.2	Perform an analysis of the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV). If this probability is a significant contributor to the probability of small-break LOCAs from all causes, provide a description and evaluation of the effect on small-break LOCA probability of an automatic PORV isolation system that would operate when the reactor coolant system pressure falls after the PORV has opened.	Not applicable. There is no PORV for the APR1400.	N/A

Table 1.9-4 (3 of 11)

10 CFR 50.34(f) Item / Issue No.	Requirements	Conformance Discussion	DCD Tier 2 Section
(2)(i) / I.A.4.2	Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCAs.	Not applicable (COL)	N/A
(2)(ii) / I.C.9	Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with Institute of Nuclear Power Operations (INPO) and other industry efforts.	Not applicable (COL)	N/A
(2)(iii) / I.D.1	Provide, for Commission review, a control room design that reflects state-of-the-art human factors principles prior to committing to fabrication or revision of fabricated control room panels and layouts.	The APR1400 conforms with this TMI-related requirement.	18.7
(2)(iv) / I.D.2	Provide a plant safety parameter display console that displays to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded.	The APR1400 conforms with this TMI-related requirement.	7.5.2.5 18.7
(2)(v) / I.D.3	Provide for automatic indication of the bypassed and operable status of safety systems.	The APR1400 conforms with this TMI-related requirement.	7.1.2.5, 7.5.2.3, 7.6.2.1, 8.3.1
(2)(vi) / II.B.1	Provide the capability of high-point venting of noncondensable gases from the RCS, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room, and their operation shall not lead to an unacceptable increase in the probability of LOCA or an unacceptable challenge to containment integrity.	The high-point vent system is installed to meet this requirement.	5.4.12.1

Table 1.9-4 (4 of 11)

10 CFR 50.34(f) Item / Issue No.	Requirements	Conformance Discussion	DCD Tier 2 Section
(2)(vii) / II.B.2	Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term11 radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment.	The APR1400 conforms with this TMI-related requirement.	12.4.1.2.7
(2)(viii) / II.B.3	Provide a capability to promptly obtain and analyze samples from the RCS and containment that may contain accident source term11 radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, radioiodines and cesiums, nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations.	The APR1400 has a post-accident sampling system to conform with this action item.	9.3.2.1, Table 15.0-11

Table 1.9-4 (5 of 11)

10 CFR 50.34(f) Item / Issue No.	Requirements	Conformance Discussion	DCD Tier 2 Section
(2)(ix) / II.B.8	Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (f)(1)(xii) of this section is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide, with reasonable assurance, that: a. Uniformly distributed hydrogen concentrations in the containment do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion. b. Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features. c. Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction including the environmental conditions created by activation of the hydrogen control system. d. If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.	The APR1400 conforms with this TMI-related requirement.	6.2.5.1

Table 1.9-4 (6 of 11)

10 CFR 50.34(f) Item / Issue No.	Requirements	Conformance Discussion	DCD Tier 2 Section
(2)(x) / II.D.1	Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWRs, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transients without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed.	Performance testing for the POSRV is performed on both normal and accident conditions, excluding anticipated transient without scram (ATWS), to provide the stable valve operation.	5.2.2.10
(2)(xi) / II.D.3	Provide direct indication of relief and safety valve position (open or closed) in the control room.	The APR1400 conforms with this TMI-related requirement.	5.2.2.1.1, 5.2.2.8, 7.1.2.6
(2)(xii) / II.E.1.2	Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide auxiliary feedwater system flow indication in the control room. (PWRs only)	The APR1400 conforms with this TMI-related requirement.	7.1.2.7, 7.2, Table 15.0-11
(2)(xiii) / II.E.3.1	Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available. (PWRs only)	The APR1400 conforms with this TMI-related requirement.	8.3.1.1.2
(2)(xiv) / II.E.4.2	Provide containment isolation systems that: a. Ensure all non-essential systems are isolated automatically by the containment isolation system b. For each non-essential penetration (except instrument) c. Do not result in reopening of the containment isolation d. Utilize a containment setpoint pressure for initiating containment isolation as low as is compatible with normal e. Include automatic closing on a high radiation signal for all systems that provide a path to the environs	The APR1400 conforms with this TMI-related requirement.	6.2.4.2, 7.1.2.8, 7.2, 7.5

Table 1.9-4 (7 of 11)

10 CFR 50.34(f) Item / Issue No.	Requirements	Conformance Discussion	DCD Tier 2 Section
(2)(xv) / II.E.4.4	Provide a capability for containment purging/venting designed to minimize the purging time consistent with as low as reasonably achievable (ALARA) principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions.	As described in NUREG-0933, under item II.E.4.4 (4), this item required the U.S. NRC to generically evaluate the radiological consequences of containment purging of nuclear plants while in the power operation mode. It was envisioned that, as a result of this evaluation, new requirements would be needed beyond those in SRP 6.2.4 and BTP 6-4. The NRC subsequently determined that this issue was a low-priority item; it was then resolved without issuance of new requirements. The valve operability guidance provided in SRP Section 6.2.4 and BTP 6-4, Rev.3, dated March 2007, was considered adequate by the U.S. NRC. The APR1400 conforms with SRP 6.2.4 and BTP 6-4 (Rev. 3, 03/2007).	9.4.6.2
(2)(xvii) / II.F.1	Provide instrumentation to measure, record, and readout in the control room (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples.	The APR1400 conforms with this TMI-related requirement.	7.1.2.9, 11.5.1.2, 12.3.4.1.5, Table 15.0-11

Table 1.9-4 (8 of 11)

10 CFR 50.34(f) Item / Issue No.	Requirements	Conformance Discussion	DCD Tier 2 Section
(2)(xviii) / II.F.2	Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWRs, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWRs and BWRs.	The APR1400 conforms with this TMI-related requirement based on NUREG-0737.	7.1.1.5, 7.1.2.10, 7.5.1.2, Table 15.0-11
(2)(xix) / II.F.3	Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage.	The APR1400 conforms with this TMI-related requirement based on NRC RG 1.97 Rev. 4.	7.1.2.11 Table 15.0-11
(2)(xx) / II.G.1	Provide power supplies for pressurizer relief valves, block valves, and level indicators such that (A) level indicators are powered from vital buses; (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety, and (C) electric power is provided from emergency power sources. (PWRs only)	The APR1400 conforms with this TMI-related requirement.	7.1.2.12, 7.4.2, 8.3.1
(2)(xxv) / III.A.1.2	Provide an onsite Technical Support Center, an onsite Operational Support Center, and for construction permit applications only, a near-site Emergency Operations Facility.	The APR1400 conforms with this TMI-related requirement.	9.5.2.1

Table 1.9-4 (9 of 11)

10 CFR 50.34(f) Item / Issue No.	Requirements	Conformance Discussion	DCD Tier 2 Section
(2)(xxvi) / III.D.1.1	Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency.	The APR1400 conforms with this TMI-related requirement.	9.3.3
(2)(xxvii) / III.D.3.3	Provide for monitoring of inplant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions.	The APR1400 conforms with this TMI-related requirement.	11.5.1.2, 12.3.4
(2)(xxviii) / III.D.3.4	Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in an accident source term release, and make necessary design provisions to preclude such problems.	The APR1400 conforms with this TMI-related requirement.	6.4.2.5, 15.6.5.5
(3)(i) / I.C.5	Provide administrative procedures for evaluating operating, design and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant.	The APR1400 conforms with this TMI-related requirement.	13.5
(3)(ii) / I.F.1	Ensure that the quality assurance (QA) list required by Criterion II, Appendix. B, 10 CFR 50 includes all structures, systems, and components important to safety.	Not applicable (COL)	N/A

Table 1.9-4 (10 of 11)

10 CFR 50.34(f) Item / Issue No.	Requirements	Conformance Discussion	DCD Tier 2 Section
(3)(iii) / I.F.2	Establish a quality assurance (QA) program based on consideration of (A) ensuring independence of the organization performing checking functions from the organization responsible for performing the functions, (B) performing quality assurance/quality control functions at construction sites to the maximum feasible extent, (C) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation, (D) establishing criteria for determining QA programmatic requirements, (E) establishing qualification requirements for QA and QC personnel, (F) sizing the QA staff commensurate with its duties and responsibilities, (G) establishing procedures for maintenance of “as-built” documentation, and (H) providing a QA role in design and analysis activities.	The APR1400 conforms with this TMI-related requirement.	Ch. 17
(3)(iv) / II.B.8	Provide one or more dedicated containment penetrations: Equivalent in size to a single 3-foot-diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system.	The APR1400 conforms with this TMI-related requirement.	19.2.3.3.8
(3)(vi) / II.E.4.1	For plant designs with external hydrogen recombiners: Provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere.	The dedicated containment penetration is not necessary because the APR1400 has applied the passive autocatalytic recombiners for hydrogen control, which are located inside the containment.	6.2.5

Table 1.9-4 (11 of 11)

10 CFR 50.34(f) Item / Issue No.	Requirements	Conformance Discussion	DCD Tier 2 Section
(3)(vii) / II.J.3.1	Provide a description of the management plan for design and construction activities, to include (A) the organizational and management structure singularly responsible for direction of design and construction of the proposed plant, (B) technical resources director by the applicant, (C) details of the interaction of design and construction within the applicant’s organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor, (D) proposed procedures for handling the transition to operation, (E) the degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort.	Not applicable (COL)	N/A

Table 1.9-5

Generic Communications Applicability to APR1400

GC No.	Title	Comment	DCD Tier 2 Section
GL 2008-01	Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems.	The APR1400 conforms with this Generic Letter.	6.2.2, 6.3.2.5.2
BL 2007-01	Security Officer Attentiveness	Not applicable (COL)	N/A
BL 2011-01	Mitigation Strategies	Not applicable	N/A
BL 2012-01	Design Vulnerability in Electric Power System	Not applicable (COL)	N/A

Table 1.9-6 (1 of 2)

Summary of SECY Documents Provided in Section C.I.1.9.5 of NRC RG 1.206

SECY Paper No.	Title	Discussion
89-013	Design Requirements Related to the Evolutionary Advanced Light-Water Reactors (ALWRs)	See Table 1.9-7 for SECY-93-087
90-016	Evolutionary Light-Water Reactor (ELWR) Certification Issues and Their Relationship to Current Regulatory Requirements.	See Table 1.9-7 for SECY-93-087
90-241	Level of Detail Required for Design Certification under 10 CFR 52.	The recommendations in this SECY were incorporated into 10 CFR Part 52. Conformance is addressed in Section 1.9 of this DCD Tier 2.
90-377	Requirements for Design Certification under 10 CFR 52.	The recommendations in this SECY were incorporated into 10 CFR Part 52. Conformance is addressed in Section 1.9 of this DCD Tier 2.
91-074	Prototype Decisions for Advanced Reactor Designs.	APR1400 does not contain "Significant deviation" from reference plants of standard technologies, and thus does not require prototypical demonstration.
91-178	ITAAC for Design Certifications and Combined Licenses.	The recommendations in this SECY were incorporated into 10 CFR Part 52. Conformance is addressed in Section 1.9.1 through 1.9.5 of this DCD Tier 2.
91-210	ITAAC Requirements for Design Review and Issuance of FDA.	The recommendations in this SECY were incorporated into 10 CFR Part 52. Conformance is addressed in Section 1.9 of this DCD Tier 2.
91-229	Severe Accident Mitigation Design Alternatives for Certified Standard Designs.	Severe accidents are addressed in Section 19 of this DCD Tier 2, and severe accident mitigation design alternatives (SAMDAs) are addressed in Subsection 19.2.6.

Table 1.9-6 (2 of 2)

SECY Paper No.	Title	Discussion
92-053	Use of Design Acceptance Criteria During the 10 CFR 52 Design Certification Reviews.	The recommendations in this SECY were incorporated into 10 CFR Part 52. Conformance is addressed in Subsection 1.9.1 through 1.9.5 of this DCD Tier 2.
92-092	The Containment Performance Goal, External Events Sequences, and the Definition of Containment Failure for Advanced LWRs.	This SECY does not impose any new requirements.
93-087	Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs.	See Table 1.9-7.
94-084	Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Design.	Not applicable
94-302	Source-Term-Related Technical and Licensing Issues Relating to Evolutionary and Passive Light-Water-Reactor Designs.	The APR1400 conforms with the positions presented in the SECY. the significant issues will be addressed in applicable to the APR1400 DCD Tier 2.
95-132	Policy and Technical Issues Associated with Regulatory Treatment of Non-Safety Systems in Passive Plant Designs.	Not applicable (AP1000 design only).

Table 1.9-7 (1 of 5)

Conformance with SECY-93-087

Item No.	Title	Discussion
I.A	Use of a Physically Based Source Term	Addressed for the APR1400 in DCD Tier 2 Subsection 3.11.6 and Appendix 15A.
I.B	Anticipated Transient Without Scram	Addressed for the APR1400 in DCD Tier 2 Section 15.8.
I.C	Mid-Loop Operation	Addressed for the APR1400 in DCD Tier 2 Subsection 5.4.7.2.6 and 19.2.2.2.
I.D	Station Blackout	Addressed for the APR1400 in DCD Tier 2 Section 8.4.
I.E	Fire Protection	Addressed for the APR1400 in DCD Tier 2 Subsections 3.1.1 and 9.5.1.
I.F	Intersystem Loss-of-Coolant Accident	Addressed for the APR1400 in DCD Tier 2 Subsections 5.2.5.4 and 19.2.2.5.
I.G	Hydrogen Control	Addressed for the APR1400 in DCD Tier 2 Subsections 6.2.5 and 19.2.3.3.2. Non-safety related HMS consisting of igniters and passive autocatalytic recombiners are located in containment adequately.
I.H	Core Debris Coolability	Addressed for the APR1400 in DCD Tier 2 Subsection 19.2.3.3.3. Core coolability is confirmed using MAAP code.
I.I	High-Pressure Core Melt Ejection	Addressed for the APR1400 in DCD Tier 2 Subsection 19.2.3.3.4.

Table 1.9-7 (2 of 5)

Item No.	Title	Discussion
I.J	Containment Performance	Designed robustly the APR1400 containment to withstand containment pressure challenges.
I.K	Dedicated Containment Vent Penetration	Dedicated containment vent is not provided.
I.L	Equipment Survivability	Addressed for the APR1400 in DCD Tier 2, Section 19.2.
I.M	Elimination of Operating-Basis Earthquake	Addressed for the APR1400 in DCD Tier 2, Subsection 3.2.6.
I.N	Inservice Testing of Pumps and Valves	Addressed for the APR1400 in DCD Tier 2, Subsections 3.1.4 and 3.9.6, Sections 6.6 and 13.4, and Chapter 16.
II.A	Industry Codes and Standards	Addressed for the APR1400 in DCD Tier 2, Subsection 3.2.8.
II.B	Electrical Distribution	The APR1400 conforms with the requirement of SECY 91-078.
II.C	Seismic Hazard Curves and Design Parameters	Not applicable (information only)
II.D	Leak-Before-Break	Addressed for the APR1400 in DCD Tier 2, Subsection 3.6.3.
II.E	Classification of Main Steamlines in Boiling Water Reactors	Not applicable (BWR)
II.F	Tornado Design Basis	The design basis tornado with a maximum wind speed of 230 mph is employed in accordance with Revision 1 of NRC RG 1.76.

Table 1.9-7 (3 of 5)

Item No.	Title	Discussion
II.G	Containment Bypass	Addressed for the APR1400 in DCD Tier 2, Subsection 19.2.2.5. The principal contributors of containment bypass are steam generator tube ruptures (SGTRs) with MSSVs or ADVs and interfacing-system LOCAs (ISLOCAs). The APR1400 is designed to prevent and mitigate the following accidents: • The APR1400 has performed the Interfacing System LOCA (ISLOCA) evaluation to address the issue of containment bypass. Improvements made to the APR1400 resulting from the ISLOCA evaluation include: • Increasing the design pressure rating of equipment or systems to at least 900 psig. • Adding equipment and instrumentation that alert the operator to an ISLOCA challenge, or terminate and limit the scope of an ISLOCA. • Improvement the capability for leak testing pressure isolation valves. • Pressure isolation valve position indication and control in the control room. • High pressure alarms to warn the operator when rising pressure approaches the design pressure of low-pressure systems. All the improvements made to the APR1400 as a result of the ISLOCA evaluation are addressed in this DCD, Tier 2, Subsection 19.2.2.5.

Table 1.9-7 (4 of 5)

Item No.	Title	Discussion
II.H	Containment Leak Rate Testing	The maximum interval between Type C leakage rate tests, which is stated in the policy, is not addressed in the APR1400 DCD Tier 2. This policy is closely related to plant operation, so the maximum Type C test interval will be considered in the course of developing plant operator's containment leak rate testing program.
II.I	Post-Accident Sampling System	Conformance is described in Subsection 9.3.2.
II.J	Level of Detail	The APR1400 has the level of detail of information required to acquire design certification.
II.K	Prototyping	Not applicable (information only)
II.L	ITAAC	Development guidance for ITAAC is addressed in Section 14.3 and ITAAC for each system is described in Tier 1 of this DCD.
II.M	Reliability Assurance Program	The APR1400 reliability assurance program, addressing the requirements appropriate for design certification is presented in the DCD Tier 2, Section 17.4.
II.N	Site-Specific Probabilistic Risk Assessments and Analysis of External Events	Addressed for the APR1400 in DCD Tier 2, Section 19.1. PRA covers seismic events, internal fire events, and internal flooding events as well as internal events. The COL applicant is to perform site-specific PRA evaluations to address any site-specific hazards.
II.O	Severe Accident Mitigation Design Alternatives	Addressed for the APR1400 in DCD Tier 2, Subsection 19.2.6.
II.P	Generic Rulemaking Related to Design Certification	Not applicable (information only)

Table 1.9-7 (5 of 5)

Item No.	Title	Discussion
II.Q	Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems	Addressed for the APR1400 in DCD Tier 2, Subsection 7.1.2.36, Table 7.1-1, and Subsections 7.3.2.4, 7.8.2.1, 7.8.2.2, and 7.8.2.3.
II.R	Steam Generator Tube Ruptures	Addressed for the APR1400 in DCD Tier 2, Subsection 15.6.3.
II.S	PRA Beyond Design Certification	Not applicable (COL)
II.T	Control Room Annunciator (Alarm) Reliability	Addressed for the APR1400 in DCD Tier 2, Subsection 7.1.2.37, Table 7.1-1, and Subsection 7.5.2.4.
III.E	Control Room Habitability	Not applicable
III.F	Radionuclide Attenuation	Not applicable

Table 1.9-8 (1 of 19)

APR1400 Strategies for Addressing Tier 1, 2 and 3 NTTF Recommendations

NTTF Rec. No	NRC Recommendations/Requirements in SECY-11-0093, SECY-11-0137, SECY-12-0025, SECY-12-0095, EA-12-049, EA-12-051	APR1400 Design	Applicable DCD Section	COL Action	Note
Tier 1 (Actions to be taken without delay)					
2.1	Seismic Reevaluation a) Evaluate the potential impacts of the newly released Central and Eastern United States Seismic Source Characterization (CEUS-SSC) model, with potential local and regional refinements as identified in the CEUS-SSC model, on the seismic hazard curves and the site-specific ground motion response spectra (GMRS)/foundation input response spectra (FIRS). For re-calculation of the probabilistic seismic hazard analysis (PSHA), please follow either the cumulative absolute velocity (CAV) filter or minimum magnitude specifications outlined in Attachment 1 to Seismic Enclosure 1 of the March 12, 2012 letter "Request for information pursuant to Title 10 of the Code of Federal Regulations 50.54(f) regarding Recommendations 2.1, 2.3, and 9.3, of the near-term task force review of insights from the Fukushima Dai-ichi accident." (ML12053A340). b) In your response, please identify the method you selected from the above choices to perform the evaluation. Modify and submit the site-specific GMRS and FIRS changes, as necessary, given the evaluation performed in part (a) above. Provide the basis supporting your position.	NA	NA	COL 19.3(1)	Request for information via 50.54 (f) letter.

Table 1.9-8 (2 of 19)

NTTF Rec. No	NRC Recommendations/Requirements in SECY-11-0093, SECY-11-0137, SECY-12-0025, SECY-12-0095, EA-12-049, EA-12-051	APR1400 Design	Applicable DCD Section	COL Action	Note
2.1	Flooding Reevaluation • Perform a reevaluation of all appropriate external flooding sources, including the effects from local intense precipitation on the site, probable maximum flood (PMF) on stream and rivers, storm surges, seiches, tsunami, and dam failures. It is requested that the reevaluation apply present-day regulatory guidance and methodologies being used for ESP and COL reviews including current techniques, software, and methods used in present-day standard engineering practice to develop the flood hazard.	NA	NA	COL 19.3(2)	Request for information via 50.54 (f) letter.
2.3	Seismic Walkdowns • Perform seismic walkdowns in order to identify and address plant specific degraded, non-conforming, or unanalyzed conditions and verify the adequacy of strategies, monitoring, and maintenance programs such that the nuclear power plant can respond to external events. The walkdown will verify current plant configuration with the current licensing basis, verify the adequacy of current strategies, maintenance plans, and identify degraded, non-conforming, or unanalyzed conditions.	NA	NA	NA	Request for information via 50.54 (f) letter.

Table 1.9-8 (3 of 19)

NTTF Rec. No	NRC Recommendations/Requirements in SECY-11-0093, SECY-11-0137, SECY-12-0025, SECY-12-0095, EA-12-049, EA-12-051	APR1400 Design	Applicable DCD Section	COL Action	Note
2.3	Flooding Walkdowns • Perform flood protection walkdowns using an NRC-endorsed walkdown methodology, • Identify and address plant-specific degraded, non-conforming, or unanalyzed conditions as well as cliff-edge effects through the corrective action program and consider these findings in the Recommendation 2.1 hazard evaluations, as appropriate, • Identify any other actions taken or planned to further enhance the site flood protection, • Verify the adequacy of programs, monitoring and maintenance for protection features, and, • Report to the NRC the results of the walkdowns and corrective actions taken or planned.	NA	NA	NA	Request for information via 50.54 (f) letter.
4.1	Station Blackout (SBO) (NTTF Recommendations) Initiate rulemaking to revise 10 CFR 50.63 to require each operating and new reactor licensee to (1) establish a minimum coping time of 8 hours for a loss of all ac power, (2) establish the equipment, procedures, and training necessary to implement an “extended loss of all ac” coping time of 72 hours for core and spent fuel pool cooling and for reactor coolant system and primary containment integrity as needed, and (3) preplan and prestage offsite resources to support uninterrupted core and spent fuel pool cooling, and reactor coolant system and containment integrity as needed, including the ability to deliver the equipment to the site in the time period allowed for extended coping, under conditions involving significant degradation of offsite transportation infrastructure associated with significant natural disasters.	See Technical Report APR1400-E-P-NR-14005-P, Rev 0; Section 5.1.2	See DCD Section 19.3.2.3	COL 19.3(3), 19.3(4) and 19.3(5)	

Table 1.9-8 (4 of 19)

NTTF Rec. No	NRC Recommendations/Requirements in SECY-11-0093, SECY-11-0137, SECY-12-0025, SECY-12-0095, EA-12-049, EA-12-051	APR1400 Design	Applicable DCD Section	COL Action	Note
4.2	Mitigation Strategies for Beyond-Design-Basis External Events (EA-12-049) 1. Licensees shall develop, implement and maintain guidance and strategies to maintain or restore core cooling, containment and SFP cooling capabilities following a beyond-design-basis external event. 2. These strategies must be capable of mitigating a simultaneous loss of all alternating current (ac) power and loss of normal access to the ultimate heat sink and have adequate capacity to address challenges to core cooling, containment, and SFP cooling capabilities at all units on a site subject to this Order. 3. Licensee must provide reasonable protection for the associated equipment from external events. Such protection must demonstrate that there is adequate capacity to address challenges to core cooling, containment, and SFP cooling capabilities at all units on a site subject to this order. 4. Licensee must be capable of implementing the strategies in all modes. 5. Full compliance shall include procedures, guidance, training, and acquisition, staging, or installation of equipment needed for the strategies.	See Technical Report APR1400-E-P-NR-14005-P, Rev 0; Section 5.1.2	See DCD Section 19.3.2.3	COL 19.3(3), 19.3(4) and 19.3(5)	

Table 1.9-8 (5 of 19)

NTTF Rec. No	NRC Recommendations/Requirements in SECY-11-0093, SECY-11-0137, SECY-12-0025, SECY-12-0095, EA-12-049, EA-12-051	APR1400 Design	Applicable DCD Section	COL Action	Note
5.1	Reliable Hardened Vents for Mark I and Mark II containments Boiling-Water Reactor (BWR) Mark I and Mark II containments shall have a reliable hardened vent to remove decay heat and maintain control of containment pressure within acceptable limits following events that result in the loss of active containment heat removal capability or prolonged Station Blackout (SBO). The hardened vent system shall be accessible and operable under a range of plant conditions, including a prolonged SBO and inadequate containment cooling.	NA	NA	NA	
7.1	SFP Instrumentation (EA-12-051 to COL Holder) Licensee requires reliable indication of the water level in associate spent fuel storage capable of supporting identification of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred. 1. The spent fuel pool level instrumentation shall include the following design features:	See Technical Report APR1400-E-P-NR-14005-P, Rev 0; Section 5.1.3	See DCD Section 19.3.2.4	COL 19.3(6)	

Table 1.9-8 (6 of 19)

NTTF Rec. No	NRC Recommendations/Requirements in SECY-11-0093, SECY-11-0137, SECY-12-0025, SECY-12-0095, EA-12-049, EA-12-051	APR1400 Design	Applicable DCD Section	COL Action	Note
7.1 (cont.)	1.1 Arrangement: The spent fuel pool level instrument channels shall be arranged in a manner that provides reasonable protection of the level indication function against missiles that may result from damage to the structure over the spent fuel pool. This protection may be provided by locating the safety-related instruments to maintain instrument channel separation within the spent fuel pool area, and to utilize inherent shielding from missiles provided by existing recesses and corners in the spent fuel pool structure. 1.2 Qualification: The level instrument channels shall be reliable at temperature, humidity, and radiation levels consistent with the spent fuel pool water at saturation conditions for an extended period. 1.3 Power supplies: Instrumentation channels shall provide for power connections from sources independent of the plant alternating current (ac) and direct current (dc) power distribution systems, such as mobile generators or replaceable batteries. Power supply designs should provide for quick and accessible connection of sources independent of the plant ac and dc power distribution systems. On-site generators used as an alternate power source and replaceable batteries used for instrument channel power shall have sufficient capacity to maintain the level indication function until offsite resource availability is reasonably assured.				

Table 1.9-8 (7 of 19)

NTTF Rec. No	NRC Recommendations/Requirements in SECY-11-0093, SECY-11-0137, SECY-12-0025, SECY-12-0095, EA-12-049, EA-12-051	APR1400 Design	Applicable DCD Section	COL Action	Note
7.1 (cont.)	1.4 Accuracy: The instrument shall maintain its designed accuracy following a power interruption or change in power source without recalibration. 1.5 Display: The display shall provide on-demand or continuous indication of spent fuel pool water level.				
	2. The spent fuel pool instrumentation shall be maintained available and reliable through appropriate development and implementation of a training program. Personnel shall be trained in the use and the provision of alternate power to the safety-related level instrument channels.				
8	Strengthening and integration of emergency operating procedures, severe accident management guidelines (SAMGs), and extensive damage mitigation guidelines (NTTF Recommendations) 1. Order licensees to modify the EOP technical guidelines (required by Supplement 1, “Requirements for Emergency Response Capability,” to NUREG-0737, issued January 1983 (GL 82-33), to (1) include EOPs, SAMGs, and EDMGs in an integrated manner, (2) specify clear command and control strategies for their implementation, and (3) stipulate appropriate qualification and training for those who make decisions during emergencies. 2. Modify Section 5.0, “Administrative Controls,” of the Standard Technical Specifications for each operating reactor design to reference the approved EOP technical guidelines for that plant design.	NA	NA	COL 19.3(7)	

Table 1.9-8 (8 of 19)

NTTF Rec. No	NRC Recommendations/Requirements in SECY-11-0093, SECY-11-0137, SECY-12-0025, SECY-12-0095, EA-12-049, EA-12-051	APR1400 Design	Applicable DCD Section	COL Action	Note
8 (cont.)	3. Order licensees to modify each plant's technical specifications to conform to the above changes. 4. Initiate rulemaking to require more realistic, hands-on training and exercises on SAMGs and EDMGs for all staff expected to implement the strategies and those licensee staff expected to make decisions during emergencies, including emergency coordinators and emergency directors.				
9.3	Emergency Preparedness (SECY-12-0025, DCD RAI 644-6516) Communications 1. Provide an assessment of the current communications systems and equipment used during an emergency event to identify any enhancements that may be needed to ensure communications are maintained during a large scale natural event meeting the conditions described above. The assessment should: • Identify any planned or potential improvements to existing on-site communications systems and their required normal and/or backup power supplies, • Identify any planned or potential improvements to existing offsite communications systems and their required normal and/or backup power supplies, • Provide a description of any new communications system(s) or technologies that will be deployed based upon the assumed conditions described above, and • Provide a description of how the new and/or improved systems and power supplies will be able to provide for communications during a loss of all ac power,	NA	NA	COL 19.3(8)	

Table 1.9-8 (9 of 19)

NTTF Rec. No	NRC Recommendations/Requirements in SECY-11-0093, SECY-11-0137, SECY-12-0025, SECY-12-0095, EA-12-049, EA-12-051	APR1400 Design	Applicable DCD Section	COL Action	Note
9.3 (cont.)	2. Describe any interim actions that have been taken or are planned to be taken to enhance existing communications systems power supplies until the communications assessment and the resulting actions are complete, 3. Provide an implementation schedule of the time needed to conduct and implement the results of the communications assessment.				
	Staffing 1. Provide an assessment of the on-site and augmented staff needed to respond to a large scale natural event meeting the conditions described above. This assessment should include a discussion of the on-site and augmented staff available to implement the strategies as discussed in the emergency plan and/or described in plant operating procedures. The following functions are requested to be assessed: • How on-site staff will move back-up equipment (e.g., pumps, generators) from alternate on-site storage facilities to repair locations at each reactor as described in the order regarding the NTTF Recommendation 4.2. It is requested that consideration be given to the major functional areas of NUREG-0654, Table B¬1 such as plant operations and assessment of operational aspects, emergency direction and control, notification/ communication, radiological accident assessment, and support of operational accident assessment, as appropriate. • New staff or functions identified as a result of the assessment.		NA	COL 19.3(9)	

Table 1.9-8 (10 of 19)

NTTF Rec. No	NRC Recommendations/Requirements in SECY-11-0093, SECY-11-0137, SECY-12-0025, SECY-12-0095, EA-12-049, EA-12-051	APR1400 Design	Applicable DCD Section	COL Action	Note
9.3 (cont.)	• Collateral duties (personnel not being prevented from timely performance of their assigned functions). 2. Provide an implementation schedule of the time needed to conduct the on-site and augmented staffing assessment. If any modifications are determined to be appropriate, please include in the schedule the time to implement the changes. 3. Identify how the augmented staff would be notified given degraded communications capabilities. 4. Identify the methods of access (e.g., roadways, navigable bodies of water and dockage, airlift, etc.) to the site that are expected to be available after a widespread large scale natural event. 5. Identify any interim actions that have been taken or are planned prior to the completion of the staffing assessment. 6. Identify changes that have been made or will be made to your emergency plan regarding the on-shift or augmented staffing changes necessary to respond to a loss of all ac power, multi-unit event, including any new or revised agreements with offsite resource providers (e.g., staffing, equipment, transportation, etc.).	NA			

Table 1.9-8 (11 of 19)

NTTF Rec. No	NRC Recommendations/Requirements in SECY-11-0093, SECY-11-0137, SECY-12-0025, SECY-12-0095, EA-12-049, EA-12-051	APR1400 Design	Applicable DCD Section	COL Action	Note
	Filtration of Containment Vents The staff is considering requiring the filtration of containment vents to reduce the spread of radioactive contamination during a beyond-design-basis event. The staff plans to provide the Commission a notation vote paper on these policy issues in July 2012. At this time, the staff is proposing regulatory action to require that all operating BWR facilities with Mark I and Mark II containments have a reliable hardened venting capability, without filters, for events that can lead to core damage.	NA	NA	NA	
-	Loss of Ultimate Heat Sink (SECY-12-0025) 1. Include UHS systems in the reevaluation and walkdowns of site-specific seismic and flooding hazards using the methodology described in SECY-11-0137, and identify actions that have been taken, or are planned, to address plant-specific issues associated with the updated seismic and flooding hazards in conjunction with the resolution of NTTF Recommendations 2.1 and 2.3.	NA	NA	COL 19.3(1) and 19.3(2)	
	2. Incorporate the loss of UHS as a design assumption in the resolution of station blackout rulemaking activities in conjunction with the resolution of NTTF Recommendation 4.1.	See Technical Report APR1400-E-P-NR-14005-P, Rev 0; Section 5.1.2	See DCD Section 19.3.2.3	COL 19.3(3), 19.3(4) and 19.3(5)	
	3. Provide mitigating measures for beyond-design-basis external events to also include a loss of access to the normal UHS in conjunction with the resolution of NTTF Recommendation 4.2.	See Technical Report APR1400-E-P-NR-14005-P, Rev 0; Section 5.1.2	See DCD Section 19.3.2.3	COL 19.3(3), 19.3(4) and 19.3(5)	

Table 1.9-8 (12 of 19)

NTTF Rec. No	NRC Recommendations/Requirements in SECY-11-0093, SECY-11-0137, SECY-12-0025, SECY-12-0095, EA-12-049, EA-12-051	APR1400 Design	Applicable DCD Section	COL Action	Note
	4. Include UHS systems in the reevaluation of site-specific natural external hazards, and identify actions that have been taken, or are planned, to address plant-specific issues associated with the updated hazards in conjunction with the resolution of the new Tier 2 Recommendation 2.1 activity described in Enclosure 3, "Other Natural External Hazards."	NA	NA	Refer to Tier 2 Recommendation	
Tier 2 (Actions do not require long-term study and can be initiated when sufficient technical information and applicable resources become available)					
2.1	Other External Events Protections (SECY-12-0025) 1. Continue stakeholder interactions to discuss the technical basis and acceptance criteria for conducting a reevaluation of site-specific external natural hazards. These interactions will also help to define guidelines for the application of current regulatory guidance and methodologies being used for early site permit and combined license reviews to the reevaluation of hazards at operating reactors. 2. Develop and issue a request for information to licensees pursuant to 10 CFR 50.54(f) to (1) reevaluate site-specific external natural hazards using the methodology discussed in Item 1 above, and (2) identify actions that have been taken, or are planned, to address plant-specific issues associated with the updated natural external hazards (including potential changes to the licensing or design basis of a plant). 3. Evaluate licensee responses and take appropriate regulatory action to resolve issues associated with updated site-specific natural external hazards.	No Action (See Technical Report APR1400- E-P-NR-14005-P, Rev 0)	NA	NA	

Table 1.9-8 (13 of 19)

NTTF Rec. No	NRC Recommendations/Requirements in SECY-11-0093, SECY-11-0137, SECY-12-0025, SECY-12-0095, EA-12-049, EA-12-051	APR1400 Design	Applicable DCD Section	COL Action	Note
7	SFP Makeup Capability (NTTF 7.2, 7.3, 7.4, and 7.5) (NTTF Recommendations) 7.2 Order licensees to provide safety-related ac electrical power for the spent fuel pool makeup system.	No Action (See Technical Report APR1400- E-P-NR-14005-P, Rev 0)	NA	NA	
	7.3 Order licensees to revise their technical specifications to address requirements to have one train of on-site emergency electrical power operable for spent fuel pool makeup and spent fuel pool instrumentation when there is irradiated fuel in the spent fuel pool, regardless of the operational mode of the reactor.	No Action (See Technical Report APR1400- E-P-NR-14005-P, Rev 0)	NA	NA	
	7.4 Order licensees to have an installed seismically qualified means to spray water into the spent fuel pools, including an easily accessible connection to supply the water (e.g., using a portable pump or pumper truck) at grade outside the building.	No Action (See Technical Report APR1400- E-P-NR-14005-P, Rev 0)	NA	NA	
	7.5 Initiate rulemaking or licensing activities or both to require the actions related to the spent fuel pool described in detailed recommendations 7.1–7.4.	No Action (See Technical Report APR1400- E-P-NR-14005-P, Rev 0)	NA	NA	
9.3	Emergency preparedness regulatory actions (the remaining portions of Recommendation 9.3, with the exception of Emergency Response Data System (ERDS) capability addressed in Tier 3) 1. Engage stakeholders to inform the development of acceptance criteria for the licensee examination of planning standard elements related to the recommendations, and	No Action	NA	NA	

Table 1.9-8 (14 of 19)

NTTF Rec. No	NRC Recommendations/Requirements in SECY-11-0093, SECY-11-0137, SECY-12-0025, SECY-12-0095, EA-12-049, EA-12-051	APR1400 Design	Applicable DCD Section	COL Action	Note
9.3 (cont.)	2. Develop and issue an order to address those changes necessary in emergency plans to ensure adequate response to SBO and multiunit events specific to (1) adding guidance to the emergency plan that documents how to perform a multiunit dose assessment, (2) conduct periodic training and exercises for multiunit and prolonged SBO scenarios, (3) practice (simulate) the identification and acquisition of offsite resources, to the extent possible, and (4) ensure that EP equipment and facilities are sufficient for dealing with multiunit and prolonged SBO scenarios.				
Tier 3 (Those NTTF Recommendations that require further staff study to support a regulatory action)					
2.2	Ten-year confirmation of seismic and flooding hazards (dependent on Recommendation 2.1) Initiate rulemaking to require licensees to confirm seismic hazards and flooding hazards every 10 years and address any new and significant information. If necessary, update the design basis for SSCs important to safety to protect against the updated hazards.	No Action – See Technical Report APR1400-E-P-NR-14005-P, Rev 0	NA	NA	
3	Potential enhancements to the capability to prevent or mitigate seismically-induced fires and floods (long-term evaluation) The Task Force recommends, as part of the longer term review, that the NRC evaluate potential enhancements to the capability to prevent or mitigate seismically induced fires and floods.	No Action – See Technical Report APR1400-E-P-NR-14005-P, Rev 0	NA	NA	

Table 1.9-8 (15 of 19)

NTTF Rec. No	NRC Recommendations/Requirements in SECY-11-0093, SECY-11-0137, SECY-12-0025, SECY-12-0095, EA-12-049, EA-12-051	APR1400 Design	Applicable DCD Section	COL Action	Note
5.2	Reliable hardened vents for other containment designs (long-term evaluation) Reevaluate the need for hardened vents for other containment designs, considering the insights from the Fukushima accident. Depending on the outcome of the reevaluation, appropriate regulatory action should be taken for any containment designs requiring hardened vents.	No Action – See Technical Report APR1400-E-P-NR-14005-P, Rev 0	NA	NA	
6	Hydrogen control and mitigation inside containment or in other buildings (long-term evaluation) The Task Force recommends, as part of the longer term review, that the NRC identify insights about hydrogen control and mitigation inside containment or in other buildings as additional information is revealed through further study of the Fukushima Dai-ichi accident.	No Action – See Technical Report APR1400-E-P-NR-14005-P, Rev 0	NA	NA	
9.1 & 9.2	Emergency preparedness (EP) enhancements for prolonged SBO and multiunit events (dependent on availability of critical skill sets) 9.1 Initiate rulemaking to require EP enhancements for multiunit events in the following areas: • personnel and staffing • dose assessment capability • training and exercises • equipment and facilities	No Action – See Technical Report APR1400-E-P-NR-14005-P, Rev 0	NA	NA	

Table 1.9-8 (16 of 19)

NTTF Rec. No	NRC Recommendations/Requirements in SECY-11-0093, SECY-11-0137, SECY-12-0025, SECY-12-0095, EA-12-049, EA-12-051	APR1400 Design	Applicable DCD Section	COL Action	Note
9.1 & 9.2 (cont.)	9.1 Initiate rulemaking to require EP enhancements for prolonged SBO in the following areas: • communications capability • ERDS capability • training and exercises • equipment and facilities				
9.3	ERDS capability (related to long-term evaluation Recommendation 10) Order licensees to do the following until rulemaking is complete: • Maintain ERDS capability throughout the accident.	No Action – See Technical Report APR1400-E-P-NR-14005-P, Rev 0	NA	NA	
10	Additional EP topics for prolonged SBO and multiunit events (long-term evaluation) 10.1 Analyze current protective equipment requirements for emergency responders and guidance based upon insights from the accident at Fukushima.	No Action – See Technical Report APR1400-E-P-NR-14005-P, Rev 0	NA	NA	

Table 1.9-8 (17 of 19)

NTTF Rec. No	NRC Recommendations/Requirements in SECY-11-0093, SECY-11-0137, SECY-12-0025, SECY-12-0095, EA-12-049, EA-12-051	APR1400 Design	Applicable DCD Section	COL Action	Note
10 (cont.)	10.2 Evaluate the command and control structure and the qualifications of decision-makers to ensure that the proper level of authority and oversight exists in the correct facility for a long-term SBO or multiunit accident or both. • Concepts such as whether decision-making authority is in the correct location (i.e., at the facility), whether currently licensed operators need to be integral to the ERO outside of the control room (i.e., in the TSC), and whether licensee emergency directors should have a formal "license" qualification for severe accident management. 10.3 Evaluate ERDS to do the following: • Determine an alternate method (e.g., via satellite) to transmit ERDS data that does not rely on hardwired infrastructure that could be unavailable during a severe natural disaster. • Determine whether the data set currently being received from each site is sufficient for modern assessment needs. • Determine whether ERDS should be required to transmit continuously so that no operator action is needed during an emergency.				

Table 1.9-8 (18 of 19)

NTTF Rec. No	NRC Recommendations/Requirements in SECY-11-0093, SECY-11-0137, SECY-12-0025, SECY-12-0095, EA-12-049, EA-12-051	APR1400 Design	Applicable DCD Section	COL Action	Note
11	EP topics for decision-making, radiation monitoring, and public education (long-term evaluation) 11.1 Study whether enhanced on-site emergency response resources are necessary to support the effective implementation of the licensees' emergency plans, including the ability to deliver the equipment to the site under conditions involving significant natural events where degradation of offsite infrastructure or competing priorities for response resources could delay or prevent the arrival of offsite aid. 11.2 Work with FEMA, States, and other external stakeholders to evaluate insights from the implementation of EP at Fukushima to identify potential enhancements to the U.S. decision-making framework, including the concepts of recovery and reentry. 11.3 Study the efficacy of real-time radiation monitoring on-site and within the EPZs (including consideration of ac independence and real-time availability on the Internet). 11.4 Conduct training, in coordination with the appropriate Federal partners, on radiation, radiation safety, and the appropriate use of KI in the local community around each nuclear power plant.	No Action	NA	NA	

Table 1.9-8 (19 of 19)

NTTF Rec. No	NRC Recommendations/Requirements in SECY-11-0093, SECY-11-0137, SECY-12-0025, SECY-12-0095, EA-12-049, EA-12-051	APR1400 Design	Applicable DCD Section	COL Action	Note
12.1	Reactor Oversight Process modifications to reflect the recommended defense-in-depth framework (dependent on Recommendation 1) Expand the scope of the annual reactor oversight process (ROP) self-assessment and biennial ROP realignment to more fully include defense-in-depth considerations.	No Action	NA	NA	
12.2	Staff Training on Severe Accidents and Resident Inspector Training on SAMGs (dependent on Recommendation 8) Enhance NRC staff training on severe accidents, including training resident inspectors on SAMGs.	No Action	NA	NA	