DESIGN CONTROL DOCUMENT FOR THE US-APWR

Chapter 1
Introduction and General Description of Plant

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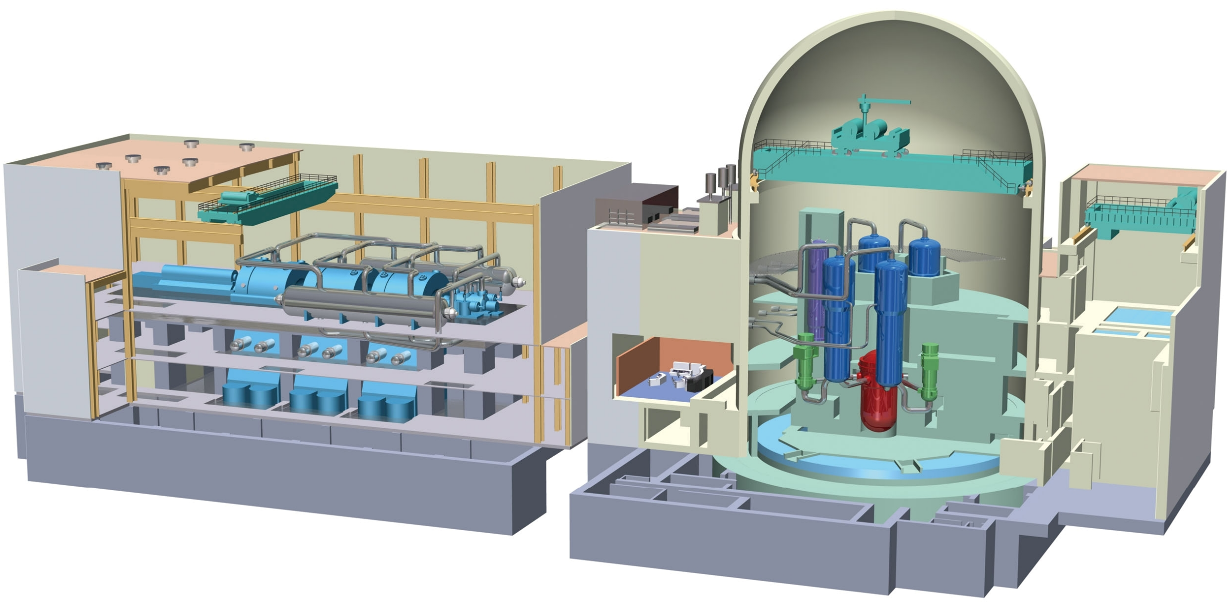

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CONTENTS

CONTENTS (Continued)

CONTENTS (Continued)

TABLES

TABLES (Continued)

TABLES (Continued)

FIGURES

ACRONYMS AND ABBREVIATIONS

A/B	auxiliary building
AAC	alternate alternating current
AAS	automatic actuation system
ABDP	A/B equipment drain sump pump
ABVS	auxiliary building ventilation system
ac	alternating current
AC/B	access building
AC/PS	ac power system
ACC	accumulator
ACCW	auxiliary component cooling water
ACCWS	auxiliary component cooling water system
ACI	American Concrete Institute
ACL	accident class
ACNSPDS	ac non safety power distribution system
ADS	automatic depressurization system
AECS	auxiliary equipment control system
AEES	annulus emergency exhaust system
AFC	automatic frequency control
AFD	axial flux difference
Ag-In-Cd	silver-indium-cadmium
AHU	air handling unit
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
ALARA	as low as reasonably achievable
ALHR	average linear heat rate
ALR	automatic load regulator
ALWR	advanced light-water reactor
AMI	audio messenger interface
AMSAC	ATWS mitigation system actuation circuitry
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
AO	axial offset
AOO	anticipated operational occurrence
AOP	abnormal operating procedure
API	American Petroleum Institute
APWR	advanced pressurized water reactor

ACRONYMS AND ABBREVIATIONS (Continued)

ARMS	area radiation monitoring system
ARO	all rods out
ARS	acceleration response spectra
ASCE	American Society of Civil Engineers
ASD	Allowable Stress Design
ASEP	accident sequence evaluation program
ASME	American Society of Mechanical Engineers
ASSS	auxiliary steam supply system
AST	alternative source term
ASTM	American Society for Testing and Materials
ATC	automatic turbine control
ATWS	anticipated transient without scram
AVR	auto voltage regulator
AVR/ALR	auto voltage regulator/automatic load regulator system
AVT	all volatile treatment
AWS	American Welding Society
B.A.	boric acid
B/A	boric acid
BA	burnable absorber
BAC	bounding analysis curve
BAP	boric acid transfer pump
BAS	boric acid system
BAT	boric acid tank
BBR	BBR VT International Ltd
BDB	beyond design basis
BE	best estimate
BEF	best estimate flow
BHEP	basic human error probability
BHN	Brinell hardness number
BISI	bypassed and inoperable status indication
BNL	Brookhaven National Laboratory
BOC	beginning-of-cycle
BOL	beginning-of-life
BOP	balance of plant
BRL	Ballistics Research Laboratory
BRS	boron recycle system
BTP	branch technical position

ACRONYMS AND ABBREVIATIONS (Continued)

BTU	british thermal unit
BWR	boiling water reactor
BWROG	boiling water reactor owners' group
C/V	containment vessel
CAGI	Compressed Air and Gas Institute
CAGS	compressed air and gas system
CAMS	containment atmosphere monitoring system
CAOC	constant axial offset control
CAS	central alarm station
CASS	compressed air supply system
CAV	cumulative absolute velocity
CBP	computer-based procedure
CBS	condenser water box vacuuming priming system
CCDP	conditional core damage probability
CCF	common cause failure
CCFP	conditional containment failure probability
CCTV	closed circuit television
CCW	component cooling water
CCWP	component cooling water pump
CCWS	component cooling water system
CCWT	component cooling water train
CD	complete dependence
CDF	core damage frequency
CDR	Certified Design Report
CDS	condensate system
CEDE	committed effective dose equivalent
CET	containment event tree
CF	core flooding
CFCS	containment fan cooler system
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
CFS	condensate and feedwater system
CGS	compressed gas supply system
CHF	critical heat flux
CHP	charging pump
CHR	cooling water/hot water return
CHS	containment hydrogen monitoring and control system

ACRONYMS AND ABBREVIATIONS (Continued)

CI	containment isolation
CIS	containment internal structure
CIV	containment isolation valve
CMT	chemical mixing tank
CMTR	Certified Material test Report
COC	certificate of compliance
COL	Combined License
COLA	Combined License Application
COLR	core operating limits report
COT	channel operational test
CPET	containment phenomenological event tree
CPG	containment performance goal
CPS	condensate polishing system
CPU	central processing unit
Cr	chromium
CRDM	control rod drive mechanism
CRDMCS	control rod drive mechanism control system
CRDS	control rod drive system
CRE	control room envelope
CRHS	control room habitability system
CRMP	configuration risk management program
CS	containment spray
CS/RHR	containment spray/residual heat removal
CS/RHRS	containment spray/residual heat removal system
CSA	channel statistical accuracy
CSDRS	certified seismic design response spectra
CSET	containment system event tree
CSNI	Committee on the Safety of Nuclear Installations
CSS	containment spray system
CSTF	condensate storage and transfer facilities
CT	compact tension
CTS	condenser tube cleaning equipment
CTW	cooling tower
CV	control valve
CVCS	chemical and volume control system
CVDP	C/V reactor coolant drain pump
CVDT	containment vessel reactor coolant drain tank

ACRONYMS AND ABBREVIATIONS (Continued)

CVN	charpy v-notch
CVTR	Carolinas-Virginia Tube Reactor
CVVS	containment ventilation system
CWS	circulating water system
DAAC	diverse automatic actuation cabinet
DAS	diverse actuation system
DBA	design-basis accident
DBE	design-basis event
DBFL	design-basis flooding level
DBPB	design-basis pipe break
dc	direct current
DC/PS	dc power system
DCD	Design Control Document
DCH	direct containment heating
DCS	data communication system
DDE	deep dose equivalent
DDT	deflagration to detonation transition
DE	dose equivalent
DECLG	double-ended cold leg (pump discharge) guillotine
DEGB	double-ended guillotine break
DEH	digital electro-hydraulic
DEHLG	double-ended hot leg guillotine
DEPSG	double-ended pump suction guillotine
DF	decontamination factor
DHP	diverse HSI panel
DICS	digital instrumentation and control system
DIF	dynamic impact factor
DLF	dynamic load factor
DMIMS	digital metal impact monitoring system
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOF	degree of freedom
DOP	dioctyl phthalate
DOT	Department of Transportation
D-RAP	design reliability assurance program
DRS	storm drain system
DS	decontamination system

ACRONYMS AND ABBREVIATIONS (Continued)

DSS	digital safety system
DTM	design team manager
DV	depressurization valve
DVI	direct vessel injection
DWS	demineralized water system
DWTSS	demineralized water transfer and storage system
E/O	electrical to optical (or optical to electrical)
EAB	exclusion area boundary
EAC/PSS	emergency ac power supply system
EARWS	evacuation alarm and remote warning system
ECC	emergency core cooling
ECCS	emergency core cooling system
ECOM	error of commission
ECP	electrical corrosion potential
ECS	emergency communications system
ECWS	essential chilled water system
EDE	effective dose equivalent
EDS	equipment drain system
EF	error factor
EFPD	effective full power days
EFW	emergency feedwater
EFWPAVS	emergency feedwater pump area HVAC system
EFWS	emergency feedwater system
EH/C	electric heating coil
EHGS	turbine electro-hydraulic governor control system
EIA	Energy Information Administration
EIF	electrical interface system
ELS	emergency letdown system
EMI	electromagnetic interference
EOC	end-of-cycle
EOF	emergency operations facility
EOL	end-of-life
EOM	error of omission
EOP	emergency operating procedure
EOST	electrical overspeed trip device
EPA	containment electric penetration assembly
EPG	emergency procedure guideline

ACRONYMS AND ABBREVIATIONS (Continued)

EPRI	Electric Power Research Institute
EPS	emergency power source
EQ	environmental qualification
EQDP	equipment qualification data package
EQSDS	equipment qualification summary data sheet
ERAC	electrical rigid aluminum conduit
ERDA	Energy Research and Development Administration (now U.S. DOE)
ERDS	emergency response data system
ERSC	electrical rigid steel conduit
ESF	engineered safety features
ESFAS	engineered safety features actuation system
ESFVS	engineered safety features ventilation system
ESLS	electrical system logic system
ESQDSR	Equipment Qualification Data Summary Report
ESQR	Equipment Seismic Qualification Report
ESW	essential service water
ESWP	essential service water pump
ESWPT	essential service water pipe tunnel
ESWS	essential service water system
ESX	ex-vessel steam explosion
ET	event tree
ETSB	effluent treatment system branch
EV	elevartor
EZB	exclusion zone boundary
FA	function allocation
FAB	feed and bleed
FAC	flow-accelerated corrosion
FATT	fracture appearance transit temperature
FCC	Federal Communications Commission
FCV	feedwater control valve
FDS	fire detection systems
FE	finite element
Fe	iron
FEM	finite element method
FHA	fire hazard analysis
FHS	fuel handling system
FIRS	foundation input response spectra

ACRONYMS AND ABBREVIATIONS (Continued)

FLB	feedwater line break
FLML	failure to maintain water level
FMEA	failure modes and effects analysis
FOS	fuel oil storage and transfer system
FP	fission product
FPP	fire protection program
FPS	fire protection system
FRA	functional requirements analysis
FS	fuel system
FSAR	Final Safety Analysis Report
FSHS	fuel storage and handling system
FSS	fire protection water supply system
FT	fault tree
FTS	fuel transfer system
FV	Fussell Vesely
FVW	Fussell Vesely worth
FWLB	feedwater line break
FWS	feedwater system
g	gravity
GA	general arrangement
Gd2O3	gadolinia
GDC	General Design Criteria
GFO	governor free operation
GLBS	generator load break switch
GMAW	gas metal arc welding
GMRS	ground motion response spectra
GOMS	goals, operators, methods, and selection
GSS	gland seal system
GT/B	gas turbine building
GT/GS	gas turbine generator system
GTAW	gas tungsten arc welding
GTG	gas turbine generator
GTPS	generator transformer protection system
GWMS	gaseous waste management system
HA	human action
HAZ	heat-affected zone
HCI	hydrochloric acid

ACRONYMS AND ABBREVIATIONS (Continued)

HCLPF	high confidence of low probability of failure
HCS	generator hydrogen and CO2 system
HD	high dependence
HDSR	historical data storage and retrieval
HE	human error
HED	human engineering deficiency
HEI	Heat Exchange Institute
HELB	high-energy line breaks
HEP	human error probability
HEPA	high-efficiency particulate air
HF	human factors
HFE	human factors engineering
HFEVTM	HFE V&V team manager
HFP	hot full power
HHI	high head injection
HHIS	high-head injection system
HI	hydriodic acid
HID	high intensity discharge
HIS	hydrogen ignition system
HJTC	heated junction thermocouple
HMS	hydrogen monitoring system
HNO3	nitric acid
HPME	high pressure melt ejection
HPT	high-pressure turbine
HRA	human reliability analysis
HRC	Rockwell C hardness
HSI	human-system interface
HSIS	human-system interface system
HSLA	high strength low alloy
HSSC	highly safety significant component
HT	holdup tank
HVAC	heating, ventilation, and air conditioning
HX	heat exchanger
HZP	hot zero power
I&C	instrumentation and control
I/F	interface
I/O	input/output

ACRONYMS AND ABBREVIATIONS (Continued)

IAS	instrument air system
IBC	International Building Code
ICC	inadequate core cooling
ICCC	incore control component
ICDP	incremental core damage probability
ICIGS	incore instrument gas purge system
ICIS	incore instrumentation system
ICS	instrumentation and control system
IE	initiating event
IEEE	Institute of Electrical and Electronics Engineers
IESNA	Illuminating Engineering Society of North America
IFPRA	Internal flood probalistic risk assessment
IG	implementation guideline
IGA	intergranular attack
IHL	induced hot leg rupture
ILRT	integrated leak rate test
IPB	isolated phase busduct
ISA	Instrumentation, Systems, and Automation Society
ISI	inservice inspection
ISLH	inservice leak and hydrostatic
ISM	independent support motion
ISO	International Standards Organization
ISRS	in-structure response spectra
IST	inservice testing
ITAAC	inspections, tests, analyses, and acceptance criteria
ITC	isothermal temperature coefficient
ITP	initial test program
ITS	industrial television system
ITV	industrial television
IV	intercept valve
JAERI	Japan Atomic Energy Research Institute
JAPEIC	Japan Power Engineering and Inspection Corporation
J-APWR	Japanese - Advanced Pressurized Water Reactor
JNES	Japan Nuclear Energy Safety Organization
JRC	Joint research Centre
JSME	Japan Society of Mechanical Engineers
KZK	Kernforschungszentrum Karlsruhe

ACRONYMS AND ABBREVIATIONS (Continued)

LB	lower bound
LBB	leak before break
LBLOCA	large break loss of coolant accident
LCO	limiting condition for operation
LCS	local control station
LD	low dependence
LDP	large display panel
LER	licensee event report
LERF	large early release frequency
LHR	linear heat rate
LHSI	low-head safety injection
LiOH	lithium hydride
LMS	leak monitoring system
LOCA	loss-of-coolant accident
LOESW	loss of essential service water
LOF	left-out-force
LOFF	loss of feedwater flow
LOOP	loss of offsite power
LOP	loss of power
LPDS	large panel display (LPD) system
LPMS	loose parts monitoring system
LPSD	low-power and shutdown
LPT	low-pressure turbine
LPZ	low-population zone
LRB	last rotation blade
LRF	large release frequency
LRT	leakage rate testing
LS	lighting system
LSSS	limiting safety system settings
LTOP	low temperature overpressure protection
LWMS	liquid waste management system
LWR	light-water reactor
M signal	main control room isolation signal
M/D	motor-driven
M/G	motor generator
MAAP	modular accident analysis program
MACCS2	MELCOR accident Consequence Code system 2

ACRONYMS AND ABBREVIATIONS (Continued)

MCC	motor control center
MCCB	molded case circuit breaker
MCCI	molten core concrete interaction
MCES	main condenser evacuation system
MCP	main coolant piping
MCR	main control room
MCREFS	main control room emergency filtration system
MCRVS	main control room HVAC system
MD	movable neutron detector
MDF	mechanical design flow
MELB	moderate-energy line break
MELCO	Mitsubishi Electric Corporation
MFBRV	main feedwater bypass regulation valve
MFCV	main feedwater check valve
MFIV	main feedwater isolation valve
MFRV	main feedwater regulation valve
MFW	main feedwater
MFWS	main feedwater system
MG	main generator
MGL	multiple greek letter
MHI	Mitsubishi Heavy Industries, Ltd.
MLOCA	medium pipe break LOCA
MLS	maintenance lifting system
MMF	minimum measured flow
MMI	man-machine-interface
MN	mega newton
MoS2	molybdenum disulfide
MOST	mechanical overspeed trip devices (turbine)
MOV	motor operated valve
MSBIV	main steam bypass isolation valve
MSCV	main steam check valve
MSDIV	main steam drain line isolation valve
MSDV	main steam depressurization valve
MSFWS	main steam and feedwater system
MSIV	main steam isolation valve
MSLB	main steam line break
MS/R	moisture separator reheater

ACRONYMS AND ABBREVIATIONS (Continued)

MSR	maximum steaming rate
MSRV	main steam relief valve
MSRVBV	main steam relief valve block valve
MSS	main steam supply system
MSS-SP	Manufacturer Standardization Society-Standard Practice
MSSV	main steam safety valve
MT	main transformer
MTC	moderator temperature coefficient
MTCDS	main turbine control and diagnostic system
MTCV	main turbine control valves
MTS	main transmission system
MTSV	main turbine stop valve
MTTR	mean time to repair
MV	medium voltage
N Center	Nuclear Energy Systems Engineering Center
N/A	not applicable
N/E	normal/emergency
N/ELS	normal/emergency lighting system
NaTB	sodium tetraborate decahydrate
NCIG	National Construction Issues Group
NDE	nondestructive examination
NDRC	National Defense Research Committee
NDS	non-radioactive drain system
NDTT	nil ductility transition temperature
NEC	National Electric Code
NEI	Nuclear Energy Institute
NEMA	National Electrical Manufacturer Association
NESH	Nuclear Energy Systems Headquarters
NFPA	National Fire Protection Association
NFR	new fuel rack
NGHS	noble gas holdup system
NIS	nuclear instrumentation system
NIST	National Institute of Standards and Technology
NLS	normal lighting system
non-ECWS	non-essential chilled water system
non-ESW	non-essential service water
NPGS	nuclear power generating stations

ACRONYMS AND ABBREVIATIONS (Continued)

NPS	nominal pipe size
NPSH	net positive suction head
NQA	nuclear quality assurance
NR	neutron reflector
NRC	U.S. Nuclear Regulatory Commission
NRCA	non-radiological controlled area
NRDS	non-radioactive drain system
NS	non-seismic
NSSS	nuclear steam supply system
NUREG	NRC Technical Report Designation (Nuclear Regulatory Commission)
O/B	outside building
OBE	operating-basis earthquake
OC	operator console
OD	outside diameter
ODCM	offsite dose calculation manual
ODSCC/IGA	outside diameter stress corrosion cracking/intergranular attack
OECD	Organization for Economic Cooperation and Development
OEM	original equipment manufacturer
OEPS	onsite electrical power system
OER	operating experience review
OHLHL	overhead heavy load handling system
OLM	on-line maintenance
OLTC	on-load tap changer
OMCS	off-microwave communication system
OMS	operation and monitoring system
OP	over-pressure
OPΔT	over power delta-T
OPC	overspeed protection controller
OPDMS	on-line power distribution monitoring system
OPS	offsite power system
OPSDS	onsite power system distribution system
O-RAP	operational reliability assurance program
ORE	occupational radiation exposure
ORIGEN2	buildup, decay, and processing of radioactive materials calculation code (ORNL)
ORNL	Oak Ridge National Laboratory
OS	operating system
OSD	operational sequence diagram

ACRONYMS AND ABBREVIATIONS (Continued)

OT	over temperature
OTΔT	over temperature delta-T
OTPS	over Temperature delta-T protection system
OTS	offsite transmission systems
OVDR	over-drain
P signal	containment isolation signal
P&ID	piping and instrumentation diagram
P/T	pressure and temperature
PA	postulated accident
PA/PL	public address system/page
PABX	private automatic branch telephone exchange
PAM	post accident monitoring
PASS	post accident sampling system
PAW	plasma arc welding
PC	plant condition
PCCV	prestressed concrete containment vessel
PCMI	pellet/cladding mechanical interaction
PCMS	plant control and monitoring system
PCT	peak cladding temperature
PDS	plant damage state
PERMS	process effluent radiation monitoring and sampling system
PGA	peak ground acceleration
PGS	plant gas systems
PGSS	primary gaseous sampling system
PHT	RCP purge water head tank
PIV	pressure isolation valve
PLHR	peak linear heat rate
PLS	plant lighting system
PLSS	primary liquid sampling system
PM	project manager
PMF	probable maximum flood
PMP	probable maximum precipitation
PMW	primary makeup water
PMWS	primary makeup water system
POL	problem oriented language
POS	plant operational state
POV	power-operated valve

ACRONYMS AND ABBREVIATIONS (Continued)

PPASS	process and post-accident sampling systems
PPS	preferred power supply
PRA	probabilistic risk assessment
PRDF	probabilistic risk assessment fundamental
PRDS	pressurizer and relief discharge system
PRS	pressure relief system
PRSV	pressurizer safety valve
PRT	pressurizer relief tank
PS	prestress
PS/B	power source building
PSB	power systems branch
PSF	performance shaping factor
PSFSV	power source fuel storage vault
PSI	preservice inspection
PSMS	protection and safety monitoring system
PSS	process and post-accident sampling system
PSWS	potable and sanitary water systems
PT	liquid penetrant examination method
PTFE	polytetrafluoroethylene
PTLR	pressure and temperature limits report
PTS	pressurized thermal shock
PWR	pressurized-water reactor
PWS	potable water system
PWSCC	prevention of primary water stress corrosion cracking
QA	quality assurance
QAP	quality assurance program
QAPD	quality assurance program document
QPTR	quadrant power tilt ratio
R/B	reactor building
RADTRAD	radionuclide transport, removal, and dose
RAI	request for additional information
RAP	reliability assurance program
RAT	reserve auxiliary transformer
RAW	risk achievement worth
RCA	radiological controlled area
RCC	rod control cluster
RCCA	rod cluster control assembly

ACRONYMS AND ABBREVIATIONS (Continued)

RCCS	reactor cavity cooling system
RCDT	reactor coolant drain tank
RCL	reactor coolant loops
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
REA	rod ejection accident
RESAR	reference safety analysis report
RF	recovery factors
RFI	radio frequency interference
RFT	resin fill tank
RG	Regulatory Guide
RHR	residual heat removal
RHRS	residual heat removal system
RIA	reactivity initiated accident
RICT	risk-informed completion time
RIM	required input motion
RLE	review level earthquake
RMAT	risk management action time
RMS	radiation monitoring system
RMTS	risk-managed technical specifications
RO	reactor operator
RPI	rod position indication
RPS	reactor protection system
RPV	reactor pressure vessel
RRS	required response spectra
RRW	risk reduction worth
RSC	remote shutdown console
RSR	remote shutdown room
RSS	remote shutdown system
RSSS	reactor safety shutdown system
RSV	reheat stop valve
RT	reactor trip
RTB	reactor trip breaker
RTD	resistance temperature detector
RTDP	revised thermal design procedure
RTNDT	reference nil ductility temperature

ACRONYMS AND ABBREVIATIONS (Continued)

RTNSS	regulatory treatment of non safety-related systems
RTP	rated thermal power
RTS	reactor trip system
RV	reactor vessel
RVH	reactor vessel head
RVR	reactor vessel rupture
RVWL	reactor vessel water level
RWMS	radioactive waste management system
RWP	refueling water recirculation pump
RWS	refueling water storage system
RWSAT	refueling water storage auxiliary tank
RWSP	refueling water storage pit
RWSPVS	refueling water storage pit vent system
RY	reactor-year
S signal	safety injection signal
SAFDL	specified acceptable fuel design limits
SAM	seismic anchor motion
SAMDA	severe accident mitigation design alternative
SAMG	severe accident management guideline
SAS	secondary alarm station
SAT	systems approach to training
SAW	submerged arc weld
SBLOCA	small break loss of coolant accident
SBO	station blackout
SC	steel concrete
SCAVS	safeguard component area HVAC system
SCC	stress corrosion cracking
SCIS	secondary side chemical injection system
SDCV	spatially dedicated continuously visible
SDM	shutdown margin
SDV	safety depressurization valve
sec	second, seconds
SECY	Secretary of the Commission, Office of the (NRC)
SER	significant event report
SFDP	safety function determination program
SFP	spent fuel pit
SFPC	spent fuel pit cooling

ACRONYMS AND ABBREVIATIONS (Continued)

SFPCS	spent fuel pit cooling and purification system
SG	steam generator
SGBDS	steam generator blowdown system
SGBDSS	steam generator blowdown sampling system
SGBSS	steam generator blowdown sampling system
SGTR	steam generator tube rupture
SGWFCV	steam generator water filling control valve
SI	safety injection
SIP	safety injection pump
SIS	safety injection system
SL	safety limit
SLB	steam line break
SLBO	steam line break/leak outside containment
SLOCA	small pipe break LOCA
SLS	safety logic system
SMA	seismic margin analysis
SMACNA	sheet metal and air conditioning contractors national association
SMAW	shielded metal arc weld
SNL	Sandia National Laboratories
SOER	significant operating experience report
SOR	senior reactor operator
SORV	stuck-open relief valve
SPCS	steam and power conversion system
SPDS	safety parameter display system
SPLB	NRC plant systems branch
SPS	sound powered system
SPTS	sound powered telephone system
SQR	Seismic Qualification Report
SR	surveillance requirement
SRHV	spent resin holding vessel
SRM	staff requirements memorandum
SRO	senior reactor operator
SRP	Standard Review Plan
SRSS	square root sum of the squares
SRST	spent resin storage tank
SRV	safety relief valve
SS	stainless steel

ACRONYMS AND ABBREVIATIONS (Continued)

SSA	signal selection algorithm
SSAS	station service air system
SSC	structure, system, and component
SSE	safe-shutdown earthquake
SSEA	safe-shutdown earthquake anchor
SSEI	safe-shutdown earthquake inertia loads
SSI	soil-structure interaction
SSS	secondary sampling system
SST	station service transformer
STA	shift technical advisor
STDP	standard thermal design procedure
SV	(main) stop valve
SW	service water
SWMS	solid waste management system
T/B	turbine building
T/D	turbine driven
T/G	turbine generator
TADOT	trip actuating device operational test
Tavg	average temperature
TBD	to be determined
TBE	thin bed effect
TBS	turbine bypass system
TBV	turbine bypass valve
TC	thermocouple
Tcold	cold temperature
TCS	turbine component cooling water system
TD	theoretical density
TDC	thermal diffusion coefficient
TDF	thermal design flow
TDS	total dissolved solids
TEDE	total effective dose equivalent
T-H	thermal hydraulic
THERP	technique for human error rate prediction
Thot	hot temperature
TIA	Telecommunication Industry Association
TIG	tungsten inert gas
TI-SGTR	temperature induced steam generator tube rupture

ACRONYMS AND ABBREVIATIONS (Continued)

TMI	Three Mile Island
TPS	turbine protection system
TRANS	general transients
Tref	reference temperature
TRS	test response spectrum
TS	technical specification
TS	telecommunication system
TSC	technical support center
TSCVS	technical support center (TSC) HVAC system
TSIS	turbine supervisory instrument system
TT	thermal treatment
TVS	turbine building area ventilation system
U.S.	United States
UAT	unit auxiliary transformer
UB	upper bound
UCC	underclad cracking
UHS	ultimate heat sink
UHSRS	ultimate heat sink related structures
UL	Underwriters Laboratories
UPS	uninterruptible power supply
URD	Utility Requirement Document
US, U.S.	United States
USA	United States of America
USM	uniform support motion
UT	ultrasonic examination method
UTS	ultimate tensile strength
UV/IR	ultraviolet/infrared
V&V	verification and validation
VA	vital area
VAC	volts alternating current
VAS	auxiliary building ventilation system
VCS	containment ventilation system
VCT	volume control tank
VDS	vent drain system
VDU	visual display unit
VE	vital equipment
VFTP	ventilation filter testing program

ACRONYMS AND ABBREVIATIONS (Continued)

VRS	engineered safety features ventilation system
Vs	shear wave velocity
VSL	VSI International, ltd.
VWO	valve wide open
VWS	chilled water system
WCAP	Westinghouse Commercial Atomic Power (report)
WG	water gauge
WHP	waste holdup tank pump
WHT	waste holdup tank
WMS	waste management system
WMT	waste monitor tank
WPS	welding procedure specifications
wt	weight
WWS	waste water system
ZD	zero dependency
ZOI	zone of influence
ZPA	zero period acceleration
ΔT	delta temperature (temperature difference or change)

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

1.1 Introduction

Mitsubishi Heavy Industries, Ltd. (MHI) has designed the US-APWR as an advanced light water reactor (LWR) plant. The US-APWR is described in this Design Control Document (DCD), which is submitted to the U.S. Nuclear Regulatory Commission (NRC) for review and approval under the provisions of the Code of Federal Regulations, 10 CFR 52. MHI is requesting NRC approval and certification for the US-APWR standard design.

1.1.1 Plant Location

The US-APWR standard nuclear power plant is designed to be constructed on a site with parameters including those described in Chapter 2, "Site Characteristics", of this DCD. Parameters described in Chapter 2 relate to seismology, hydrology, meteorology, geology, and other site-related characteristics. The Combined License (COL) Applicant is to identify the actual plant location.

1.1.2 Containment Type

The US-APWR containment vessel consists of a prestressed, post-tensioned concrete structure with a cylindrical wall, hemispherical dome, and flat reinforced concrete foundation slab. The inside surface of the structure is lined with carbon steel. The US-APWR reactor and reactor coolant system (RCS) are completely enclosed in the prestressed concrete containment vessel (PCCV). The PCCV is designed to assure essentially no leakage of radioactive materials to the environment, even if a major failure of the reactor coolant system were to occur.

1.1.3 Reactor Type

The US-APWR reactor is a Mitsubishi-designed 4-loop pressurized water reactor (PWR).

1.1.4 Power Output

The US-APWR net electrical power rating is approximately 1600 MWe, depending on site conditions. The rated core thermal power level of the US-APWR is 4451 MWt. In some safety evaluations, the core thermal power level of 4540MWt is used for taking 2 percent allowance for calorimetric error into account.

1.1.5 Schedule

The COL Applicant is to provide the scheduled completion date and estimated commercial operation date of nuclear power plants referencing the US-APWR standard design.

1.1.6 Format and Content

1.1.6.1 Regulatory Guide 1.206

The format and content of this DCD are based on the guidance contained in NRC Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)", Revision 0, June 2007. The DCD chapter, section, subsection, and paragraph headings are the same as those used in RG 1.206.

1.1.6.2 Standard Review Plan

The preparation of this document has followed the technical guidance provided in NRC's Standard Review Plan, NUREG-0800. A detailed evaluation of conformance with the Standard Review Plan is provided in Subsection 1.9.2.

1.1.6.3 Text, Tables and Figures

The tables are identified by the section number followed by a sequential number (for example, Table 1.2-3 is the third table of Section 1.2). Tables are provided at the end of the applicable sections immediately following the text. Drawings, sketches, etc., are treated as figures. They are also numbered sequentially by section and are placed at the end of the applicable sections, following the text.

1.1.6.4 Page Numbering

Text pages are numbered sequentially and are identified by the section number followed by a sequential number.

1.1.6.5 Proprietary Information

This document has no proprietary information. Some portions of this document are classified as sensitive and withheld from public disclosure pursuant to 10 CFR 2.390 and Regulatory Issue Summary (RIS) 2005-26. Such material is clearly marked, and the withheld material is separately provided for the NRC review.

1.1.6.6 Acronyms and Abbreviations

The Acronym and Abbreviation List at the beginning of this chapter provides a list of acronyms and abbreviations used throughout this application. The list also provides the US-APWR system designators along with their names. Acronyms and Abbreviations are defined in the individual chapters in which they are used.

1.1.6.7 Combined License Information

COL 1.1(1) *The COL Applicant is to provide scheduled completion date and estimated commercial operation date of nuclear power plants referencing the US-APWR standard design.*

COL 1.1(2) *The Combined License (COL) Applicant is to identify the actual plant location.*

1.2 General Plant Description

This section describes design criteria, safety considerations and operating characteristics for the US-APWR, provides site characteristics, and describes the plant arrangement.

1.2.1 Principal Design Criteria, Safety Considerations and Operating Characteristics

The most important aspect of the US-APWR design philosophy is the protection of the public and workers by the installation of effective barriers against radioactive materials and other hazards. Main design concepts of the US-APWR are utilization of proven technology and a well-balanced safety design. Significant experience in the design, fabrication, installation, construction, and operation of PWRs in Japan has resulted in proven technologies being developed by MHI. These technologies have been incorporated in the design of the US-APWR. The US-APWR features highly reliable prevention functions, well-established mitigation systems with active safety functions and passive safety functions, and measures that protect against beyond design basis accidents. These three functions are integrated in a balanced US-APWR design, which has been developed using a deterministic design approach, and the application of risk management technology and probabilistic risk assessment. Furthermore, the reliability of the physical barriers and the protection level are improved based on the concept of defense-in-depth, which is applied from normal operation to beyond design basis accidents. The design of the US-APWR based on the above principles is in accordance with U. S. regulatory requirements.

This section provides an overview of the US-APWR principal design criteria, safety considerations and operating characteristics.

1.2.1.1 Principal Design Criteria

The US-APWR is designed to have its safety design made on the following basic principles.

(1) The US-APWR is designed not to result in radiation exposures exceeding the allowable standard specified under U. S. regulatory requirements.

(2) The US-APWR is made highly reliable throughout the design, manufacture, construction, and test and inspection stages, and is so designed that its operators are able to take a countermeasure by an alarm against the abnormal transient condition during operation caused by erroneous operation and so forth. Even in case such a corrective measure is not taken, the reactor's inherent safety and the safety protection system actuation can prevent such condition from developing into a major accident.

(3) The US-APWR is equipped with defense-in-depth to prevent the radioactive fission products released from fuel from reaching the offsite areas, so as to assure the safety of the public in the area surrounding the plants, even in an accident.

(4) The design of the US-APWR structures is such that the plant safety is not impaired even by postulated natural phenomena.

(5) In order to prevent the US-APWR safety being endangered by a fire, fire protection measures are incorporated in the design, in accordance with the defense-in-depth concept for fires.

1.2.1.2 Safety Design Criteria

1.2.1.2.1 General

Design and manufacturing of the structures, systems and components (SSCs) that fulfill safety functions shall follow the basic criteria described below:

(1) Design, selection of materials, manufacturing and inspection of SSCs that fulfill safety functions shall comply with standards and criteria that are appropriate, based on the importance of their intended safety functions.

(2) The SSCs that fulfill safety functions shall be designed taking into account the postulated natural environment.

(3) The SSCs that fulfill safety functions shall be designed so that the safety of nuclear facilities is not impaired by external human induced events or by missiles assumed to generate within the reactor facility. Appropriate arrangement shall be also made in the design to protect unauthorized access to the areas of the plant containing safety equipments.

(4) The SSCs that fulfill safety functions shall be designed to assure and maintain sufficient reliability consistent with the importance of their safety functions.

The systems with particularly important safety functions shall be provided with redundancy, diversity and independency, based on their structures, operating principles and nature of the intended safety functions, and shall be designed in such a way that their safety functions may be fulfilled even with loss of off-site power (LOOP) as well as an assumed single equipment failure.

(5) The SSCs that fulfill safety functions shall be designed to allow testing and inspection during reactor operation or shutdown using appropriate methods that are consistent with the importance of their intended safety functions and assure their integrity and operational capabilities.

1.2.1.2.2 Basic Concept of Safety Design

1.2.1.2.2.1 Defense-in-Depth

The defense-in-depth philosophy is the basic principle of the US-APWR safety design. It provides multiple means to accomplish safety functions and to prevent the release of radioactive material.

The objectives of the multiple stages of defense-in-depth are:

- Prevention of equipment abnormal operation and failures
- Detection of equipment failures and control of abnormal operation
- Control of accidents included within the design basis
- Prevention of beyond design basis accidents, and mitigation of consequences of severe accidents.

(1) Prevention of Abnormal Equipment Operation and Failure

The following prevention measures are included in the US-APWR design:

- The reactor is designed to be inherently safe, using self-regulating characteristics, such as the Doppler effect and the moderator density effect, to prevent transients or accidents.
- The causes of abnormal operation are minimized by providing sufficient margin, improving equipment and control system reliability, and performing strict quality control during the manufacture of components.
- There is enhanced reliability of the reactor coolant pressure boundary, via the use of Alloy 690 for the vessel head nozzle, and the achievement of cold temperature (T_{cold}) at the vessel head plenum by increasing core bypass flow.
- SSCs are designed so that occurrences of violations of Technical Specifications are reduced through on-line maintenance by increasing redundancy of functions, such as four trains of emergency core cooling, electrical power, instrumentation and control, and plant cooling water systems.
- Improvement of safety during shutdown - Based on the shutdown probabilistic risk assessment, measures are provided to enhance plant safety by improving operational management.
- Reduction of operator work load and enhanced reliability of instrumentation and control (I&C) systems - The plant has an advanced control room with enhanced operability, and integrated digital technology with redundant architecture, These features reduce operator work load and increase the reliability of the I&C system.

(2) Detection of Equipment Failures and Control of Abnormal Operation

In the case of certain system failures or human errors occurring during operation, abnormal conditions are detected at an early stage, and the following measures are taken for prevention of further progress of the abnormal conditions:

- Signals - Full four-train protection systems with reactor trip breakers are provided to initiate a reactor trip when a system failure or human error occurs and there is the potential for phenomena capable of causing further deterioration in the plant status.

- Shutdown system - A reactor safe shutdown system is provided, which consists of the control rods for reactor trip. Furthermore, cold shutdown can be achieved through several means including by the emergency core cooling system (ECCS), emergency letdown line, and safety depressurization and vent system.

- If a malfunction occurs, the following protection measures provide defense-in-depth:

  - Sufficient design margin

  - Use of fail-safe design approaches where possible

  - Improved reliability of safeguard systems

  - Strict quality control in the manufacture of components

  - Use of redundancy in the design

(3) Control of Accidents Included within the Design Basis

The following measures are taken to prevent the progression of accidents, to mitigate effects of accidents, and to protect both the public and site workers:

- Signals - If accidents do occur, signals initiate engineered safety features.

- Safety systems - High reliability ECCS, which cool the core in response to safeguards signals.

  - Safety systems such as accumulators, safety injection system (SIS), containment spray system (CSS), and emergency feedwater system (EFWS), all have four trains in separate divisions. Additionally, electrical safety systems, emergency power generation, and service water systems all have four trains.

  - The refueling water storage pit in the containment eliminates the need for switchover of suction source for the SIS and CSS.

  - The advanced accumulators, each with a flow damper, have two injection modes: large flow and small flow. The advanced accumulator is a passive component employed to enhance safety.

- Sufficient time for operation - If manual operation is needed, the operator has sufficient time to make the required decisions, and operator actions are generally easy to executes.

- The PCCV is an effective pressure barrier, and serves as a barrier to the dispersion of radioactive materials to the environment.

- The containment spray (CSS) system has four trains of cooling spray that assure the integrity of the PCCV.

- The containment annulus provides an airtight space between the PCCV and the reactor building (R/B). The pressure in annulus is kept negative with respect to ambient atmosphere to prevent the release of radioactive materials to the environment in the case of accidents.

(4) Prevention of beyond design basis accidents, and mitigation of consequences of severe accidents

The US-APWR establishes the following accident measures guided by the use of probabilistic risk assessment. These measures are diverse from those provided by the above safety systems.

Prevention of accidents progressing to beyond design basis accidents:

Measures against anticipated transient without scram (ATWS) - The safety-related reactor protection system is highly reliable due to its independent four train design. The diverse actuation system (DAS), which has functions to prevent ATWS, is installed to provide a response to common cause failure (CCF) of the digital I&C systems and will preclude ATWS events.

Measures against Mid-Loop Operation - To prevent over-draining during mid-loop operation, a loop water level gage and an interlock (actuated by the detection of water level decrease), act to isolate water extraction.

Measures against station blackout - A diversity of emergency power sources is provided to mitigate station blackout (SBO). The design provides the capability of achieving Safe Shutdown to a cool down state following a SBO.

Additional protection against an interfacing system loss-of-coolant accident (LOCA) - Use of higher rated piping in the residual heat removal systems reduces the probability of occurrence of interfacing system LOCA. Even if residual heat removal system isolation valves open due to malfunction during normal operation, reactor coolant from main coolant pipe would flow to the refueling water storage pit without experiencing a pipe break outside containment.

Mitigation of consequences of severe accidents:

Measures against severe accident after core damage - The plant design provides special features for the prevention and/or mitigation of severe accident phenomena such as hydrogen combustion, core debris coolability, temperature-induced steam generator tube rupture (SGTR), high pressure melt ejection and direct containment heating, and long-term containment overpressure.

1.2.1.2.2.2 Probabilistic Safety Target

The following safety goals regarding core damage frequency and large release frequency are defined as the basis for evaluating the safety of the US-APWR:

Core Damage Frequency - The target Core Damage Frequency is less than 10^{-5}/reactor-year for internal and external events during all operation modes.

Large Release Frequency - The target Large Release Frequency is less than 10^{-6}/reactor-year.

1.2.1.2.3 Inherent Safety of Reactor

The reactor core and associated coolant systems are designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for the rapid increase in reactivity. The US-APWR uses low-enriched sintered uranium dioxide pellets as fuel, and has the following inherent feedback characteristics:

(1) The moderator temperature coefficient remains negative under normal operation.

(2) Low enriched-uranium has negative temperature reactivity coefficient due to Doppler effect. Therefore, even if reactivity is suddenly inserted in the reactor, the nuclear power excursion is automatically compensated through Doppler effect as a result of sudden increase in pellet temperature.

1.2.1.2.4 Nuclear Design and Thermal and Hydraulic Design Criteria

1.2.1.2.4.1 Nuclear Design Criteria

The main nuclear design parameters of the US-APWR such as fuel enrichment, number of control rods, and burnable absorber arrangement are established based on the reactivity changes caused by the following phenomena:

(1) Change in quantity of fissile materials, such as uranium-235, due to fuel burning

(2) Moderator temperature rise due to power operation

(3) Fuel temperature rise due to power operation

(4) Accumulation of neutron-absorbing fission products such as xenon and samarium

Reactivity is maintained by the control rod and by the level of soluble boron in the primary coolant. In addition, excess reactivity is controlled using burnable absorber, where necessary.

The control rods are so designed that a hot shutdown can be reached with a sufficient margin, even in the postulated case in which one control rod with maximum reactivity worth is stuck at fully withdrawn position. In addition, soluble boron from the chemical and volume control system (CVCS) provides a cold shutdown capability with a sufficient margin.

Furthermore, the maximum reactivity insertion and reactivity insertion rate of the control rods are so limited that in a postulated accident the integrity of the reactor coolant pressure boundary (RCPB) is not impaired and the reactor internals can perform their core cooling function.

The design is such that power distributions in excess of thermal limits do not occur during normal operation and anticipated operational occurrences.

Moreover, negative reactivity feedback characteristics are maintained by having negative Doppler coefficients and keeping moderator temperature coefficients negative during normal operation. In addition, the design is such that the horizontal power distribution oscillation has damping characteristics and the axial distribution is easily controllable.

1.2.1.2.4.2 Thermal and Hydraulic Design Criteria

The thermal and hydraulic design meets the following criteria intended to maintain fuel integrity.

(1) The hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) phenomenon with at least a 95-percent probability at a 95-percent confidence level during normal operation and anticipated operational occurrences (AOOs).

(2) The fuel rod with the most limiting linear heat rate (kW/ft) does not cause the fuel pellet to melt with at least a 95-percent probability at a 95-percent confidence level during normal operation and AOOs.

(3) Sufficient coolant flow is provided for the reactor core. Conservatively adopted reactor coolant system (RCS) flow and effective core flow are applied in thermal-hydraulic designs.

(4) Hydraulic instability does not occur at any operational modes during normal operation and AOOs.

1.2.1.2.5 Prevention and Control Measures to prevent release of Fission Products

Offsite release of the fission products produced in the fuel is prevented and controlled as follows:

(1) Since sintered uranium dioxide pellets have a retention capability for fission products, most of the fission products produced in the pellets are retained in the pellets.

(2) The fission products released from the sintered uranium dioxide pellet are sealed in the fuel cladding.

(3) Even if the fuel cladding is damaged, the leaked fission products are retained in the RCS.

(4) In case fission products are released due to failure of the RCS or some other mechanism, they are retained by the reactor containment composed of the containment vessel, the annuls portion and other elements of the containment.

Radioactive Waste Management Facilities are installed to treat and manage the radioactive waste produced as a result of plant operation, in order to keep the concentration and the quantity of radioactive substances released to the surrounding environment as low as reasonably acceptable.

1.2.1.2.6 Safety Protection System Design Criteria

The reactor trip system (RTS) and the engineered safety features system are composed of the reactor protection system, the engineered safety features actuation system (ESFAS), the safety logic system and the safety grade human system interface system, are designed to have redundancy and independency so as to actuate when necessary, and also are so designed that their protective functions are not prevented by a single failure. In addition, these systems are designed to fail to a safe state for all credible failures, such as loss of power and so forth.

1.2.1.2.7 Reactor Shutdown System Design Criteria

Two independent reactivity control systems of different design are provided. One of the systems uses control rods and is capable of reliably controlling reactivity changes to assure that under conditions of normal operation, (including anticipated operational occurrences), and with appropriate margin for malfunctions such as stuck rods, specified fuel design limits are not exceeded. The second reactivity control system is capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

The control rod drive system (CRDS) and the chemical and volume control system (CVCS) of the US-APWR are provided so that the core can be made subcritical during normal operation and be maintained subcritical. They are designed on the basis of the following principles:

(1) The CRDS is designed so that the core can be made hot subcritical from full power operation without exceeding acceptable fuel design limits.

(2) Proper operation of the CVCS prior to transients is assumed as an initial condition to evaluate the transients

(3) The reactivity control systems, for which credit is taken in the steam line break (SLB) and LOCA, are the RTS and the ECCS.

1.2.1.2.8 Engineered Safety Features Design Criteria

Engineered safety features (ESFs) are provided in nuclear plants to mitigate the consequences of design-basis accidents or LOCAs, even though the occurrence of these accidents is very unlikely.

The ESFs, composed of the PCCV, containment spray system, containment isolation system, containment hydrogen monitoring and control system, ECCS, main control room (MCR) heating, ventilation, and air conditioning (HVAC) system and annulus emergency exhaust system (AEES), are provided to mitigate the consequences of design-basis accidents or LOCAs.

(1) The ESFs are designed to be highly reliable so they perform their designed functions when their actuation become necessary, and are provided with sufficient redundancy so as to be capable of coping with a single failure.

(2) The ESFs are so designed that the test and inspection to confirm their functions can be carried out upon installation and also during operation or outages even after commencement of operation, in order to confirm that they can perform their functions when necessary through the life of the plant.

(3) The electrical power supply and other driving power sources are designed to be always available for the ESFs to perform their functions.

1.2.1.2.9 Strength Design Criteria

The building, structures, components, piping and their support structures are designed to have sufficient strength and maintain their design functions under loading conditions such as dead loads, internal pressure, external pressure, thermal loads and seismic loads.

1.2.1.2.10 Design Criteria for Natural Phenomena

Earthquake, tornado, and hurricane events are considered as natural phenomena, in accordance with design requirements of 10 CFR 50, Appendix A, GDC 2. SSCs important to safety are designed to withstand the effects of design basis natural phenomena.

- Earthquake - Design response spectra are developed based on RG 1.60. The latest information on design response spectra is considered. Design response spectra envelop potential sites.

- Tornado - Tornado wind speed is based on RG 1.76 and SRP 2.3.1. Static analysis is employed.

- Hurricane - Hurricane wind speed is based on American Society of Civil Engineers (ASCE) 7-05. Static analysis is employed.

1.2.1.2.11 Design Criteria for Internal and External Events

1.2.1.2.11.1 Pipe Rupture Protection

The SSCs important to safety are protected against the dynamic effect associated with a postulated pipe rupture based on General Design Criteria (GDC) 4 of Appendix A to 10CFR50. Leak before break (LBB) evaluation procedure is applied to some of the American Society of Mechanical Engineers (ASME) Class 1 piping and main steam piping so that the dynamic effect of pipe rupture is eliminated. Evaluation and design are based on the damage configuration assumed.

- For a pipe break, the evaluation considers pipe whip and pipe internal load, jet impingement, compartment pressurization, environmental effects, and flooding. The design includes physical separation, protective enclosures, and pipe whip restraints.

- For a pipe crack, the evaluation considers environmental effects and flooding, and the design includes physical separation and protective enclosures.

1.2.1.2.11.2 Missile Protection

The SSCs important to safety are designed to withstand the effects of missiles based on GDCs 2 and 4 of 10CFR50, Appendix A.

- Internally generated missiles are considered to have as their sources pressurized components, high-energy piping, and rotating equipment. Design evaluation/mitigation methods include locating the system or component within a missile-proof structure, and to physically separating redundant systems or components for the missile path or range (SRP 3.5.1.1, 3.5.1.2).

- Turbine missiles are defined as consisting of fragments from the turbine disk or the turbine's internal structure. The design evaluation/mitigation goal is to assure that the probability of unacceptable damage from turbine missiles is less than or equal to 10^{-7} per year for an individual plant (SRP 3.5.1.3, RG 1.115).

- Externally generated missiles are considered as tornado missiles. The design evaluation/mitigation is to establish the ability of seismic Category I structures and/or missile barriers to withstand the effect of tornado missiles (SRP 3.5.1.4).

1.2.1.2.11.3 Fire Protection

The fire protection design of the US-APWR satisfies GDC 3 of Appendix A to 10CFR50 and meets the guidance of SRP 9.5.1. The fire protection system of the US-APWR has three objectives based on the defense-in-depth concept: fire prevention, detection and extinguishing of fires, and mitigation of the adverse effects of fires.

The fire protection systems are installed so as to minimize the adverse effects of fires on SSCs important to safety. Safe shutdown can be achieved assuming that all equipment in any one fire area (excluding the control room and containment) is rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible.

1.2.1.3 Power Capability

- Net electrical power to the grid for the US-APWR is approximately 1600 MWe, depending on site conditions. The nuclear steam supply system power rating, which combines core power plus reactor coolant pump heat, is 4466 MWt.
- Within a load range of 15% to 100% of full power, the US-APWR is designed to accept a step load increase or decrease of 10%, without reactor trip or steam dump system actuation.
- The US-APWR is designed to accept a load rejection of 100% from full power to house loads (with continued stable operation of house loads), without reactor trip or operation of the pressurizer or steam generator (SG) safety valves.
- Within a load range of 15% to 100% power, the US-APWR is designed to accept ramp load changes of 5% per minute, without reactor trip or steam dump actuation (subject to core power distribution limits).

1.2.1.4 Reliability and Availability

The US-APWR designs of major power generation components (steam generators, turbine generator, reactor coolant pumps, nuclear fuel, and reactor internals) are based on evolution of proven designs. The components upon which the US-APWR designs are based have operated with excellent reliability and availability in existing nuclear power plants, and modifications have been made to these designs to further improve their reliability and availability. The overall US-APWR availability goal is 95%. This figure allows for forced and planned outages. The design objective for the US-APWR, without replacement of the reactor vessel, is 60 years.

1.2.1.5 System Design Description

1.2.1.5.1 Reactor and Core

1.2.1.5.1.1 Fuel System

The design of the US-APWR fuel system (fuel rod, fuel assembly and in-core control components) is the same as that of the Japanese - Advanced Pressurized Water Reactor (J-APWR) or the Mitsubishi current 17x17 fuel system except for axial length. The Mitsubishi current 17x17 fuel system has demonstrated high reliability, sustaining negligible low fuel rod failure rate, as evidenced through significant irradiation experience in Japan. The major design features of the US-APWR fuel, as compared to those of the Mitsubishi current fuel, are changing the active fuel length from 12 ft. to approximately 14 ft., and the number of grid spacers from 9 to 11. All of the advanced technologies incorporated into the Mitsubishi current 17x17 fuel assembly for higher burnup are applied for the US-APWR fuel assembly. The ZIRLO™ [1] cladding tube has demonstrated high corrosion resistance under demanding conditions. Pellet density of 97% theoretical density (TD), as opposed to the conventional 95% TD, improves fuel cycle cost by increasing the amount of uranium in the core. The enlarged rod plenum volume also increases the margin for rod internal pressure buildup caused by fission product gas released (especially under high power operation at high burnup), thus improving safety. Higher gadolinia (Gd_2O_3) content up to a maximum of 10 wt% enables flexible core operation to higher burnup.

Design bases for the US-APWR fuel rod are established to prevent fuel rod failure and the fuel system damage in terms of fuel temperature, internal pressure, cladding stress, cladding strain and fatigue usage. The design bases consider factors influencing irradiation behavior such as pellet density, fission product gas release, cladding creep, oxidation and other physical phenomena. Thermal-hydraulic design bases described in chapter 1.2.1.5.1.3 are also applied to prevent fuel rod failure.

The design bases for the US-APWR fuel assembly consider functional requirements for the fuel assembly and provide limiting loads and/or stresses on the fuel assembly components. The loads and stresses are those due to normal operation, AOOs and postulated accidents, in addition to non-operational condition such as shipping and handling.

For the purpose of safe shutdown and adequate reactivity control of the reactor, design bases for in-core control components are established in terms of thermal physical properties of the absorber material, compatibility of the absorber and cladding material, cladding stress-strain limits, and irradiation behavior of absorber material.

The US-APWR fuel assembly consists of the 264 fuel rods arranged in a square 17x17 array, together with 24 control rod guide thimbles, an in-core instrumentation guide tube, 11 grid spacers, and top and bottom nozzles. The fuel rods consists of ZIRLO™[1] cladding tube loaded with sintered uranium dioxide pellets slightly enriched up to 5 wt% and/or gadolinia-uranium dioxide pellets blended with maximum 10 wt% content of Gd2O3, coil spring at the upper plenum, lower plenum spacer, and end plugs welded at the top and bottom ends to seal pressurized helium gas within the rod.

The skeleton structure of the assembly consists of top/bottom nozzles, grid spacers, control rod guide thimbles and an in-core instrumentation guide tube. The fuel rods are positioned by 11 grid spacers. The grid spacers are mechanically fixed to the 24 control rod guide thimbles. The control rod guide thimbles are symmetrically arrayed according to the arrangement of the control rods in a rod cluster control assembly. The in-core instrumentation guide tube is located at the center of the square array of fuel rods.

The grid spacers consist of a square lattice structure interlocked by thin straps. For the US-APWR fuel assembly, the intermediate grid spacers are made of Zircaloy-4, improving neutron economy, while the top and bottom grid spacers are made of Inconel 718.

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[1] ZIRLO™ is a registered trademark of the Westinghouse Electric Corporation.

The top nozzle assembly has holddown springs made of Inconel 718 in order to prevent the fuel assembly from liftoff due to the hydraulic force during normal operation and AOOs except for pump-over-speed event. The top nozzle has joints to the control rod guide thimbles which enables re-construction for replacing the fuel rod in case of leakage.

The bottom nozzle has a plate on which thin plates are placed in grooved slits and welded, providing a filter for capturing debris coming into the flow holes of the bottom nozzle.

The control rod guide thimbles are made of Zircaloy-4 and are fixed to the grid spacers and the bottom nozzle. They guide in-core control components such as control rods, burnable absorber rods and neutron source rods into the fuel assembly. The lower part of the control rod guide thimble is small in diameter, to provide a buffer effect at the end of rod cluster control assembly (RCCA) drop.

The in-core instrumentation guide tube is also made of Zircaloy-4, with the ends inserted into the top and the bottom nozzles. The tube leads an in-core neutron detector into the fuel assembly through a center hole of the adapter plate in the top nozzle.

The RCCA consists of the spider, which arranges and fixes the control rods in symmetrical positions, and 24 control rods whose neutron absorber material is 80% silver, 15% indium and 5% cadmium clad with type 304 stainless steel tube.

The burnable absorber assembly consists of a holddown assembly and burnable absorber rods in which borosilicate glass is clad with type 304 stainless steel tube. A maximum of 24 absorber rods are attached to a holddown assembly and are inserted into a control rod guide thimbles of a fuel assembly.

The primary and secondary neutron source assembly consists of one or several neutron source rods, the thimble plug rods and the same holddown assembly as the burnable absorber assembly. The primary neutron source assembly is used to supply neutrons from californium for the initial start up. The secondary neutron source, which consists of mixed 50% antimony and 50% beryllium, becomes radioactive during reactor operation and functions as a neutron supplier. The secondary neutron source assembly is used instead of the primary source assembly after the first irradiation.

1.2.1.5.1.2 Nuclear Design

The US-APWR core consists of 257 mechanically identical fuel assemblies surrounded by a stainless steel neutron reflector. The US-APWR active fuel length is increased to approximately 14 ft. Since the US-APWR has the same thermal power as its predecessor, it has lower linear power density of 4.65 kW/ft, allowing flexible core and fuel management with improved thermal margins. Even under the constraints of fuel enrichment less than 5 wt% and maximum fuel rod burnup of 62,000MWD/MTU, 24-month cycles with a 2-batch reload are feasible in the US-APWR.

The core power distribution is periodically monitored by using movable in-core neutron detectors and constantly surveyed by fixed neutron ex-core detectors. The US-APWR employs a top-mounted in-core nuclear instrumentation system in order to improve the reliability of the reactor vessel. Detectors are inserted through the in-core instrumentation guide tube to monitor the entire fuel assembly's active length. The strategically located in-core detector positions provide sufficient information to reconstruct detailed three-dimensional power distributions. The ex-core detectors provide on-line global axial and radial power distribution data and power changes, and provide input to automatic control functions. Thermocouples at the outlets of a subset of fuel assemblies also provide core performance data.

The core is designed to have negative reactivity feedback characteristics associated with fuel temperature and moderator temperature or density. With these characteristics, power oscillations can be easily brought under control. Even a fast reactivity rise in the accident condition is immediately controlled by negative Doppler effect.

Control rods and soluble boron in the coolant are provided as two independent shutdown mechanisms and are designed to control the reactivity during reactor operation. The control rod system has enough reactivity to compensate for fast reactivity fluctuation during operation and the transition from full power to the hot zero power condition. In addition, the shutdown margin with the most reactive control rod stuck gives adequate subcriticality to minimize any consequences of over-cooling events. In order to guarantee the shutdown margin, the control rod banks use insertion limits during operation. During normal operation, selected groups of RCCAs are maneuvered automatically or manually to control reactor power to the load demand. When the reactor is tripped, all RCCAs are inserted into the fuel assemblies by gravity.

Slow reactivity changes, such as fuel burnup and the transition from hot shutdown to cold shutdown, are compensated with soluble boron in the reactor coolant system. The negative reactivity insertion by soluble boron is rapid enough to overcome the reactivity rise due to the decay of built-up xenon. In addition, the boron concentration is controlled to maintain subcriticality during refueling.

The concentration of boron in the RCS is adjusted through the operation of the CVCS. When increasing the boron concentration, the necessary amount of concentrated boric acid solution is injected into the RCS. When decreasing the boron concentration, pure water is added to the RCS to dilute the coolant water to the required boron concentration. The boron concentration control method allows reactor operation with the minimum insertion of RCCAs, to assure that the power distribution is not excessively distorted.

1.2.1.5.1.3 Thermal-Hydraulic Design

The reactor core and associated coolant, control, and protection systems are designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. Since fuel cladding is one of the three fission product barriers, its integrity is maintained during normal plant operation and AOOs to contain the fission products.

The US-APWR core thermal-hydraulic design assures adequate cooling for the reactor core during normal operation and AOO conditions and keeps fuel integrity in compliance with the design requirements under the various conditions.

There is at least a 95-percent probability at a 95-percent confidence level that the hot fuel rod in the core does not experience a Departure from Nucleate Boiling (DNB) phenomenon and that the fuel rod with the most limiting linear heat rate (kW/ft) shall not cause the fuel pellet to melt during normal operation and AOOs.

The core thermal-hydraulic design based on the assumption of conservative core RCS flow and core bypass flow provides sufficient core cooling and assures core safety.

1.2.1.5.1.4 Reactor Internal Design

The core support structures provide support and align the core. The reactor internals direct the amount of coolant flow and its distribution within the reactor vessel. The upper reactor internals consist of the upper core support, upper core plate, upper support columns, and RCCA guide tubes. The lower reactor internals consist of the core barrel, the lower core support plate, the neutron reflector, and the secondary core support assembly. The lower core support plate is welded to the bottom of the core barrel, and supports all the fuel assemblies, the neutron reflector, the diffuser plate and the energy absorber. The design of the US-APWR reactor internals allows the use of a standard APWR vessel height, even though the fuel length has been increased from 12 ft. to 14 ft.

1.2.1.5.2 Reactor Coolant System

1.2.1.5.2.1 Reactor Coolant System Boundary and Connected Systems

The RCS provides reactor cooling and energy transport functions. The RCS consists of the reactor vessel, steam generators, pressurizer, reactor coolant pipes, reactor coolant pumps, and valves. The RCS, including connections to related auxiliary systems, constitutes the reactor coolant pressure boundary.

The RCS has the following features. It:

- Circulates the reactor coolant through the reactor core and transfers heat to the secondary system via the steam generators.
- Cools the core sufficiently to prevent core damage during reactor operation.
- Forms the reactor coolant pressure boundary, which functions as a barrier to prevent radioactive materials in the reactor coolant from being released to the environment.
- Functions as a neutron moderator and reflector, and as a solvent for boron.
- Controls the reactor coolant pressure.

The RCS comprises the major portion of the reactor coolant pressure boundary and has great importance to the safe operation of the US-APWR in preventing accidents and controlling their consequences. A high degree of attention is paid to its design, material selection, and quality control to satisfy the following:

- The RCS is designed to provide core cooling during normal operation, transients, and accident conditions.
- The materials of the reactor vessel, SGs, pressurizer, reactor coolant pumps, piping, valves, and other components that contact the reactor coolant are selected to maximize corrosion resistance.
- Reactor coolant pressure boundary components are designed, fabricated, and tested according to the requirements of 10CFR50, 50.55a, GDC 1 and ASME code, Section III.
- Components that form the reactor coolant pressure boundary are designed and operated to prevent nil-ductility fracture. Special attention is paid to material selection, design, manufacture, and operation of ferritic steel components. This is done to assure that, under normal operation, transients, maintenance, testing, and accident conditions, the ferritic steel components behave in a non-brittle manner and the probability of rapidly propagating fracture is minimized. The operation of the RCS during startup and shutdown is controlled in accordance with heating and cooling limits that consider fast-neutron irradiation effects throughout the lifetime of the plant.
- Seismic Category I design is applied to the RCS and its supporting structures.
- The design and arrangement of supporting structures, including concrete, are such that functions important to safety are not impaired by the impact resulting from a postulated rupture of the piping, which forms the reactor coolant pressure boundary. Pipe whip restraints are installed where necessary.
- Reactor coolant pressure boundary components are designed using conservative assumptions about future plant operating conditions. These include transient conditions such as variations in temperature and pressure, and conservative estimates of the number of cycles for each transient.
- A leak monitoring system is used to provide early detection of leakage from the reactor coolant pressure boundary.
- The systems and components that form the reactor coolant pressure boundary are designed to allow periodic in-service inspections in accordance with ASME Code, Section XI.

The RCS is designed so that the system pressure can be maintained at less than 1.1 times the design pressure by the pressure relief system. The pressure relief system has the following design features:

- Spring-loaded safety valves are installed on the top of the pressurizer.
- An additional relief line has motor-operated relief valves for safety depressurization valves (SDVs). The valves are arranged in parallel and are

driven by motor operators. A remotely controlled, motor-operated isolation valve is installed upstream of each SDV to allow isolation in the event of a leak.

- Pressurizer steam is discharged to pressurizer relief tank inside the containment.
- Relief valves are installed in each residual heat removal system (RHRS) to provide over-pressurization protection for unacceptable combinations of high RCS pressure and low RCS temperature.

The reactor coolant piping consists of the pipes connecting the reactor vessel, steam generators, reactor coolant pumps, and pressurizer, together with the various branches of the main piping up to the appropriate isolating valve. It also includes instrumentation connections to the RCS that provide for flow, temperature, and pressure. The reactor coolant pipes and fittings are made of austenitic stainless steel. Pipes and fittings are seamless and comply with the requirements of the ASME Code, Section II (Parts A and C), Section III and Section IX. All smaller piping that is part of the RCS, such as the pressurizer surge line, spray line, loop drains, and connecting lines to other systems, are also made of austenitic stainless steel. The reactor coolant piping is designed using the LBB concept.

The residual heat removal function is accomplished by the residual heat removal system (RHRS), consisting of four independent trains. Each train has one containment spray/residual heat removal (CS/RHR) heat exchanger (HX), one CS/RHR pump, and connecting piping and valves.

The RHRS has the following functions:

- It removes reactor core decay heat and other residual heat from the reactor coolant.
- It transfers refueling water between the reactor cavity and the refueling water storage pit at the beginning and end of refueling operations.

The RHRS design is based on the following:

- The RHRS is designed to cool the reactor by removing decay heat and residual heat from the RCS after the initial phase of cooldown.
- The RHRS is designed with four independent subsystems.
- The CS/RHRS pumps receive power from safety electrical buses so that the system functions are maintained during a LOOP.
- The RHRS is designed to provide the ability to reduce the reactor coolant temperature with only two of the four subsystems operating.
- The RHRS is designed to transfer boric acid water from the refueling water storage pit to the refueling cavity at the beginning of a refueling operation. After

refueling, the reactor cavity is drained by pumping the water back to the refueling water storage pit or allowing it to return by gravity.

The residual heat removal function is placed in operation when the pressure and temperature of the RCS are approximately 400 psi and 350°F respectively. During system operation, each CS/RHRS pump takes suction from one of the RCS hot legs by a separate suction line. The pumps discharge through the CS/RHRS HXs, which transfer heat from the reactor coolant to the component cooling water system (CCWS). The reactor coolant is returned to the four RCS cold legs.

1.2.1.5.2.2 Reactor Vessel

The pressure boundary portions of the RCS components are designed to satisfy ASME Code Section III requirements.

The reactor vessel (RV) contains the fuel assemblies and reactor vessel internal core, including the core support structures, control rods, neutron reflector, and other structures associated with the core.

The RV consists of four inlet nozzles, four outlet nozzles, and four direct vessel injection nozzles, which are located between the upper reactor vessel flange and the top of the core, so as to be able to maintain coolant in the reactor vessel in the case of leakage in the reactor coolant loop. Reactor coolant enters the vessel through the inlet nozzles, flows down the annulus between the core barrel and RV wall, turns at the bottom of the vessel, and flows upwards through the core to the outlet nozzles.

Sealing between the closure head flange and RV vessel flange shell is by two metallic O-rings. Seal leakage is detected by means of two monitoring tubes in the vessel flange shell, one located between the inner and outer O-rings, and one located outside the outer O-ring. Leakage is indicated by a high temperature alarm signal.

The RV closure head consists of a hemispherical dome and a closure head flange. The RV closure head is equipped with control rod drive mechanism in-core instrumentation, thermocouple and reactor vessel level instrumentation system nozzles.

The main cylindrical shell of the RV consists of an upper and lower shell.

The bottom head consists of a transition ring and bottom hemispherical dome. The bottom dome does not have any penetrations.

Encapsulated test specimens are inserted between the core barrel and the reactor vessel. After being irradiated the test specimens are withdrawn at appropriate periods and destructively tested to monitor changes in material characteristics during the service periods.

Where irradiation by fast neutron can be relatively high in the RV, the vessel wall is designed so that there are no shape discontinuities that could cause stress concentrations.

1.2.1.5.2.3 Reactor Coolant Pumps

The reactor coolant pump (RCP) is located in the reactor containment and assures adequate reactor cooling flow rate to maintain a departure from nucleate boiling ratio (DNBR) greater than the limit that is evaluated in the safety analysis.

In the event of LOOP, the RCP is able to provide adequate flow rate during coastdown conditions because of the pump assembly's rotational inertia.

The RCP is a vertical shaft, single-stage, mixed flow pump with diffuser.

Leakage along the RCP shaft is normally controlled by three shaft seals, arranged in series so that any reactor coolant leakage to the containment is essentially zero.

The pump shaft, seal housing, thermal barrier, main flange and impeller of the RCP can be removed from the casing as a unit without disturbing the reactor coolant piping.

1.2.1.5.2.4 Steam Generators

The SGs are vertical shell U-tube evaporators with integral moisture separating equipment. Reactor coolant enters the channel head via the coolant inlet nozzle, flows through the inverted U-tubes, transferring heat from the primary side to the secondary side, and leaves from the channel head via the coolant outlet nozzle. The channel head is divided into a hot leg side and a cold leg side by a vertical divider plate that is welded to the channel head and tubesheet. The tube material is Alloy 690, thermally treated. The material of the tubesheet and the channel head is low alloy steel. The cladding on the primary side of the tubesheet is Ni-Cr-Fe alloy, and the cladding on the channel head is stainless steel.

Steam generated on the shell side (secondary side) flows upward and exits through the outlet nozzle at the top of the vessel. Feedwater enters the steam generator at an elevation above the top of the U-tubes through a feedwater nozzle. The feedwater enters a feedwater ring and is distributed through the perforated nozzles attached to the top of the feedwater ring. The material of the perforated nozzles and feedwater ring is low alloy steel that is resistant to erosion and corrosion for the expected secondary water chemistry and flow rate through the nozzles and the feedwater ring. After exiting the perforated nozzles, the feedwater mixes with saturated water removed by the moisture separators. The flow then enters the downcomer annulus between the wrapper and the shell.

The tubes are hydraulically expanded to the full depth of the tubesheet at each end and supported by broached tube support plates. In the U-bend region, tubes are supported by anti-vibration bars. When the water passes the tube bundle, it is converted to a steam-water mixture. The steam-water mixture from the tube bundle then rises into the primary separators and the secondary separators to remove water from steam water mixture. The dry steam exits from the steam generator through the outlet nozzle. This nozzle is equipped with a flow restrictor that controls the rate of energy release during main steam line break event.

1.2.1.5.2.5 Pressurizer

The pressurizer provides a point in the RCS where liquid and vapor can be maintained in equilibrium under saturated conditions for pressure control. The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads. It is constructed of low-alloy steel with austenitic stainless steel cladding on all surfaces exposed to the reactor coolant. Electrical immersion heaters are installed vertically through the bottom head of the vessel while the spray nozzle, safety depressurization valve and safety valve connections are located in the top head of the vessel. A manway is also provided in the top head for access to the internal space for inspections and maintenance of the spray nozzle. The manway cover is provided with a gasket and secured with threaded fasteners.

The pressurizer is designed to accommodate positive and negative volume surges caused by load transients. The surge line, which is attached to the bottom of the pressurizer, connects to the hot leg of a reactor coolant loop. A screen above the surge line is provided to prevent passage of foreign particles from the pressurizer to the RCS. Guide plates in the lower section of the pressurizer prevent in-surge of cold water from flowing directly to the steam/water interface and assist in mixing. The guide plates also provide support to limit vibration of the heaters. The pressurizer is supported by a skirt welded to the bottom head.

Each pressurizer spray line is provided with a separate, automatically controlled, air-operated spray valve with manual override, and a spray block valve. A manual throttle valve is provided in parallel with each spray control valve. This throttle valve enables a small continuous flow to be maintained in the spray lines when the spray valves are closed. An auxiliary spray line is provided from the CVCS to assure that the pressurizer spray is available to permit reactor cooldown when the RCPs are unavailable.

Spring-loaded SRVs are positioned on separate relief lines from the pressurizer. Another relief line incorporates motor-operated SDVs arranged in parallel. These valves are driven by the motor operators. All relief lines run into spargers, which carry pressurizer steam discharge to pressurizer relief tank inside the containment. Remotely controlled, motor-operated isolation valves are provided upstream of each SDV to allow isolation of a leaking SDV. The SRVs provide overpressure protection of the RCS. The spray valves limit RCS pressure rises following less severe transients to prevent undesirable opening of the pressurizer SRVs. Other SRVs in the RHRS provide cold overpressurization protection against unacceptable combinations of high RCS pressure and low RCS temperature.

1.2.1.5.2.6 Component Supports

Reactor Vessel Supports - The RV is supported by eight steel support pads, which are integrated with the inlet and outlet nozzle forgings. The support pads are placed on support brackets, which are supported by the steel structure around the RV. Radial movement, which results from vessel thermal expansion and contraction, is accommodated by sliding surfaces between the shim plates and the support pads while the horizontal load in an earthquake is supported by the support brackets and the base plate, so that the center position of the vessel always remains unchanged. The support brackets form a box-shaped structure and are air-cooled by the reactor vessel compartment cooling fans, in order to minimize heat transfer from the reactor vessel to the concrete support portions through the support brackets.

Steam Generator Supports - The SGs are supported by an upper lateral support structure, a middle lateral support structure, a lower lateral support structure, and support columns. The upper and middle lateral support structures support the SG by using snubbers. The lower lateral support structure is made of steel. The support structures for the upper shell, middle shell and lower shell are designed to accommodate thermal expansion of piping. At the same time, they can restrain the horizontal movement of the SG in the event of an earthquake or accident. The support columns support vertical loads, and the upper and lower ends of the support pipe are pin-jointed, so as not to restrain the movement of the SGs caused by thermal expansion of piping.

Reactor Coolant Pump Supports - The RCP is supported by lateral support structure and support columns. The support structure is made of steel. The lateral support structure is designed to accommodate thermal expansion of piping. At the same time, it can restrain the horizontal movement of the RCP in the event of an earthquake or accident. The support columns support vertical loads, and the support pipe upper and lower ends are pin-jointed in the same manner as the SG so as not to restrain the movement of the RCP caused by thermal expansion of the piping.

Pressurizer Supports - The pressurizer is supported by an upper support structure and lower support skirt. The steel upper support structure restrains horizontal movement of the pressurizer, while the lower support structure restrains vertical loads using a skirt welded to the bottom shell of the pressurizer. The upper support structure does not restrain the movement of the pressurizer caused by thermal expansion, but restrains horizontal movements in the event of earthquake or accident.

1.2.1.5.3 Steam and Power Conversion System Design

The steam and power conversion system consists of the turbine generator (T/G), main steam supply system (MSS), condensate and feedwater system (CFS), emergency feedwater system (EFWS), turbine bypass system (TBS), steam generator blowdown system (SGBDS), and other systems.

The steam and power conversion system is designed to remove heat energy from the reactor coolant system via the four steam generators and to convert it to electrical power in the turbine generator. The main condenser removes air and other non condensibles from the condensate and transfers heat to the circulating water system (CWS). The deaerator additionally deaerates the condensate, and supplies deaerated water to the regenerative feedwater cycle. The regenerative turbine cycle heats the feedwater, and the main feedwater system returns it to the steam generators.

The steam generated in the four steam generators are supplied to the high-pressure turbine by the MSS. After expansion through the high-pressure turbine, the steam passes through the two moisture separator reheaters (MS/Rs) and is then admitted to the three low-pressure turbines. A portion of the steam is extracted from the high and low pressure turbines for seven stages of feedwater heating.

Exhaust steam from the low-pressure turbines is condensed and deaerated in the main condenser. The heat exhausted in the main condenser is removed by the CWS. The condensate pumps take suction from the condenser hotwell and deliver the condensate through four stages of low pressure closed feedwater heaters to the fifth stage, open deaerating heater. Condensate then flows to the suction of the steam generator feedwater booster pump and is discharged to the suction of the main feedwater pump. The steam generator feedwater pumps discharge the feedwater through two stages of high pressure feedwater heaters to the four steam generators.

The moisture separator drainage is sent to the deaerator. The reheater drainage is sent to the high pressure feedwater heaters, and the high pressure feedwater heater(s) drainage is cascaded into the deaerator. Drainage from the low pressure feedwater heaters is cascaded through successively lower pressure feedwater heaters to the heater drain tank and pumped by the Heater Drain Pump(s) to the piping between the low pressure heater no. 1 and 2.

The turbine-generator has an output ranging from 1600 MWe to 1700 MWe depending on plant conditions. The nuclear steam supply system (NSSS) has a thermal output of 4466 MWt.

1.2.1.5.3.1 Turbine Generator

The turbine generator (T/G) is a power conversion system designed to change the thermal energy of the steam flowing through the turbine into rotational mechanical work, which rotates the generator to generate electrical power.

The T/G is designed based on the following:

- The T/G does not perform or support any safety-related function and therefore has no nuclear safety design basis.

- The T/G is designed, manufactured, and inspected, including the measures for prevention of turbine failures, under proper quality control measures.

The T/G is designed with consideration of the following items, so that safe operation can be achieved by various protective, monitoring, and control devices:

- Prevention of vibration of the T/G shaft. If vibration does occur, the T/G is designed so that alarms are raised by vibration monitoring devices.

- Steam valves, governor, etc. are designed with redundancy so that over-speed of the T/G does not exceed the design value.

The T/G is designed to trip automatically in the event of anticipated abnormal operating conditions. Measures are taken to protect the T/G from the occurrence of turbine missiles. The piping, bearings, etc. are designed to preclude leakage of turbine lubricating oil. A fire-fighting system is provided to account for the unlikely event of a fire in this area. The T/G is designed to accommodate periodic operational tests of the valves essential for overspeed protection and other protective devices.

The T/G consists of the turbine, generator, external moisture separator/reheaters, steam valves, and their auxiliary systems.

The turbine is of a tandem compound type, 1,800-rpm machine and consists of a double-flow high-pressure turbine and three double-flow low-pressure turbines. A one-piece low-pressure turbine rotor is used to improve the resistance against stress corrosion cracking and corrosion fatigue. The generator is a four-pole water-cooled type and is directly coupled to the turbine. Two moisture separator reheaters are located between the high-pressure turbine and low-pressure turbines to improve thermal efficiency.

Steam flow from the SGs is introduced to the two floor-mounted steam chests and then to the high-pressure turbine. Each steam chest contains two main steam stop valves and two control valves, which control steam flow. After leaving the high-pressure turbine, the steam flow is led to three low-pressure turbines through six pairs of reheat stop valves and intercept valves.

1.2.1.5.3.2 Main Steam Supply System

The main steam supply system (MSS) runs from the steam generator nozzle up to the main turbine stop valve, including the branch piping.

The main function of the MSS is to transport steam from the steam generators to the high-pressure turbine and to the moisture separator reheater over a range of flows and pressures covering the entire operating range from system warmup to valve wide open (VWO) turbine conditions.

The system also supplies steam to the main turbine gland seal system, the emergency feedwater pump turbine(s), deaerator heater and auxiliary steam supply system (ASSS). The system also dissipates heat generated by the nuclear steam supply system (NSSS) by means of turbine bypass valves (TBV) to the condenser or to the atmosphere through air operated main steam relief valves (MSRV) or the motor operated main steam depressurization valves (MSDV), or the spring-loaded main steam safety valves (MSSV) when either the turbine-generator or condenser is unavailable.

1.2.1.5.3.3 Condensate and Feedwater System

The condensate and feedwater system (CFS) provides feedwater at the required temperature, pressure, and flow rate to the steam generators. The condensate system (CDS) runs from the condenser hotwell outlet to the deaerator; and the feedwater system (FWS) runs from the outlet of the deaerator to the steam generator nozzles. Condensate is pumped from the main condenser hotwell by the condensate pumps, passes through the condensate polishing system (CPS), gland steam condenser, and low-pressure feedwater heaters to the deaerator. The feedwater booster/main feedwater pumps take suction from the deaerator, and then pumps the feedwater through the high-pressure feedwater heaters to the steam generators.

The CFS provides condensate cleanup capability and maintains condensate quality through dearation and interfacing with the main condenser, CPS, secondary system chemical injection system (CIS), secondary sampling system (SSS), and deaerator.

1.2.1.5.3.4 Emergency Feedwater System

The emergency feedwater system (EFWS) consists of two motor-driven pumps, two steam turbine-driven pumps, two emergency feedwater pits, and associated piping and valves. The EFWS removes reactor decay heat and RCS residual heat through the SGs following transient conditions or postulated accidents such as reactor trip, loss of main feedwater, steam or feedwater line breaks, SG tube rupture, and unavailability of the FWS.

The EFWS satisfies the following design requirements:

- The EFWS maintains the capability of the SGs to remove reactor decay heat and other RCS residual heat by converting the feedwater to steam, which is then discharged to the atmosphere by using the main steam depressurization valve.
- The EFWS satisfies the requirement that the pumps be powered by diverse power sources.
- The EFWS can perform all its required safety-related functions assuming a single failure in one train and a second train out of service for maintenance.
- The EFWS is automatically initiated by an SG water level low signal.

The four pump configuration and the cross connected discharge header allow the system to meet the single failure criterion with one pump out of service for maintenance. Both motor-driven and turbine-driven pumps are used to satisfy the requirement that the pumps be powered by diverse power sources. Turbine-driven pumps are available at SBO condition.

The four emergency feedwater pumps take suction from two emergency feedwater pits. Each pump is provided with a recirculation line, including a minimum flow line and a full flow line, leading back to the emergency feedwater pit. The minimum flow line assures a minimum recirculation flow whenever the pumps are operating.

1.2.1.5.3.5 Steam Generator Blowdown System

The steam generator blowdown system (SGBDS) assists in maintaining secondary side water chemistry within acceptable limits. The SGBDS consists of a flash tank, regenerative heat exchangers, non regenerative coolers, filters, demineralizers, piping, valves and instrumentation. The system includes blowdown sample coolers, pressure reducing valves, a radioactive process monitor, instruments, piping and valves.

The SGBDS has the following functions:

- Maintain acceptable secondary coolant water chemistry with monitoring through use of the blowdown sampling system

- Sample blowdown water for chemistry and detect primary-to-secondary leakage with the SG blowdown water radiation monitor

The blowdown water flows to the flash tank, and flows through regenerative heat exchangers and non regenerative coolers. The blowdown water from the coolers flows to the filter and demineralizers. The purified water from the demineralizers flows to the condenser. Blowdown water is sent to the waste water facility or liquid waste management system when disposal is required instead of recovery. The normal blowdown flow rate varies from approximately 0.5 percent to 1.0 percent of MSR.

The blowdown samples are used to check the water chemistry of the blowdown water and to detect leakage or failure of a steam generator tube by radiation monitoring.

The SGBDS is automatically isolated from the steam generator by closing the isolation valves in the event of an abnormal condition.

1.2.1.5.4 Auxiliary Fluid and Mechanical Systems Design

1.2.1.5.4.1 Engineered Safety Systems Design

Engineered safety features (ESFs) reduce the consequences of postulated accidents. Further, ESFs protect the public health and safety in the unlikely event of an accidental release of radioactive fission products from the RCS.

The ESFs consist of the prestressed concrete containment vessel (PCCV), containment spray system (CSS), containment isolation system, containment hydrogen monitoring and control system, emergency core cooling system (ECCS), main control room (MCR) HVAC system, and annulus emergency exhaust system.

Prestressed Concrete Containment Vessel (PCCV) - The PCCV is designed to completely enclose the reactor and RCS and assure that essentially no leakage of radioactive materials to the environment would result even if a major failure of the RCS were to occur. The PCCV consists of a prestressed, post-tensioned concrete structure with a cylindrical wall, a hemispherical dome, and a flat reinforced-concrete basement. The inside surface of the PCCV structure is lined with carbon steel.

The design pressure and temperature of the PCCV are defined by the following postulated accidents:

- Loss-of-coolant accident (LOCA)

- Main steam line break (MSLB)

The PCCV is designed to contain the energy and radioactive materials that result from a postulated LOCA, and for 68 psig internal pressure to assure a high degree of leak tightness during normal operation and under accident conditions. An internal polar crane is supported by the PCCV. A continuous crane girder transfers the polar crane loads to the C/V wall. Hydrogen igniters are provided for the mitigation of combustible gas generated by a beyond design basis accident.

Containment Spray System (CSS) - The CSS consists of four independent trains, each containing a CS/RHR HX, a CS/RHR pump, spray nozzles, piping and valves. The CS/RHR HXs and the CS/RHR pumps are used for both CSS and RHRS function. The CSS sprays boric acid water into the PCCV in the event of a LOCA.

In the unlikely event of a design basis LOCA or secondary system piping failure, the CSS is designed to limit and control the containment pressure, such that:

- The peak containment accident pressure is well below the containment design pressure
- The containment pressure is reduced to less than 50 percent of the peak calculated pressure for the design basis LOCA within 24 hours after the postulated accident.

The CSS also removes particulate iodine.

The CSS satisfies the following design requirements:

- The CSS includes four 50% capacity CS/RHR pump trains and assumes one is out of service for maintenance and one becomes inoperative due to a single failure upon the initiation of the CSS.
- The emergency power sources supply electrical power to the essential components of the CSS, so that safety functions can be maintained during a LOOP.
- The CSS is automatically initiated by a containment spray signal.
- The CSS design permits periodical tests and inspections to verify integrity and operability.

The CSS includes four CS/RHR pumps and four CS/RHR heat exchangers (HXs), piping, spray nozzles and valves. The CSS is automatically actuated on receipt of a containment spray signal. When the signal is received, the CS/RHR HX outlet valves open and the CS/RHR pumps start. The CS/RHR pump motor is connected to a safety bus, so the emergency power sources can supply electrical power in case of a LOOP. The CS/RHR pumps take suction from the refueling water storage pit, and the stop valve on the inlet line is always open during reactor operation. The water in the pit is cooled by the CS/RHR HXs and is delivered to the spray headers located in the top of the PCCV. The refueling water storage pit (RWSP) in the containment provides a continuous suction source for the CS/RHR pumps, thus eliminating the conventional realignment from the refueling water storage pit to the recirculation sump. The CSS has sufficient redundancy to perform its required safety functions following an accident assuming a single failure in one train, with a second train out of service for maintenance.

Containment Isolation System - The lines that penetrate the PCCV are provided with containment isolation valves. Lines that are part of the reactor coolant pressure boundary or connected directly to the containment atmosphere are provided with containment isolation valves inside and outside containment. Lines that are neither part of the reactor pressure boundary nor connected directly to the containment atmosphere are provided with containment isolation valves outside containment. The containment isolation valves whose safe failure position is closed are designed not to fail open upon loss of actuating power after closing. In addition, the containment isolation valves that close automatically upon receiving an isolation signal are designed not to open automatically if the isolation signal is removed. Containment isolation valves are designed to be tested for both function and leakage.

Emergency Core Cooling System (ECCS) - The ECCS includes the accumulator system, high-head injection system and emergency letdown system. The ECCS injects boric acid water into the reactor coolant system following a postulated accident and performs the following functions:

- Following a LOCA, the ECCS removes stored and fission product decay heat from the reactor core.
- Following a MSLB, the ECCS provides negative reactivity to shut down the reactor.
- In the event that the normal CVCS letdown and boration capability is lost, the ECCS provides emergency letdown and boration of the RCS.
- Following a LOCA, the ECCS provides adjustment of the pH of the water in the containment .

The ECCS design is based on the following:

- In combination with control rod insertion, the ECCS is designed to shut down and cool the reactor during the following accidents.
  - LOCA
  - Ejection of a control rod cluster assembly
  - Secondary steam system piping failure
  - Steam generator tube failure
- The ECCS includes four 50% capacity SI pump trains and assumes one is out of service for maintenance and one becomes inoperative due to a single failure upon the initiation of the ECCS.
- The emergency power sources supply electrical power to the essential components of the ECCS, so the safety functions can be maintained during a LOOP.

- ECCS is automatically initiated by a safety injection signal.
- ECCS design permits periodical tests and inspections to verify integrity and operability.

The accumulator system stores boric acid water under pressure and automatically injects it if the reactor coolant pressure decreases significantly. The accumulator system consists of four accumulators and the associated valves and piping, one for each RCS loop. The system is connected to the cold legs of the reactor coolant piping and injects boric acid water when RCS pressure falls below the accumulator operating pressure. The system operates passively; pressurized nitrogen gas forces boric acid water from the tanks into the RCS.

The accumulators incorporate internal passive flow dampers, which function to inject a large flow to refill the reactor vessel in the first stage of injection, and then reduce the flow as the accumulator water level drops. When the water level is above the top of the standpipe, water enters the flow damper through both inlets at the top of the standpipe and at the side of the flow damper, and injects water at a large flow rate. When the water level drops below the top of the standpipe, the water enters the flow damper only through the side inlet, and injects water at a small flow rate.

The accumulators perform the large flow injection to refill the reactor vessel and the following small flow injection during core reflooding in association with the safety injection pumps. The combined performance of the accumulator system and the high head injection system eliminates need for a conventional low head injection system.

The high head injection system consists of four independent trains, each containing a safety injection pump, associated valves and piping. The safety injection pumps start automatically upon receipt of the safety injection signal. One of four independent safety electrical buses is available to each safety injection pump.

The safety injection pumps are aligned to take suction from the refueling water storage pit and to deliver boric acid water to the direct vessel injection nozzles on the reactor vessel. Two safety injection trains are capable of meeting the design cooling function for a large LOCA, assuming a single failure in one train and another train out of service for maintenance.

The RWSP in the containment provides a continuous boric acid water source for the safety injection pumps thus eliminating the conventional realignment from the refueling water storage tank to the containment recirculation sump.

The emergency letdown system consists of two emergency letdown lines from the RCS hot legs to the RWSP. In the event that the normal CVCS letdown and boration capability is not available, the feed and bleed emergency letdown and boration operation can be utilized to achieve a cold shutdown boration level in the reactor coolant prior to the safe shutdown operation. The emergency letdown directs reactor coolant to the RWSP. The safety injection pumps provide boric acid coolant to the RCS from the RWSP.

Sodium tetraborate decahydrate (NaTB) contained in NaTB baskets provides adjustment of the pH of the water in the containment following an accident.Twenty three NaTB baskets containing NaTB as a buffer agent are located inside three NaTB basket containers at an elevation that is below the lowest spray ring. NaTB in basks is dissolved in spray water in the containers. The solution containing NaTB is discharged from each container to the RWSP through NaTB solution transfer pipe.

Main Control Room (MCR) HVAC System - The MCR HVAC system is designed to provide proper environmental conditions for the control room envelope (CRE) during normal operation and accident conditions. The MCR HVAC system is also designed to protect operators against a postulated release of radioactive material and toxic gases.

The MCR HVAC system design is based on the followings:

- MCR HVAC system maintains proper environmental conditions for CRE in the normal operation mode and emergency modes.
- MCR HVAC system has two emergency modes; pressurization mode and isolation mode.
- Pressurization mode protects the MCR operators and staff within the CRE during the accident conditions.
- Isolation mode protects the MCR operators and staff within the CRE from external toxic gas or smoke.
- MCR HVAC system is powered from Class 1E busses so the safety functions can be maintained during a LOOP.

Annulus Emergency Exhaust System (AEES) - The AEES is designed for fission product removal and retention by filtering the air it exhausts from the annulus and safeguard component area following accidents.

The AEES satisfies the following design requirements:

- The AEES establishes and maintains a negative pressure in the annulus and safeguards component area relative to adjacent areas.
- The AEES removes and retains fission products by High-Efficiency Particulate Air (HEPA) filters and discharges exhaust air through the vent stack.
- The AEES is powered from Class 1E busses so its safety functions can be maintained during a LOOP.
- The AEES is automatically initiated by ECCS actuation signal.

1.2.1.5.4.2 Spent Fuel Pit Cooling and Purification System

The spent fuel pit (SFP) cooling and purification system (SFPCS) is a closed circuit system that includes the spent fuel pit coolers, spent fuel pit pumps, spent fuel pit demineralizers, spent fuel pit filters, piping, and valves.

The SFPCS has the following functions:

- Removal of decay heat from spent fuel in the SFP
- Purification of the boric acid water in the SFP, refueling water storage pit (RWSP), refueling water storage auxiliary tank (RWSAT), and reactor cavity.

The SFPCS design is based on the following:

- The SFPCS removes decay heat released by spent fuel stored in the SFP by cooling the SFP water.
- Demineralizers and filters remove particulate and ionic impurities from the SFP water.
- The emergency power sources can supply electrical power to the SFP pumps, so that SFP cooling functions can be maintained during a LOOP.
- Water can be added to the system using the supply line from the demineralized water storage tank. In an emergency, replenishment of boric acid water can be accomplished using the supply line from the RWSP.
- The system is designed to maintain the water level of the SFP to prevent uncovering of stored fuel even if there is leakage due to failure of the piping.

The piping connected to the SFP is equipped with siphon breakers to prevent uncovering stored fuel in the event there is leakage in the system. During decay heat removal operation, SFP water flows from the SFP to the SFP pump suction, through the SFP cooler, transferring heat from the SFP water to the component cooling water, and returns to the SFP. A portion of the SFP water is diverted through the demineralizers and the filters in the purification part of the system to maintain water purity. During normal decay heat removal operation, one train can be used to purify the reactor cavity, the RWSP, and the RWSAT.

1.2.1.5.4.3 Fuel Storage and Handling System

The function of the fuel storage and handling system is to carry out fuel storage and handling safely and securely from the time the new fuel is brought into the power plant to the time the spent fuel is removed from the plant. The new fuel is stored in the new fuel pit in the R/B. After reactor shutdown, the spent fuel in the reactor is transferred to the spent fuel pit through the reactor cavity, refueling canal and fuel transfer tube, using the refueling machine, fuel handling machine, and relative fuel handling equipment.

All of the spent fuel transfer functions are carried out under boric acid water, which performs the functions of shielding and cooling. The spent fuel is stored in the spent fuel pit. After cooling is complete, the spent fuel is inserted into the spent fuel cask using the spent fuel cask handling crane, and then transported outside of the plant.

The major equipment of fuel storage and handling system are as follows.

- New Fuel Storage Pit
- Spent Fuel Storage Pit
- Cask Pit
- Cask Washdown Pit
- Reactor Cavity and Refueling Canal
- Refueling Machine
- Fuel Handling Machine
- New Fuel Elevator
- Fuel Transfer System

1.2.1.5.4.4 Water Systems

Essential Service Water System - The essential service water system (ESWS) consists of the essential service water pumps, piping, valves, and instrumentation. The system provides service water for the component cooling water HXs and the essential chiller units. The ESWS transfers heat from those components to the ultimate heat sink (UHS).

The ESWS satisfies the following design requirements:

- The essential service water pumps and the related piping are designed to take service water from the UHS and deliver it to the component cooling water HXs and the essential chiller units.
- The system consists of four independent trains. Each train has one essential service water pump.
- The ESWS is designed to provide sufficient cooling capacity for normal operation, transients, and accidents such as a LOCA and a LOOP.
- The essential service water pumps can be powered from the safety buses so safety-related functions are maintained during a LOOP.
- The ESWS is designed to perform safety-related functions assuming a single failure in a one train, with another train out of service for maintenance.
- The ESWS is automatically initiated by a safety injection signal.

Each essential service water pump takes water from the UHS, pumps it through the component cooling water HXs and essential chiller units, and discharges it to the discharge pit. In an accident situation, the necessary safety functions can be performed by two of the four trains.

The system configuration for each operating mode is as follows:

- During normal operation, two trains are used to supply service water to two trains of the CCWS. In hot weather, if the outlet temperature of the component cooling water HXs increases to near 100°F, three service water trains are used.
- During plant cooldown by the SGs, a similar configuration is employed.
- During residual heat removal operation, after cooldown by the steam generators, all four CCWS trains and ESWS trains are operated.
- During refueling, the number of essential service water pumps and CCWS trains is determined by the decay heat to be removed.

Component Cooling Water System - The CCWS provides cooling water for the components of the primary systems during normal operation, plant shutdown, and after an accident. It also serves as an intermediate system between the reactor coolant and the ESWS to prevent leakage of radioactive material into the environment.

One subsystem consists of trains A & B, and the other subsystem consists of trains C & D, for a total of four trains. Each train has one CCW pump and one CCW heat exchanger and provides 50% of the cooling capacity required for the CCWS safety function. Electrical power to all trains is supplied by the safety-related buses, which are backed up by Class 1E power supplies. The CCWS provides cooling water for safety-related components such as the CS/RHR HXs, the spent fuel pit HXs, the safeguard pump coolers and other components used during normal operation, such as the CVCS coolers, the radwaste management system coolers, and the RCP coolers. The surge tanks accommodate the thermal expansion and contraction of the cooling water and potential leakage.

The CCWS satisfies the following design requirements:

- The design is based on the service water maximum design temperature (95°F).
- The system is designed to assure that leakage of radioactive fluid from the cooled components is held within the plant.
- The CCWS is designed to provide sufficient cooling capacity not only for the components required during normal plant conditions such as power operation and residual heat removal operation, but also for those components important to the safety in the event of an accident such as the LOCA, or an abnormal operational transient involving the LOOP.

- The CCW pumps can be powered from the safety related buses so safety functions can be maintained during a LOOP.
- The CCWS is designed to perform safety functions assuming a single failure in the one train and another train being out of service for maintenance.
- The CCW pumps are automatically initiated by a safety injection signal.
- The radiation monitors are installed to detect the leakage of radioactive materials into the CCWS.

The system has cooling lines to the various components to be cooled (i.e., associated piping, valves, and instrumentation). The component cooling water flows from the component cooling water pumps, through the component cooling water HXs, to the components to be cooled, and returns to the pumps.

The CCW surge tanks are connected to the suction side of the CCW pumps. Makeup water is also supplied to the suction side. The CCW surge tank accommodates expansion and contraction of the system water due to temperature change or leakage. In case of a small leak in the system, the tank can supply makeup water until the leak is isolated. Isolation valves are installed for each component.

The CCW surge tanks are installed at an elevation high enough to provide the CCWS pumps with sufficient suction head.

The system configuration for each operating mode is as follows:

- During normal power operation, the CCWS is operated with two pumps and two HXs, one train in each subsystem. Should a running pump fail, the other pump in the same subsystem automatically starts.
- During plant cooldown by the SGs, the CCWS is operated in the same way as during normal power operation.
- During residual heat removal operation, after cooldown by the SGs, the RHRS is initiated. At this time, the CCWS isolation valves to the CS/RHR HXs are opened and all four CCWS trains are operated. The failure of one train may increase the time for cooldown, but does not affect the safe operation of the plant.
- During refueling, the CCWS is operated with two or three pumps and HXs. The system is aligned the same as it is for the latter phase of normal cooldown.
- Following receipt of a safety injection signal, all four CCW pumps are automatically initiated. Upon receipt of a safety injection signal plus the respective train CCW pump start signal, the isolation valves to the CS/RHR HXs are opened.

Primary Makeup Water System - The primary makeup water system (PMWS) consists of primary makeup water storage tanks, primary makeup water pumps, piping, valves, and instrumentation.

The PMWS has the following functions:

- Supply makeup water to primary system equipment
- Provide containment isolation

The PMWS is designed to store and provide degasified, demineralized makeup water, and has no safety function except for containment isolation. The primary makeup water storage tank is designed to store sufficient water to meet the demands of the primary system components. The primary makeup water pumps provide makeup water to several locations, including:

- The boric acid blender (makeup to the reactor coolant system).
- The chemical mixing tank (as a solvent).
- The CCW surge tanks (emergency makeup).
- The demineralizers, spent resin discharge header, and spent resin storage tanks.

The PMWS receives water from the demineralized water system and also receives the recycled water from the Boron Recycle System (BRS).

Chilled Water Systems - The chilled water systems are designed to provide chilled water for the HVAC systems as a cooling medium to satisfy the indoor ambient temperatures. The chilled water systems consist of the essential chilled water system and the non-essential chilled water system.

The essential chilled water system satisfies the following design requirements:

- The essential chilled water system provides chilled water for the safety-related HVAC systems during normal operation and accident conditions.
- The essential chilled water system consists of four independent trains. Each train includes a chiller unit, a chilled water pump, a compression tank, a chilled water distribution loop, a make-up water loop and a control system.
- The essential chilled water system is powered from Class 1E buses so the safety functions can be maintained during a LOOP.
- This essential chilled water system is automatically initiated by ECCS actuation signal.

The non-essential chilled water system design is based on following:

- The non-safety chilled water system provides chilled water for the HVAC system serving the non-safety related area during normal operation.

1.2.1.5.4.5 Process Auxiliary Systems

Compressed Air and Gas System (CAGS) - The CAGS includes of the instrument air system (IAS), the station service air system (SSAS) and compressed gas system. The major function of CAGS is that of the IAS. The IAS supplies clean, dry, and oil-free compressed air to the following equipment:

- Air-operated valves
- HVAC air dampers
- Pneumatic instruments and controls
- Measuring instruments
- Other equipment, which is not safety related.

The IAS satisfies the following design requirements:

- Compressed air is not used for safety function
- The IAS comprises the redundant compressor packages.
- The IAS compressor package consists of an inlet air filter/silencer, a compressor, an aftercooler, an air reservoir and a drier.

The instrument air compressors are of an oil-free type so that the clean compressed air can be provided. Two instrument air compressors, each with a 100% capacity, are provided.

Process and Post-accident Sampling System - The process and post-accident sampling system (PSS) consists of equipment to collect representative samples of the various process fluids in a safe and convenient manner and provide the means to monitor the plant's various system conditions using the collected and analyzed samples. The system's design adheres to the as low as reasonably achievable (ALARA) principle during both normal and post-accident conditions. The PSS provides the following functions:

- Collect cooled and depressurized liquid samples at high temperature and high pressure, and liquid samples from the RCS, the CVCS, and the RHRS for purposes of analysis.
- Provide gaseous samples of containment atmosphere to monitor hydrogen concentration and radioactivity.

- Obtain post accident liquid and gaseous samples following an accident for the purpose of analyzing the post accident conditions to augment the monitoring capability in the long term.
- Monitor impurity levels in the steam, feedwater and condensate systems and maintain purity within predetermined limits during normal plant operation.
- Monitor impurity levels in the secondary water in the SGs and keep them within predetermined limits during normal plant operation by providing continuous blowdown sample at an adequate flow rate.

The PSS satisfies the following design requirements:

- It cools and depressurizes samples collected at high temperature and high pressure.
- It collects liquid samples for monitoring the reactor coolant during normal plant operation and after an accident.
- It collects gaseous samples of the containment atmosphere following an accident. Containment isolation is not violated while collecting samples of the reactor coolant and the containment atmosphere after an accident.

CVCS, including the Boron Recycle System (BRS) - The CVCS includes heat exchangers, letdown orifices, purification filters and demineralizers, volume control tank, boric acid tanks and transfer pumps, charging pumps, seal injection filters, piping, valves, and instrumentation.

The CVCS has the following functions:

- Maintain the coolant inventory in the RCS for all normal modes of operation, including startup, full-power operation, and cool down.
- Provide makeup capability for small RCS leaks.
- Perform purification by removal of the fission and activation products in the reactor coolant.
- Regulate the boron concentration in the reactor coolant during normal operation.
- Borate the RCS for shutdown.
- Control the reactor coolant water chemistry.
- Provide seal-water flow to the reactor coolant pumps.
- Provide pressurizer auxiliary spray water for depressurization of the RCS when none of the RCPs are operating.

Letdown flow comes from the RCS and flows through the regenerative HX, where its temperature is reduced by transferring heat to the incoming charging flow. The coolant is then depressurized as it passes through the letdown orifices and is further cooled in the letdown HX. A second stage of pressure reduction is performed by the low-pressure letdown valve, which maintains upstream pressure to prevent flashing downstream of the letdown orifices. The letdown flow rate is controlled to suit various plant operating requirements by selecting a combination of letdown orifices.

The letdown flow then passes through the mix bed demineralizers for purification. The flow may also pass through the cation bed demineralizer, which is used intermittently when additional purification is required. The deborating demineralizers are used near the end of core life for the removal of boron.

The coolant then flows to the volume control tank (VCT) through a spray nozzle in the top of the tank. Hydrogen is supplied to the VCT, where it mixes with fission gases that are stripped from the reactor coolant. Contaminated hydrogen is vented to the gaseous waste management system. The partial pressure of hydrogen in the VCT is controlled to establish the concentration of hydrogen dissolved in the reactor coolant.

Two centrifugal charging pumps are provided to take suction from the VCT and return the cooled, purified reactor coolant to the RCS. The charging flow is pumped to the RCS through the regenerative HX, and injected into a cold leg of the reactor coolant system. A portion of the flow is directed to the RCPs through a seal water injection filter. An auxiliary pressurizer spray provides a means of cooling and depressurizing the pressurizer near the end of plant cooldown, when the RCPs, which normally provide the driving head for the pressurizer spray, are not operating. An excess letdown path is provided in the event that the normal letdown path is inoperable. The excess letdown flow path is also used to provide additional letdown capability during the final stages of plant heatup.

Changes in the RCS inventory due to load changes are accommodated primarily in the pressurizer. The VCT provides surge capacity for reactor coolant expansion not accommodated by the pressurizer. If the water level in the VCT exceeds the normal operating range, a three-way valve downstream of the reactor coolant filter diverts a portion of the letdown fluid to the BRS. The BRS recycles reactor coolant for the reuse of boric acid and primary makeup water. The system decontaminates the coolant by means of demineralization and gas stripping, and uses evaporation to separate and recover the boric acid and primary makeup water.

HVAC Systems - The MCR HVAC systems and the annulus emergency exhaust system have been previously described. The other HVAC systems are designed to provide the following functions:

- Provide proper environmental conditions within plant areas during normal operation and in accident conditions.

- Maintain airflow from areas of low radioactivity to areas of potentially higher radioactivity during normal operation.

- Limit concentration of airborne radioactivity to levels below the allowable values set by 10CFR20 by supplying and exhausting sufficient airflow during normal operation.
- Minimize exfiltration from the areas with potential airborne radioactive contaminants by maintaining a slight negative pressure relative to the outside atmosphere during normal operation.

The safety related HVAC systems satisfy the following design requirements:

- The safety related HVAC systems can maintain proper environmental conditions within the safety related equipment areas during accident condition.
- The safety related HVAC systems perform the specified functions assuming a single active component failure.
- The safety-related HVAC systems are powered from Class 1E buses so the safety functions are maintained during a LOOP.
- The safety related HVAC systems are automatically initiated by an ECCS actuation signal or the high temperature of served area.

The safety related HVAC systems are:

- Class 1E Electrical Room HVAC System
- Safeguard Component Area HVAC System
- Emergency Feed Water Pump Area HVAC System
- Safety Related Component Area HVAC System

1.2.1.5.4.6 Radioactive Waste Management Systems

Liquid Waste Management System (LWMS) - The LWMS is designed to monitor, control, collect, process, handle, store, and dispose of liquid radioactive waste generated as a result of normal operations, including AOOs. The LWMS is classified as comprising the liquid waste processing system and the reactor coolant drain system. The collected liquid waste is treated adequately and monitored prior to discharge.

The LWMS satisfies the following design requirements:

- Provide the capability to segregate the collection of equipment drains and floor drains
- Provide the capability to treat the liquid waste to acceptable recycle specifications for plant use

- Provide the capability to treat liquid waste to the acceptable release specifications
- Provide the capability to store, sample, and analyze treated liquid
- Provide the capability to safely control and dispose of treated liquid
- Provide the capability to stage reactor coolant drains

Tanks, equipment, pumps, etc., used for storing and processing radioactive material are located in controlled areas and shielded in accordance with their design basis source term inventories. As a result, occupational doses comply with dose limits and are ALARA. After the waste has been processed, it is temporarily stored in monitor tanks where it is sampled prior to recycle or discharge.

The LWMS has different subsystems so that the liquid wastes from various sources can be segregated and processed separately in the most appropriate manner for the type of waste. These systems are interconnected in order to provide additional flexibility in processing the wastes and to provide redundancy.

Gaseous Waste Management System (GWMS) - The GWMS is designed to monitor, control, collect, process, handle, store, and dispose of gaseous radioactive waste and consists of waste gas compressor packages, gas surge tanks, and a noble gas holdup system.

The GWMS satisfies the following design requirements:

- Provide the capability to monitor, control, collect, process, handle, store, and dispose of gaseous radioactive gaseous waste generated as the result of normal operation and AOOs.
- Provide reasonable assurance that the release of radioactive material in gaseous effluents is kept ALARA.
- Remove and reduce radioactive materials to the environment

The GWMS uses four gas surge tanks to provide temporary storage of radioactive gas for the decay of the short-lived isotopes that contribute the majority of radioactivity. It also includes four charcoal beds for the removal of radioactive gases for decay before some of the gases are released into the environment. The charcoal beds provide adequate delay and decay time before the gas effluent is routed to the discharge structure.

Nitrogen waste gas is compressed by the waste gas compressor packages in order to decrease its volume. It is then sent, intermittently, after passing through the gas surge tank, to an active carbon noble gas holdup system. After decay, the nitrogen waste gas is released from the vent stack through the charcoal filters of the ventilation system.

Hydrogen waste gas from the volume control tank containing fission products is released from the vent stack through an active carbon type noble gas holdup system and the charcoal filters of the ventilation system. The radioactivity level is monitored before release. The treatment of the hydrogen waste gas prevents leakage, which might lead to a hydrogen explosion. The atmosphere of each room where components containing hydrogen waste gas are located is continuously ventilated.

Solid Waste Management System (SWMS) - The SWMS is designed to provide collection, processing, packaging, and storage of radioactive wastes produced during normal operation and AOO including startup, shutdown, and refueling operations.

The SWMS has the following functions:

- Provide the capability to segregate and package dry and wet solid wastes.
- Provide the capability for processing, packaging, and storing radioactive wastes such as spent resin, spent activated carbon generated from various systems.
- Process and package wastes into disposal containers that are approved and are acceptable to waste disposal facilities.

The SWMS provides separate treatment and handling methods for the different waste types. The waste types include spent resins, spent carbon, sludge, oil waste, and dry active waste.

1.2.1.5.4.7 Fire Protection Systems

Fire protection systems are installed to minimize the adverse effects of fires on SSCs important to safety.

The fire protection systems are designed to perform the following functions:

- Detect and locate fires and provide operator indication of the location.
- Provide the capability to extinguish fires in any plant area, to protect site personnel, limit fire damage, and enhance safe shutdown capabilities.
- Supply fire suppression water at a flow rate and pressure sufficient to satisfy the demand of any automatic sprinkler system plus 500 gpm for fire hoses, for a minimum of 2-hours.
- Maintain 100% of fire pump design capacity, assuming failure of the largest fire pump or the LOOP.
- Following a safe shutdown earthquake, provide water to hose stations for manual fire fighting in areas containing safe shutdown equipment.

The fire protection system detects fires and provides the capability to extinguish or control them using fixed automatic and manual suppression systems, manual hose streams, and/or portable fire fighting equipment. The fire protection system consists of a number of fire detection and suppression subsystems, including:

- Detection systems for early detection and notification of a fire occurrence.
- A water supply system including fire pumps, adequate fire water supply source, yard main, and interior distribution piping.
- Fixed automatic fire suppression systems and equipment, including hydrants, standpipes, hose stations and portable fire extinguishers.

1.2.1.5.5 Instrumentation and Control Systems Designs

The instrumentation and control (I&C) systems provide capability to control and regulate the plant systems manually and automatically during normal plant operation, and provide protection against unsafe plant operation. The primary purpose of the I&C systems is to provide automatic protection and exercise proper control against unsafe and improper reactor operation during steady state and transient power operations. It also provides initiating signals to actuate safety functions, which are assigned to mitigate the consequences of faulted conditions and assure safe shutdown. Safety functions are those actions required to achieve the system responses assumed in the safety analyses and those credited to achieve safe shutdown of the plant. The I&C system is primarily a digital system with the exception of the analog diverse actuation system (DAS). The DAS design consists of conventional equipment that is totally diverse and independent from the digital platform of the PSMS and PCMS, so that a postulated beyond design basis CCF in these digital systems will not impair the DAS functions.

The overall I&C system consists of the following four categories of systems:

- Human-system interface (HSI) System, including HSI portions of the protection and safety monitoring system, the plant control and monitoring system and the DAS
- Protection and safety monitoring system (PSMS)
- Plant control and monitoring system (PCMS)
- DAS

The PSMS has the following functions:

- Reactor trip
- ESF actuation
- Safe shutdown
- Post accident monitoring

- Interlock important to safety

The PSMS design is based on the safety system requirements, including the following:

- Single failures: A credible single failure within a safety system does not prevent the initiation or accomplishment of a protective function at the system level, even when a channel is intentionally bypassed for test or maintenance.

- Completion of protective actions: Once initiated, either automatically or manually, protective functions proceed to completion.

- Hardware quality: Safety system equipment is designed, manufactured, inspected, installed, tested, operated, and maintained in accordance with a prescribed quality assurance program.

- Qualification: Safety system equipment is qualified by type test, previous operating experience, or analysis or any combination of these three methods, to assure its performance as specified in the design basis.

- Independence: Physical separation is used to achieve separation of all redundant train components. The functional capability is maintained during and after a design basis earthquake. Isolation devices used to effect a safety system boundary are classified as part of the safety system. Credible failure on the non-safety side of an isolation device does not prevent a safety system from meeting its performance requirements.

- Testability: Capability for testing and calibration of safety system equipment is provided while retaining the capability of the safety system to accomplish its safety function.

- Monitoring and information: The display information for manual actions for which no automatic control is provided and the display instrumentation required for safety systems to accomplish their safety functions is part of the safety systems.

- Bypasses: If the protective actions of a safety system have been bypassed or deliberately rendered inoperative for any purpose other than operating bypass, continued indication of this fact for each affected safety group is provided in the control room.

- Software quality and life cycle processes: Software development process is in accordance with an approved Software quality assurance (QA) Plan.

- Independent verification and validation: Verification and validation (V&V) is performed in the development and modification of software. The development activities and tests are verified and validated by individuals or groups independent from those who developed the original design.

- Communications Independence: Data communication between safety divisions or between safety and non-safety divisions does not inhibit the performance of the safety function.

The PSMS satisfies the following design requirements:

- PSMS consists of four train redundant reactor protection system (RPS), engineered safety features actuation system (ESFAS), safety logic system (SLS) and safety grade human-system interface system (HSIS), and conventional switches for system level manual actuation.

- Once initiated, either automatically or manually, protective functions proceed to completion. In addition, system level signals cannot be manually reset until the plant condition is restored to a pre-determined setpoint.

- The quality of PSMS components and modules and the quality of the PSMS design process is controlled by a program that meets the requirements of ASME NQA-1-1994.

- The PSMS is qualified for worst-case environmental and seismic requirements for the place of its installation. The PSMS qualification envelopes the environmental and seismic boundary conditions.

- Each train of the PSMS is independent from each other and from non-safety systems, including the PCMS. Electrical independence is maintained through qualified isolation devices, including fiber optic data communications cables. Functional independence between controllers is maintained through communication processors that are separate from function processors, and through (1) logic that assures prioritization of safety functions over non-safety functions and (2) logic that does not rely on signals from outside its own train to perform the safety function within the train.

- Testing from the sensor inputs of the PSMS through to the actuated equipment and HSI is accomplished through a series of overlapping sequential tests and calibrations. The majority of the tests are conducted automatically through self-diagnostics. Most remaining manual tests may be performed with the plant at full power.

- PAM Type A, B and C variables have redundant instrumentation and are displayed on at least two redundant safety-related visual display units .

- The PSMS system level bypassed or inoperable status indication is provided. These indications are displayed as the spatially dedicated continuously visible (SDCV) information on large display panel in the MCR.

- Software life cycle process is controlled using the software program manual to improve the functional reliability and design quality of software.

- The independent V&V is performed in the development and modification of software in accordance with the Software Program Manual.

- The PSMS employs communication processors that are separate from the processors that perform safety logic functions. The safety processors and communication processors communicate via dual ported memory. This assures there is no potential for communications functions, such as handshaking, to disrupt deterministic safety function processing.

1.2.1.5.6 Electric Power

Offsite electric power is provided to the onsite power system from the grid and other generating stations by at least two physically independent transmission lines. During the plant startup, shutdown, maintenance, and during all postulated accident conditions, offsite electric power can be supplied to the plant site from the plant high-voltage switchyard through two physically independent transmission tie lines.

- During the plant's startup and shutdown and in all postulated accident conditions, offsite electric power is supplied to the plant site from the plant high voltage switchyard through two physically independent transmission tie lines. One of these two transmission tie lines connects to the high voltage side of the main transformer (MT), and the other connects to the high voltage side of the reserve auxiliary transformers (RATs).

- The main generator (MG) is connected to the low voltage side of the MT and the high voltage side of the unit auxiliary transformers (UATs). There is a generator load break switch (GLBS) between the MG and the MT. When the MG is on-line, it provides power to the onsite non safety-related electric power system through the UATs.

- When the GLBS is open, offsite power to the onsite non safety-related electric power system is provided through the MT and the UATs. With the GLBS open or closed, offsite power to the onsite safety-related electric power system is provided through the RATs. If power is not available through the UATs, offsite power is provided to both safety-related and non safety-related onsite electric power system through the RATs. Similarly, if power is not available through the RATs, offsite power is provided to both safety-related and non-safety related onsite electric power system through the UATs.

The plant's high-voltage switchyard is site specific and is not a part of the reference plant design.

The onsite ac power system is supplied with offsite power from the grid by two physically independent circuits. Each offsite power circuit has enough capacity and capability to power the loads required during all modes of plant operation, including emergency shutdown and postulated design basis events. The onsite power system consists of both ac power system and dc power system. Both ac and dc systems include Class 1E and non-Class 1E systems. The Class 1E onsite ac and direct current (dc) power systems provide power to the safety loads required during LOOP and design basis accident (DBA) conditions. The power from the transmission system to the Class 1E distribution is preferred to furnish electric power under accident and post-accident conditions. The redundant trains are physically separated and electrically isolated from each other and also from the non-Class 1E systems.

- The onsite electric power system provides power to all plant auxiliary and service loads. The onsite electric power system is comprised of alternating current (ac) and direct current (dc) systems. Both ac and dc onsite electric power systems have a safety-related Class 1E power system feeding all Class 1E loads, and a non safety-related non-Class 1E power system feeding all non-Class 1E loads.

- The Class 1E onsite power system has four independent divisions. Each division of the Class 1E ac onsite power system, in addition to its connection to offsite power sources from the grid, has an onsite emergency power source, consisting of a generator driven by a gas turbine (Class 1E GTG). The plant also has two non-Class 1E GTGs as alternate ac (AAC) power sources.

- There are two non-Class 1E 13.8 kV MV buses N1 and N2, four non-Class 1E 6.9 kV MV buses N3, N4, N5 and N6, two non-Class 1E 6.9 kV MV permanent buses P1 and P2 and four Class 1E 6.9 kV MV buses A, B, C and D. Each of the Class 1E 6.9 kV MV buses has its own Class 1E GTG. Similarly, each of the non-Class 1E 6.9 kV MV permanent buses has its own AAC-GTG. All medium voltage buses can be powered from either UAT or RAT. In addition to the 13.8 kV and 6.9 kV medium voltage levels, the onsite power distribution system has other low voltage (480 volt ac, 208/120 V ac, 125 V dc.) power distribution systems. The safety-related medium voltage buses A, B, C and D feed the corresponding safety-related low voltage buses, and the non safety-related medium voltage buses feed the non safety-related low voltage buses.

- Both Class 1E and non-Class 1E dc systems are normally powered by the battery chargers connected to the onsite ac power system. When power supply from the battery charger is not available, the onsite dc power system is supplied power from station batteries.

- The onsite power distribution system also includes both safety and non-safety I&C power supply systems. The I&C power supply systems are 120 V ac uninterruptible power supply (UPS) systems used for the reference plant's instrumentation and control systems. The UPS systems are normally powered from the 480 V MCCs through inverters with battery backup.

The term "Station Blackout" (SBO) means the complete loss of ac electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., the loss of offsite electric power system concurrent with a turbine trip and the unavailability of the onsite emergency ac power system). An SBO does not include the loss of available ac power to buses fed by the station batteries through inverters or by alternate ac (AAC) sources, nor does it assume a concurrent LOCA, a single failure or a DBA. The plant is able to withstand on SBO of specified duration and recover from it.

- During an SBO, all ac sources and the onsite Class 1E GTGs are assumed to be inoperable. Two non-Class 1E GTGs are provided as AAC sources. To minimize the potential for common mode failures with the Class 1E GTGs, different rating

GTGs with diverse starting system are provided as AAC sources. The auxiliary and support systems for the AAC GTGs are independent and separate from the Class 1E GTGs to minimize the potential for common mode failures.

- In the US-APWR design, power to the shutdown buses can be restored from the AAC sources within 60 minutes and, hence, a coping analysis for a duration of 60 minutes is performed.
- Until AAC GTG restores the Class 1E power system within one hour after an SBO occurs, all pumps and fans can not be operated. However, during this time period, the plant is in a condition similar to hot shut down. The turbine driven emergency feedwater pump (EFW) pump and the main steam relief valve remove decay heat so that the core and the reactor coolant system (RCS) are kept in a safe mode. RCP seal can keep its integrity for at least one hour without water cooling. The all Class 1E electrical cabinets and I&C cabinets can keep their integrity within one hour without HVAC.

1.2.1.5.7 Main Control Room and Other Human-System Interface

The main control room (MCR) is designed to perform centralized monitoring and control of the instrumentation and control systems that are necessary for use during normal operation, abnormal transients, and accidents. Furthermore, the main control room boards are designed to reduce the potential for misoperation and misjudgment and to allow easy operation. The human-system interface (HSI) other than the MCR includes the remote shutdown console (RSC), local control stations such as the auxiliary equipment control console, the technical support center (TSC), and the emergency operations facility (EOF).

1.2.1.6 Site Characteristics

The US-APWR is a standard nuclear power plant designed to be constructed on a site, whose parameters are as described in DCD Chapter 2 (Site Characteristics), which are used as the basis for design certification. The allowable site interface parameters described in Chapter 2 are selected by MHI to bound most potential sites in the U.S.

The site-specific details of a US-APWR site plan is to be presented in the licensee's combined license application (COLA). A typical site plan has, however, been prepared by MHI and is shown in Figure 1.2-1.

The area within the perimeter fence of a US-APWR installation includes a site-specific portion of the facility. The control structure at the main gate controls site ingress and egress. As shown on the site plan, the main building structures are arranged with the R/B in the center and the other buildings clustered around the R/B to facilitate safe and efficient operation. The final configuration of the main cooling system is site-specific; however, the reference plant main cooling complex is of the once-through cooling type. The unit's auxiliary transformers, reserve auxiliary transformer, and the main step-up transformers are located in the transformer area. The main switchyard area is site-specific.

1.2.1.7 Plant Arrangement

1.2.1.7.1 General Plant Arrangement

The main US-APWR power block is comprised of the following buildings and structures:

- The reactor building (R/B), including prestressed concrete containment vessel (PCCV)
- The power source buildings (PS/Bs)
- The power source fuel storage vaults (PSFSV)
- The essential service water pipe tunnel (ESWPT)
- The auxiliary building (A/B)
- The access building (AC/B)
- The turbine building (T/B)

The outline and the arrangement of those buildings and structures are shown in Figure 1.2-1. The equipment layout within the buildings provides for ease of plant operation and maintenance, and minimizes personnel radiation exposure. Provisions, including redundant train separation and segregation barriers, have been made to assure that the functions of the safety-related systems are maintained in the event of postulated incidents such as fires, floods, and high-energy pipe break events. Within the buildings, access control zonings are established to regulate access to radiation areas and to define the required radiation shielding and monitoring during operation, shutdown, and accident conditions.

The R/B, PS/B, PSFSV, and ESWPT are designed and constructed as safety-related structures, to the requirements of seismic Category I, as defined in RG 1.29. These safety-related structures are designed for the effects of all applicable loads and their combinations, including the postulated seismic response loads. These structures are designed to withstand the effects of natural phenomena such as hurricanes, floods, tornados, tsunamis, and earthquakes without loss of capability to perform their safety functions. They are also designed to withstand the effects of postulated internal events such as fires and flooding without loss of capability to perform their safety functions.

The remaining power block buildings are designed as non safety-related structures, and are free-standing on separate concrete base mats. The A/B and T/B are designed to meet seismic Category II requirements as defined in RG 1.206. Other structures are designed to American National Standard Institute (ANSI), ASCE and other applicable codes, and meet non-seismic Category requirements.

Radioactive equipment and piping in all buildings are arranged and shielded to minimize radiation exposure. Pathways through the plant are designed to accommodate equipment maintenance and equipment removal from within the plant. The size of the pathways is dictated by the largest piece of equipment that may have to be removed or installed after initial installation. Where required, laydown space is provided for disassembling large pieces of equipment to accommodate the removal or installation process. Adequate space is provided for equipment maintenance, laydown, removal, and inspection. Hatches, monorails, hoists, and removable shield walls are provided to facilitate maintenance.

The general arrangement drawings for the US-APWR are provided in Figures 1.2-2 through 1.2-51.

1.2.1.7.2 Specific Building Descriptions

1.2.1.7.2.1 Reactor Building (R/B)

The R/B has five main floors. The R/B consists of the following five functional areas:

- Containment facility and inner structure
- Safety system pumps and HXs area
- Fuel storage and handling area
- Main steam and feed water area
- Safety-related electrical area

The containment facility is comprised of the PCCV and the annulus enclosing the containment penetration area, and provides an efficient leak-tight barrier and radiation protection under all postulated conditions, including a LOCA. The PCCV is a prestressed concrete structure designed to endure peak pressure under LOCA conditions. Access galleries are provided for periodic inspection and testing of circumferential and axial prestressing tendons.

For ease of access during operation, maintenance, repair, and refueling, the following accesses to the PCCV are also provided:

- A normal personnel airlock, located at floor level below the operating floor
- An equipment hatch and emergency airlock, located at operating floor level

The annulus is located adjacent to the PCCV and includes all penetration areas, to prevent the direct release of containment atmosphere to the environment through the containment penetrations. The pressure in the annulus is kept at a slightly negative level following accident conditions to control the release of radioactive materials to the environment.

The RWSP is located in the lowest part of the containment. The RWSP provides a continuous suction source for both the safety injection pumps and the CS/RHR pumps, thereby eliminating the switchover of suction source from the injection to the recirculation phase of accident recovery. The RWSP has four recirculation strainers on the floor, and the wall and floor of the RWSP are lined with stainless steel liner plates.

The reactor vessel is located at the center of the containment and is surrounded by a cylindrical concrete wall as a primary biological shield. There are four reactor coolant loops, each loop comprised of a steam generator, an RCP, and loop piping. Concrete walls surrounding the loops are provided as supporting media and as secondary biological shields. The pressurizer is located in its own compartment and is adjacent to the steam generators to minimize the length of the surge piping to the reactor coolant loop.

A refueling cavity with stainless steel liner is provided above the reactor vessel for refueling operations. The fuel transfer tube connects this cavity to the fuel storage and handling area located outside the containment. The main steam and feedwater pipes that connect to the steam generators are routed within the containment with consideration of minimizing pipe run lengths, while providing sufficient flexibility to accommodate thermal expansion.

Safety System Pumps and HX*s Area* - The safety system pumps (CS/RHR pumps and safety injection pumps), which require sufficient net positive suction head (NPSH) to draw water from the recirculation sumps inside the containment, are located at the lowest level of the R/B to secure the required NPSH. In addition, they are located adjacent to the containment to minimize pipe lengths. The safety system HXs (CS/RHR HXs) are located on the upper floor of the R/B.

Fuel Storage and Handling Area - The fuel storage and handling area is located in the R/B.

Fuel handling operations are performed on the top floor of the area at the same level as the C/V operating floor. The containment emergency airlock is located adjacent to the fuel handling area to facilitate easy access between the containment and fuel handling area when refueling procedures are in progress. The bridge crane is located to span the spent fuel pit, the transfer canal, and the cask loading pit. The spent fuel cask handling crane is capable of lifting the spent fuel cask from ground level to the operating floor.

Main Steam and Feed Water Piping Area - The main steam and feedwater piping room is located on the top floor of this area and contains the main steam and feedwater pipes, where they pass between the T/B and the containment.

Safety-related Electrical Area - The safety-related electrical area has two floors and is located in the R/B and under the main steam and feed water piping area. It is normally a nonradioactive zone and is completely separated from the radioactive zones of the R/B. This area houses the following safety-related facilities:

- MCR
- Safety metal clad switchgear and load center
- Safety I&C room

Separation of Redundant Systems - Four redundant safety systems containing radioactive material are located in each zone of the four quadrants surrounding the containment structure. Each of the quadrant areas is separated by a physical barrier to assure that the functions of the safety-related systems are maintained in the event of postulated incidents such as fires, floods, and high-energy pipe break events. Nonradioactive safety systems such as EFWS, CCWS and electrical systems are located in the non-radioactive control area of R/B. This area is also separated into four divisions by physical barriers to assure that the functions of the safety-related systems are maintained in the event of postulated incidents such as fires, floods, and high-energy pipe break events.

1.2.1.7.2.2 Power Source Buildings (PS/B)

Two PS/Bs are arranged adjacent to the R/B. These buildings are freestanding on reinforced concrete mats, and each building contains two identical emergency power sources, which are separated from each other by physical barriers. The safety-related HVAC chillers are also located in these buildings. The electrical, I&C and HVAC equipment related to the EPSs are also contained in the PS/Bs.

1.2.1.7.2.3 Power Source Fuel Storage Vault (PSFSV)

The PSFSVs are underground structure constructed with reinforced concrete, and classified as seismic Category I. The vaults contain the fuel oil tanks of safety-related gas-turbine generators.

1.2.1.7.2.4 Essential Service Water Pipe Tunnel (ESWPT)

The ESWPT is an underground structure constructed with reinforced concrete, and is classified as seismic Category I. Terminating in part under the T/B, the structure is isolated from other structures to prevent any seismic interaction. The other termination point is located at the Ultimate Heat Sink Related Structure (UHSRS) that connects to the Ultimate Heat Sink (UHS) water.

1.2.1.7.2.5 Auxiliary Building (A/B)

The A/B is located adjacent to the R/B. The A/B contains the main components of the waste disposal systems and the non safety-related electrical area. The non safety-related electrical area is normally a non-radioactive zone and is completely separated from the radioactive zones of the A/B.

1.2.1.7.2.6 Access Building (AC/B)

The AC/B is located adjacent to the A/B. The AC/B houses the access control area, and the chemical sampling and laboratory area.

1.2.1.7.2.7 Turbine Building (T/B)

The T/B houses the non safety-related equipment of the T/G and its auxiliary systems, (main condenser, feedwater heaters, moisture separator reheaters, etc.). The T/B is

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Figure 1.2-1 Typical US-APWR Site Arrangement Plan

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-2 Power Block at Elevation -26’-4” - Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-3 Power Block at Elevation -8'-7" - Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-4 Power Block at Elevation 3'-7" - Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-5 Power Block at Elevation 13'-6" - Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-6 Power Block at Elevation 25’-3” - Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-7 Power Block at Elevation 35'-2" - Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-8 Power Block at Elevation 50’-2” - Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-9 Power Block at Elevation 76'-5" - Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-10 Power Block at Elevation 101’-0” - Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-11   Power Block at Elevation 115’-6” - Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-12 Power Block Sectional View A-A

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-13 Power Block Sectional Views B-B and C-C

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-14 Reactor Building at Elevation -26'-4" - Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-15 Reactor Building at Elevation -8'-7" – Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-16 Reactor Building at Elevation 3'-7" – Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-17 Reactor Building at Elevation 13’-6” – Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-18 Reactor Building at Elevation 25’-3” – Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-19 Reactor Building at Elevation 35'-2" – Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-20 Reactor Building at Elevation 50'-2" – Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-21 Reactor Building at Elevation 76'-5" – Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-22 Reactor Building at Elevation 101'-0" – Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-23 Reactor Building at Elevation 115'-6" – Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-24 Reactor Building Sectional View A-A

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-25 Reactor Building Sectional View B-B

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-26 Power Source Building at Elevations -26'-4" and -14'-2'' - Plan Views

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-27 Power Source Building at Elevations 3’-7”, 24’-2” and 39’-6” – Plan Views

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-28 Power Source Building Sectional View A-A

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-29 Auxiliary Building at Elevation -26'-4" – Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-30 Auxiliary Building at Elevation -8'-7" - Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-31 Auxiliary Building at Elevation 3'-7" - Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-32 Auxiliary Building at Elevation 13’-6” - Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-33 Auxiliary Building at Elevation 25'-3" - Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-34 Auxiliary Building at Elevation 35'-2" - Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-35 Auxiliary Building at Elevation 50'-2" - Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-36 Auxiliary Building at Elevation 76’-5” - Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-37 Auxiliary Building at Elevation 89'-7" - Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-38 Auxiliary Building Sectional View A-A

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-39 Auxiliary Building Sectional View B-B

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-40 Turbine Building at Elevation -18'-0" – Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-41 Turbine Building at Elevation 3’-7” – Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-42 Turbine Building at Elevation 34’-0” – Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-43 Turbine Building at Elevation 61'-0" – Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-44 Turbine Building at Elevation 88’-10” – Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-45 Turbine Building at Elevations 108'-4" and 113'-6" – Plan Views

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-46 Turbine Building at Elevation 165’-4”– Plan View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-47 Turbine Building Sectional View A-A

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-48 Turbine Building Sectional View B-B

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-49 Access Building at Elevations -26’-4”, -8’-0” and 3’-7” – Plan Views

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-50 Access Building at Elevations 17'-9", 30'-2" and 48'-2" – Plan Views

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 1.2-51 Access Building Sectional Views A-A and B-B

steel structure, which is designed to withstand all loads including the load of the overhead traveling crane. The foundation of the building is made of concrete.

The building is designed based on the following:

- The T/B is oriented in such a way that any plane perpendicular to the turbine generator axis shall not intersect with the R/B. This arrangement minimizes the probability of a turbine missile striking the R/B (consistent with the guidance of Reg. Guide 1.115).

- The T/B is independent of the R/B to prevent internal hazards in the T/B from spreading.

1.2.2 Combined License Information

COL 1.2(1) *The COL Applicant is to develop a complete and detailed site plan in the site-specific licensing process.*

1.3 Comparison with Other Facilities

1.3.1 Comparison Table

The major US-APWR design parameters and nominal values of these parameters are shown in Table 1.3-1 through 1.3-6. These values are shown in comparison with the Japanese - Advanced Pressurized Water Reactor (J-APWR) and a current operating U.S. four-loop plant design. All values provided are nominal and provided for comparison. The four-loop U.S. plant parameters are representative of the Standardized Nuclear Unit Power Plant System (SNUPPS).

Table 1.3-1 Comparison of General Information and Reactor Core Characteristics

Parameter	US-APWR	J-APWR	US Current four-loop
Gross electrical output (MWe)	1,700 class	1,538	1,186
Core thermal output (MWt)	4,451	4,451	3,411
Operation pressure (psia)	2,250	2,250	2,250
Hot leg temperature (°F)	617	617	618
Number of fuel assemblies	257	257	193
Fuel assembly lattice	17x17	17x17	17x17
Effective fuel length (ft.)	14	12	12
Number of fuel rods per fuel assembly	264	264	264
Average linear heat rate (kW/ft.)	4.6	5.3	5.4
Number of Rod Cluster Control Assemblies (RCCA)	69	69	53
Design life (years)	60	60	40

Table 1.3-2 Comparison of Reactor Coolant System and Connecting Systems (sheet 1 of 2)

Parameter	US-APWR	J-APWR	US Current four-loop
Reactor Coolant System			
Number of heat transfer loops	4	4	4
Operation pressure (psia)	2,250	2,250	2,250
Hot leg temperature (°F)	617	617	618
Reactor Vessel			
Vessel inner diameter(in)	203	203	173
Thermal shield/ reflector design	Neutron Reflector	Neutron Reflector	Neutron Pad Design
In-core instrumentation	Top mounted	Bottom mounted	Bottom mounted
Steam Generator			
Number	4	4	4
Type	Vertical U-tube heat exchanger	Vertical U-tube heat exchanger	Vertical U-tube heat exchanger
Heat transfer area (ft2)	91,500	70,000	55,000
Number of U-tube	6,747	5,830	5,626
Reactor Coolant Pump			
Number	4	4	4
Type	Vertical shaft, single-stage centrifugal	Vertical shaft, single-stage centrifugal	Vertical shaft, single-stage centrifugal

Table 1.3-2 Comparison of Reactor Coolant System and Connecting Systems (sheet 2 of 2)

Parameter	US-APWR	J-APWR	US Current four-loop
Thermal design flow (gpm/loop)	112,000	113,600	95,700
Motor output(hp/unit)	8,200	8,200	7,000
Pressurizer			
Internal volume (ft3)	2,900	2,300	1,800
Surge nozzle nominal diameter (in)	16	16	14
Reactor Coolant Pipes			
Pipe inner diameter(in)	31	31	Reactor inlet 27-1/2 Reactor outlet 29 RCP suction 31
Residual heat removal system			
Residual heat removal pump			
Number	4	4	2
Type	Horizontal, centrifugal	Horizontal, centrifugal	Vertical centrifugal
Flow rate(gpm)	3,000	3,000	3,800
SI use	no	no	yes
Containment Spray use	yes	yes	no
Residual heat exchanger			
Number	4	4	2
Type	Shell and U-tube type	Shell and U-tube type	Shell and U-tube type

Table 1.3-3 Comparison of Engineered Safety Features (sheet 1 of 3)

Parameter	US-APWR	J-APWR	US Current four-loop
Containment			
Type	PCCV	PCCV	PCCV
Inner Diameter(ft-in)	149-2	149-2	140
Inner Height(ft-in)	226-5	226-5	205
Containment Heat Removal System			
Containment Spray Pump			
Number	4	4	2
Type	Horizontal, centrifugal type	Horizontal, centrifugal type	Vertical, centrifugal type
Design flow rate(gpm)	3,000	3,000	Injection 3,165 Recirculation 3,750
RHR use	yes	yes	no
Residual heat exchanger			
Number	4	4	- (containment air fan cooler)
Type	Horizontal, U-tube type	Horizontal, U-tube type	-
Containment Spray Nozzles			
Number	348	344	197/header
Type	Hollow cone	Hollow cone	Hollow cone

Table 1.3-3 Comparison of Engineered Safety Features (sheet 2 of 3)

Parameter	US-APWR	J-APWR	US Current four-loop
Containment Air Fan Cooler (Safety)			
Number	-	-	4
Emergency Core Cooling Systems **High Pressure Safety Injection Pump**			
Number	4	4	2
Type	Multi-stage centrifugal pump with Inducer	Multi-stage centrifugal pump with Inducer	Multi-stage centrifugal
Flow rate(gpm)	1,540	1,540	440
Charging / Safety Injection Pump			
Number	-	-	2
Type	-	-	Centrifugal
Flow rate(gpm)	-	-	150
Low Pressure Safety Injection Pump			
Number	-	-	2
Type	-	-	Vertical centrifugal
Flow rate(gpm)	-	-	3,800
RHR use	-	-	yes
Accumulator			
Number	4	4	4
Type	Dual flow rate	Dual flow rate	Single flow rate

Table 1.3-3 Comparison of Engineered Safety Features (sheet 3 of 3)

Parameter	US-APWR	J-APWR	US Current four-loop
Water volume(gallon)	15,850	15,850	6,358
Emergency Water Storage Pit			
Number	1	1	1
Type	Pit inside containment	Pit inside containment	Tank outside containment
Capacity(gallon)	607,640	607,640	394,000

Table 1.3-4 Comparison of Instrumentation and Control System, and Electrical System

Parameter	US-APWR	J-APWR	US Current four-loop
Type of I&C system	Fully digital with exception of the analog Diverse Actuation System (DAS)	Fully digital with exception of the analog Diverse Actuation System (DAS)	analog
Electric Power System			
Safety Power System			
Number of Power Generator	4	2	2
Type	Gas Turbine Generator	Diesel Generator	Diesel Generator

Table 1.3-5 Comparison of Turbine System

Parameter	US-APWR	J-APWR	US Current four-loop
Turbine			
Type	Tandem Compound Six Flow	Tandem Compound Six Flow	Tandem Compound Six Flow
Number of elements	Four (one HPT and three LPTs)	Four (one HPT and three LPTs)	Four (one HPT and three LPTs)

Table 1.3-6 Comparison of Auxiliary System

Parameter	US-APWR	J-APWR	US Current four-loop
Emergency Feedwater System			
Motor Driven Emergency Feedwater Pump			
Number	2	2	2
Type	Horizontal, centrifugal	Horizontal, centrifugal	Horizontal, centrifugal
Turbine Driven Emergency Feedwater Pump			
Number	2	2	1
Type	Horizontal, centrifugal	Horizontal, centrifugal	Horizontal, centrifugal
Chemical and Volume Control System			
Charging Pump			
Number	2	3	2
Type	Horizontal centrifugal	Horizontal centrifugal	Horizontal centrifugal
SI use	no	no	yes
Volume Control Tank			
Number	1	1	1
CCW Pump			
Number	4	4	4
Type	Horizontal, centrifugal	Horizontal, centrifugal	Horizontal, centrifugal
CCW Heat Exchanger			
Number	4	4	4
Type	Plate type	Plate type	Shell and straight tube

1.4 Identification of Agents and Contractors

1.4.1 Applicant/Program Manager

MHI is responsible for the overall design and is the applicant seeking design certification of the US-APWR .

MHI is a diversified company that manufactures a range of products, including heavy machinery, ships, industrial equipment, wind turbines, aircraft engines, and power plants. MHI has over 32,000 employees worldwide.

MHI constructed Japan's first nuclear power plant, Mihama-1, in cooperation with Westinghouse, in 1971, using PWR technology. MHI incorporates the latest national LWR improvement and standardization programs and other standards initiatives into all of its technologies. MHI has built 23 nuclear power plants in Japan and currently has one under construction. The Tsuruga-3 and -4 units of the Japan Atomic Power Company are now in the licensing approval stages. The Tsuruga units, with an electrical capacity of over 1500 MW each, will incorporate the APWR plant technology.

MHI developed the APWR technology to meet the demands of the global nuclear power market. This plant design features improved performance, cost efficiency, and ease of maintenance. The APWR technology was originally developed as a part of Japan's Ministry of Economy, Trade, and Industry's "Third Phase Improvement Standardization Program for Light Water Reactors." The US-APWR design has been modified slightly to comply with the requirements of the U.S. nuclear industry. It integrates the operational experiences and technological improvements obtained from MHI's nuclear power plant experience in Japan. The advanced technology was achieved by careful research and development, including building and operating scale models of the plant components.

To meet U.S. customers' needs, MHI has established Mitsubishi Nuclear Energy Systems, Inc. (MNES), headquartered in Washington, D.C. MHI, through MNES, intends to establish its presence in the US market with efficient, safe and economical nuclear products and services.

1.4.2 Other Contractors and Participants

1.4.2.1 Obayashi Corporation

Obayashi Corporation has entered into a contract with MHI to provide consulting assistance in the engineering, structural design and seismic analysis of the US-APWR. Obayashi Corporation, headquartered in Tokyo, Japan, is one of the leading Japanese engineering, design and construction management companies, with approximately 9,500 employees at work in more than 10 countries. Obayashi Corporation has more than 30 years of experience in the construction of nuclear facilities including 27 nuclear power plants in Japan. In addition, all of the five PCCVs in Japan were designed and constructed by Obayashi Corporation.

1.4.2.2 Engineering Development Co., Ltd.

Engineering Development Co., Ltd., a specialist in nuclear engineering within the MHI Group, performs planning, design and analyses of nuclear power plants and associated facilities, and related software development. For the US-APWR, it mainly conducts analyses and design activities regarding the reactor and the primary system.

1.4.2.3 Washington Division of URS Corporation

Washington Division of URS Corporation has entered into a contract with MHI to provide consulting assistance in the preparation of the DCD for US-APWR. It has decades of experience in the reactor projects including being the engineer/constructor of 49 nuclear power plants around the world.

1.4.3 Combined License Information

COL 1.4(1) *The COL Applicant is to identify major agents, contractors, and participants for the COL application development, construction, and operation.*

1.5 Requirements for Further Technical Information

This section provides further technical information on verification and confirmation tests of certain unique design features of the US-APWR.

1.5.1 Advanced Accumulator Scale Test

In the conventional PWR plant, the functions of the emergency core cooling system (ECCS) during a LOCA are assigned to three subsystems: the accumulator system, the low head injection system, and the high head injection system. In the US-APWR, the advanced accumulator, which shifts the flow rate from large flow to small flow automatically, has been incorporated into the safety system design. The function of the low head injection system is assigned to the accumulator system; therefore, the low head injection system has been eliminated resulting in the simplification of the ECCS configuration.

The advanced accumulator has a flow damper, primarily consisting of a stand pipe and a vortex chamber. When the advanced accumulator water level is above the top of the standpipe, water enters the vortex damper through both inlets at the top of the standpipe and at the side of the vortex damper and thus it injects water at a large flow rate. When the water level drops below the top of the standpipe, water enters the vortex damper only through the side inlet and thus vortex formation in the vortex chamber achieves the small flow injection.

In order to verify this unique design for the APWR, four kinds of test were performed: 1/8.4, 1/3.5, 1/5, and full height 1/2 scale model tests. In these tests, visualization tests were conducted to confirm flow rate switching, vortex formation, and prevention of gas entraining into the vortex chamber at the end of large flow, as well as injection tests to provide performance data required for quantitative evaluation of advanced accumulator flow.

The core output and the size of the reactor vessel are the main parameters to determine advanced accumulator design requirements and specifications, therefore the US-APWR advanced accumulators are the same as those of the APWR. For that reason, the confirmation test program of the advanced accumulator previously performed for the APWR is applicable to the US-APWR. This test program was conducted as a joint study among five Japanese utilities and MHI, from September 1994 to September 1996.

The detailed description of the advanced accumulator is given in Reference 1.5-1.

1.5.2 Other tests for unique design features in US-APWR

1.5.2.1 Reactor Internals

1.5.2.1.1 APWR Reactor Internal Scale Model Test

The APWR reactor internals, including those of the US-APWR, have ring block type neutron reflectors instead of the baffle structures of current 4-loop plants. To verify the integrity of this unique type of reactor internal design for flow induced vibration, an

APWR reactor internal scale model test was performed at room temperature using 1/5 scale model reactor vessel and internals which simulated a 12 ft core type APWR. The results of this test are applied for verification of the analysis methodology in the FIV assessment program.

Further information of the test is reported in Reference 1.5-2.

1.5.2.1.2 Reactor Vessel Lower Plenum 1/7 Scale Model Flow Test

The US-APWR reactor internals has been designed for a core consisting of 257 fuel assemblies of 14 ft length. The design of the reactor internals, which includes integrated lower core support system, has been proven through considerable operating experiences as the standard design of 14 ft core reactors in USA. Although there is no operating experience with 257 fuel assemblies, and the structure in the vessel lower plenum is simplified, the geometries of the lower plenum are similar to those of current 4-loop PWRs. Thus hydraulic and vibration characteristics are also similar to those of current PWR.

A reactor vessel lower plenum 1/7 scale model flow test was performed to obtain the following data related to the lower plenum design configuration and to confirm the design:

- Hydraulic characteristics such as core inlet flow distribution

- FIV of the structures in the lower plenum

The report on the test is given in Reference 1.5-3.

1.5.2.1.3 Neutron Reflector Reflooding Confirmation Test

The neutron reflector (NR) reflooding test is a separate test focused on the thermal hydraulic phenomena of a single neutron reflector flow hole with 1/1 scale in a large break loss of coolant accident (LBLOCA).

The test and calculation results were presented as a technical report that provided the additional information to the topical report (Reference 1.5-4), describing the code applicability for the LBLOCA.

1.5.2.2 Digital Instrumentation and Control System

Verification and validation (V&V) of the safety digital Instrumentation and control (I&C) system and the human system interface (HIS) system have been already conducted in Japan, and detailed descriptions of those tests given as topical reports in Reference 1.5-5 through 1.5-8. As described in these topical reports, the safety I&C system consists of a digital platform, four train redundancy and safety video display units. The I&C system has a diverse actuation system (DAS) as a diversity of a digital platform.

The platform and integrated I&C and HSI systems have been qualified for service as safety equipment according to the guidelines of IEEE 7-4.3.2 and NUREG-0711 guidelines.

In order to evaluate whether the US-APWR design conforms to human factor engineering (HFE) design principles and enables plant personnel to successfully perform their tasks to achieve plant safety and other operational goals based on the V&V methodology, the HSI system V&V for plant personnel in USA will be conducted in 2008 and the summary results will be issued as a technical report in 2008.

1.5.2.3 Gas Turbine Generator

The class 1E gas turbine generator (Class 1E GTG) is used as emergency power source for the US-APWR.

A class 1E GTG has not been used in conventional PWR plants for emergency power source. However, Class 1E GTG has some advantages compared with diesel generators from view points of reliability and maintenance. A class 1E GTG qualification program is planned. This Class 1E qualification program consists of the following items:

- Evaluation of basic specification (electrical and mechanical)
- Evaluation of reliability
- Evaluation of seismic capability
- Verification of compliance with codes, regulations and standards
- Testing in accordance with IEEE 387 and other regulations or standards

Detail of this Class 1E qualification program, including schedule, are provided in the technical report issued on Nov.2007 (Reference 1.5-9). This qualification program was started in early 2008 and its results will be submitted as a technical report.

1.5.3 Combined License Information

No additional information is required to be provided by a COL applicant in connection with this section..

1.5.4 References

1.5-1 The Advanced Accumulator, MUAP-07001-P (Proprietary) and MUAP-07001-NP (Non-Proprietary), Revision 1, Mitsubishi Heavy Industries, Ltd., Feburary 2007.

1.5-2 APWR Reactor Internals 1/5 Scale Model Flow Test Report, MUAP-07023, Mitsubishi Heavy Industries, Ltd., December 2007.

1.5-3 US-APWR Reactor Vessel Lower Plenum 1/7 Scale Model Flow Test Report, MUAP-07022, Mitsubishi Heavy Industries, Ltd., June 2008.

1.5-4 Large Break LOCA Code Applicability Report for US-APWR, MUAP-07011-P (Proprietary) and MUAP-07011-NP (Non-Proprietary), Revision 0, Mitsubishi Heavy Industries, Ltd., July 2007.

1.5-5 Safety I&C System Description and Design Process, MUAP-07004-P (Proprietary) and MUAP-07004-NP (Non-Proprietary), Revision 1, Mitsubishi Heavy Industries, Ltd., July 2007.

1.5-6 Safety System Digital Platform -MELTAC-, MUAP-07005-P (Proprietary) and MUAP-07005-NP (Non-Proprietary), Revision 2, Mitsubishi Heavy Industries, Ltd., August 2008.

1.5-7 Defense-in-Depth and Diversity, MUAP-07006-P (Proprietary) and MUAP-07006-NP (Non-Proprietary), Revision 2, Mitsubishi Heavy Industries, Ltd., June 2008.

1.5-8 HIS System Description and HFE Process, MUAP-07007-P (Proprietary) and MUAP-07007-NP (Non-Proprietary), Revision 1, Mitsubishi Heavy Industries, Ltd., July 2007.

1.5-9 Qualification and Test Plan of Class 1E Gas Turbine Generator System, MUAP-07024, Revision 0, Mitsubishi Heavy Industries, Ltd., December 2007.

1.6 Material Referenced

A list of topical reports incorporated by reference as part of the US-APWR DC application is shown in Table 1.6-1.

Table 1.6-1 Material Referenced

Report Number[1]	Title	DCD Section Number
MUAP-07001-P MUAP-07001-NP	The Advanced Accumulator, Revision 1, Feburaly 2007	1.5.4, 6.3.7
MUAP-07004-P MUAP-07004-NP	Safety I&C System Description and Design Process, Revision 1, July 2007	1.5.4, 7.1.5, 7.2.5, 7.3.5, 7.4.5, 7.5.5, 7.6.5, 7.7.5, 7.8.5. 7.9.5, 14.2.14
MUAP-07005-P MUAP-07005-NP	Safety System Digital Platform -MELTAC-, Revision 2, August 2008	1.5.4, 7.1.5, 7.2.5, 7.3.5, 7,7,5, 7.8.5, 7.9.5
MUAP-07006-P MUAP-07006-P	Defense-in-Depth and Diversity, Revision 2, June 2008	1.5.4, 7.1.5, 7.3.5, 7.8.5, 7.9.5
MUAP-07007-P MUAP-07007-NP	HSI System Description and HFE Process, Revision 1, July 2007	1.5.4, 7.1.5, 7.5.5, 7.6.5, 18.1.7, 18.2.5, 18.3.5, 18.4.5, 18.7.5, 18.8.5, 18.9.5, 18.10.5
MUAP-07008-P MUAP-07008-NP	Mitsubishi Fuel Design Criteria and Methodology, May 2007	4.2.6, 4.3.6, 4.4.8, 15.0.5, 15.4.11,
MUAP-07009-P MUAP-07009-NP	Thermal Design Methodology, May 2007	4.4.8, 15.0.5, 15.1.7, 15.2.10, 15.3.6, 15.4.11, 15.6.7
MUAP-07010-P MUAP-07010-NP	Non-LOCA Methodology, July 2007	6.2.9, 15.0.5, 15.1.7, 15.2.10, 15.3.6, 15.4.11, 15.5.4, 15.6.7
MUAP-07011-P MUAP-07011-NP	Large Break LOCA Code Applicability Report for US-APWR, July 2007	1.5.4, 6.3.7, 15.0.5, 15.6.7
MUAP-07012-P MUAP-07012-NP	LOCA Mass and Energy Release Analysis Code Applicability Report for US-APWR, Revision 2, May 2008	6.2.9
MUAP-07013-P MUAP-07013-NP	Small Break LOCA Methodology for US-APWR, July 2007	6.2.9, 15.0.5, 15.6.7
PQD-DH-19005	Quality Assurance Program (QAP) Description For Design Certification of the US-APWR, Revision 1, October 2007	17.5.5, 18.1.7, 18.10.5

NOTE: -P(**p**roprietary) , -NP(**n**on-**p**roprietary)

1.7 Drawings and Other Detailed Information

Additional US-APWR drawings and other detailed technical information are provided in the DCD chapters that follow, for the topics discussed in those chapters. Table 1.7-1 contains a list of all instrument and control functional diagrams and electrical one-line diagrams. A list of system drawings is shown in Table 1.7-2.

The legend for electrical power diagrams is provided in Figure 1.7-1. Figure 1.7-2 is the legend for instrument and control function diagrams.

The legend for piping and instrumentation diagrams (P&IDs) are provided in Figure 1.7-3 through Figure 1.7-5.

An equipment numbering system is applied to all of the equipment in the US-APWR. An example of the equipment numbering system format for the steam generator on train A is shown below.

STD – RCS – MHX – 001A – S

Where: STD is the Plant Designator for standard plant
RCS is the System Code for reactor coolant system
MHX is the Equipment Function Code for heat exchanger
001 is the Serial Number
A is the Suffix
S is the Safety Designator (Safety-Related (S), Non Safety-Related (N))

The full equipment number is shown in all drawings. However, to reduce the characters in the DCD, the Plant Designator and Safety Designators are omitted.

Table 1.7-1 I&C Functional and Electrical One-line Diagrams

Figure Number	Subject
7.2-2	Functional Logic Diagram for Reactor Protection and Control System
7.6-1	Interlocks for CS/RHR Pumps Suction C/V Isolation Valves
7.6-2	Interlocks for RHR Discharge Line Containment Isolation Valve
7.6-3	Interlocks for Containment Spray Header Containment Isolation Valve
7.6-4	Interlocks for Primary Makeup Water Stop Valve
7.6-5	Interlocks for Accumulator Discharge Valve
7.6-6	Interlocks for CCW Header Tie Line Isolation Valves
8.1-1	Simplified One Line Diagram Electric Power System
8.3-1-1	Onsite AC electrical distribution system
8.3-1-2	Logic diagrams
8.3-1-3	120V AC I&C power supply panels
8.3-2-1	DC Power Distribution System

Table 1.7-2 System Drawings
(sheet 1 of 3)

Figure Number	Subject
3E-1	Reactor Coolant System Flow Diagram (1/2)
3E-2	Reactor Coolant System Flow Diagram (2/2)
3E-3	Chemical and Volume Control System Flow Diagram (1/4)
3E-4	Chemical and Volume Control System Flow Diagram (2/4)
3E-5	Chemical and Volume Control System Flow Diagram (3/4)
3E-6	Chemical and Volume Control System Flow Diagram (4/4)
3E-7	Safety Injection System Flow Diagram (Sheet 3 of 4)
3E-8	Emergency Feed water System flow Diagram (1/2)
3E-9	Emergency Feed water System flow Diagram (2/2)
3E-10	Main Feed Water System Flow Diagram
3E-11	Main Steam System Flow Diagram (1/2)
3E-12	Main Steam System Flow Diagram (2/2)
3E-13	Steam Generator Blowdown System Flow Diagram
5.1-1	Reactor Coolant System Schematic Flow Diagram
5.1-2	Reactor Coolant System Piping and Instrumentation Diagram
5.4.7-1	Residual Heat Removal System Flow Diagram
5.4.7-2	Residual Heat Removal System P&ID)
6.2.2-1	Flow Diagram of the containment Spray System
6.2.5-1	Containment Hydrogen Monitoring and Control System Schematic
6.3-1	Emergency Core Cooling System Schematic Flow Diagram
6.3-2	ECCS Piping and Instrumentation Diagram
6.3-7	Refueling Water Storage System
6.3-12	NaTB Solution Transfer Piping Diagram
6.3-13	ECCS Process Flow Diagram (R/V Injection)
6.3-14	ECCS Process Flow Diagram (Simulataneous RV and hot leg injection)
6.4-2	MCR HVAC System (Normal Operation Mode)
6.4-3	MCR HVAC System (Emergency Pressurization Mode)
6.4-4	MCR HVAC System (Emergency Isolation Mode)
6.5-1	Annulus Emergency Exhaust System Simplified Flow Diagram
9.1.3-1	Schematic of Spent Fuel Pit Purification and Cooling System (Cooling Portion)

Table 1.7-2 System Drawings
(sheet 2 of 3)

Figure Number	Subject
9.1.3-2	Schematic of Spent Fuel Pit Purification and Cooling System (Purification Portion)
9.2.1-1	Essential Service Water System
9.2.2-1	CCW Flow Diagram
9.2.4-1	Potable Water Flow Diagram
9.2.6-1	Condensate Storage Facilities System Flow Diagram
9.2.6-2	Primary Makeup Water System Flow Diagram
9.2.6-3	Demineralized Water System Flow Diagram
9.2.7-1	Essential Chilled Water System Flow Diagram
9.2.8-1	Turbine Component Cooling Water System Piping and Instrumentation Diagram
9.2.9-1	Non-Essential Service Water System Flow Diagram
9.3.1-1	Instrument Air Subsystem
9.3.1-2	Station Service Air Subsystem
9.3.2-1	PSS Flow Diagram
9.3.3-1	Equipment and Floor Drain System Flow Schematic
9.3.4-1	Chemical and Volume Control System Flow Diagram
9.4.1-1	MCR HVAC System Flow Diagram
9.4.3-1	Auxiliary Building HVAC System Flow Diagram
9.4.3-2	Non-Class 1E Electrical Room HVAC System Flow Diagram
9.4.3-3	Main Steam/Feed Water Piping Area HVAC System Flow Diagram
9.4.3-4	Technical Support Center (TSC) HVAC System Flow Diagram
9.4.4-1	Turbine Building Area Ventilation System Flow Diagram
9.4.5-1	Annulus Emergency Exhaust System Flow Diagram
9.4.5-2	Class 1E Electrical Room HVAC system Flow Diagram
9.4.5-3	Safeguard Component Area HVAC System Flow Diagram
9.4.5-4	Emergency Feedwater Pump Area HVAC System Flow Diagram
9.4.5-5	Safety Related Component Area HVAC System Flow Diagram
9.4.6-1	Containment Ventilation System Flow Diagram
9.5.4-1	Gas Turbine Generator Fuel Oil Storage and Transfer System
9.5.6.1	Gas Turbine Generator Starting System
9.5.7-1	Gas Turbine Lubrication System
10.1-1	Overall System Flow Diagram

Table 1.7-2 System Drawings
(sheet 3 of 3)

Figure Number	Subject
10.3-1	Main Steam Supply System Piping and Instrumentation Diagram (1/4)
10.3-2	Main Steam Supply System Piping and Instrumentation Diagram (2/4)
10.3-3	Main Steam Supply System Piping and Instrumentation Diagram (3/4)
10.3-4	Main Steam Supply System Piping and Instrumentation Diagram (4/4)
10.4.2-1	Main Condenser Evacuation System Piping and Instrumentation Diagram
10.4.3-1	Gland Seal System Piping and Instrumental Diagram
10.4.5-1	Circulating Water System Piping and Instrumental Diagram
10.4.6-1	Condensate Polishing System Piping and Instrumentation Diagram
10.4.7-1	Condensate and Feedwater System Piping and Instrumentation Diagram (1/4)
10.4.7-2	Condensate and Feedwater System Piping and Instrumentation Diagram (2/4)
10.4.7-3	Condensate and Feedwater System Piping and Instrumentation Diagram (3/4)
10.4.7-4	Condensate and Feedwater System Piping and Instrumentation Diagram (4/4)
10.4.8-1	Steam Generator Blowdown System Piping and Instrumentation Diagram (1/2)
10.4.8-2	Steam Generator Blowdown System Piping and Instrumentation Diagram (2/2)
10.4.9-1	Emergency Feedwater System Piping and Instrumental Diagram (1/2)
10.4.9-2	Emergency Feedwater System Piping and Instrumental Diagram (2/2)
10.4.10-1	Secondary Side Chemical Injection System Piping and Instrumentation Diagram
10.4.11-1	Auxiliary Steam Supply System Piping and Instrumentation Diagram
11.2-1	Liquid Waste Processing System Process Flow Diagram
11.3-1	Gas Management System Piping and Instrumentation Diagram (Sheet 1 of 3)
11.4-1	Process Flow Diagram of SWMS Dry Active Waste and Spent Filter Handling Sub-system
11.4-2	Process Flow Diagram of SWMS Spent Resin and Charcoal Handling Sub-System
11.4-3	Process Flow Diagram of SWMS Oil and Sludge Handling System

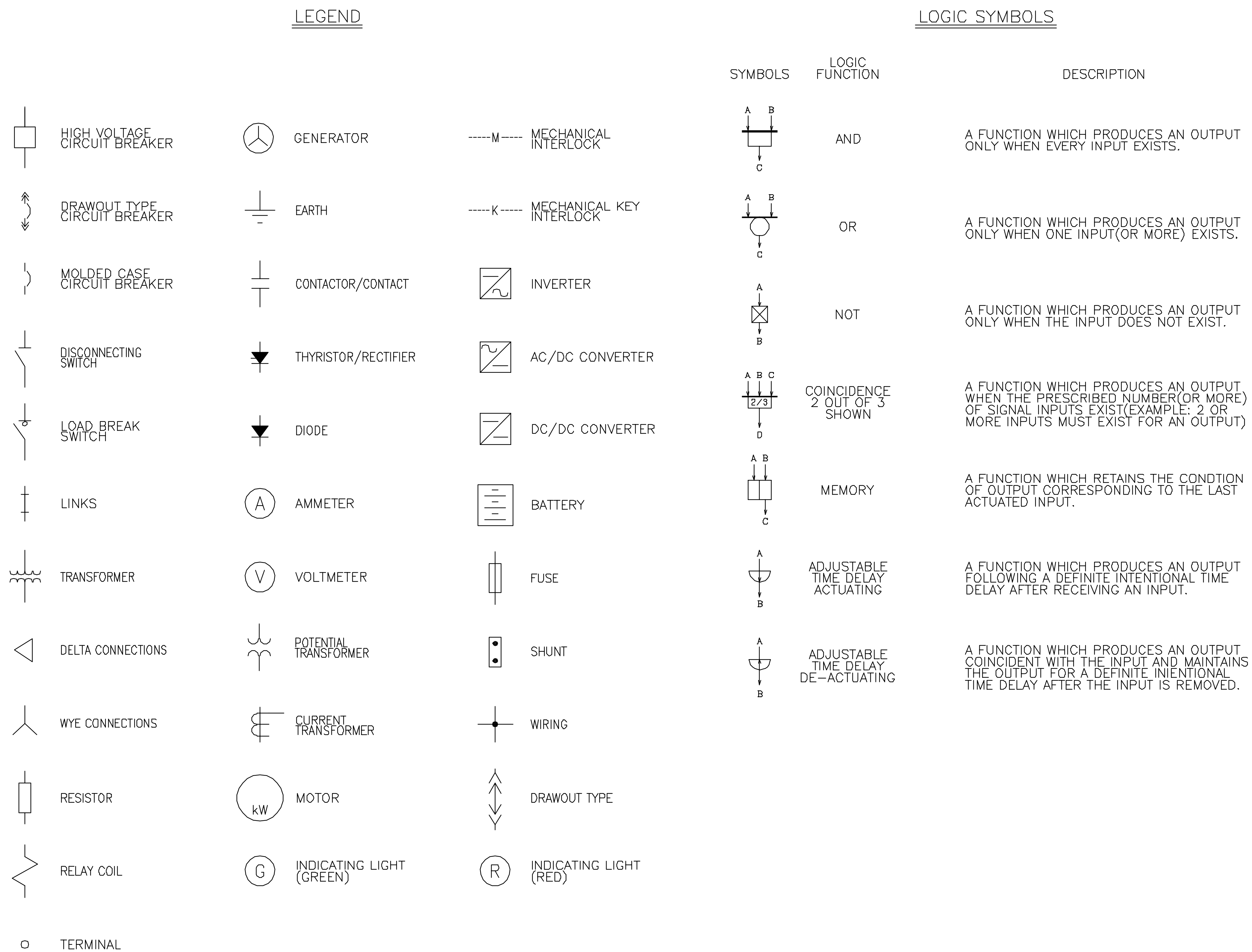

Figure 1.7-1 Legend for Electrical Power Diagrams

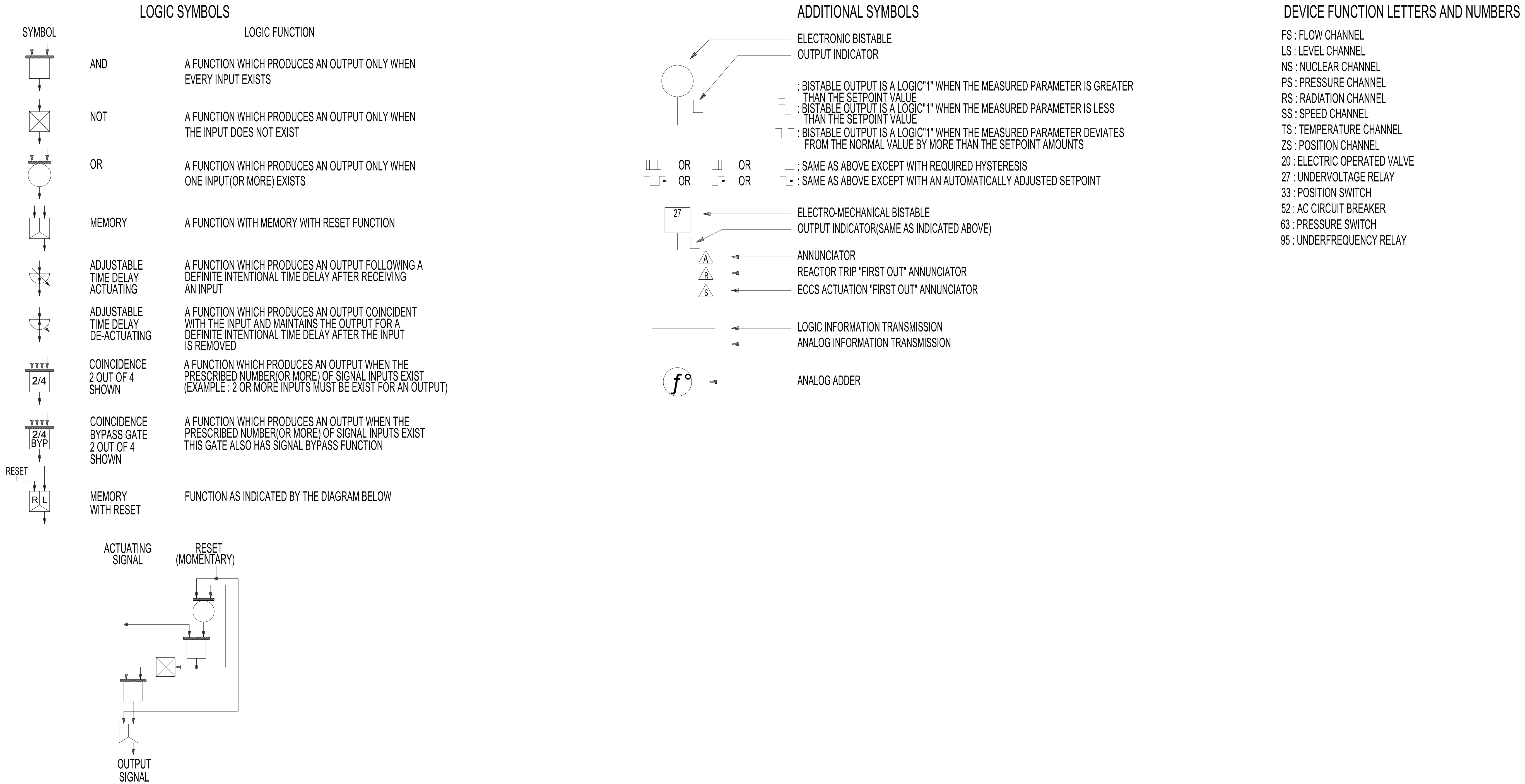

Figure 1.7-2 Legend for Instrument and Control Function diagrams

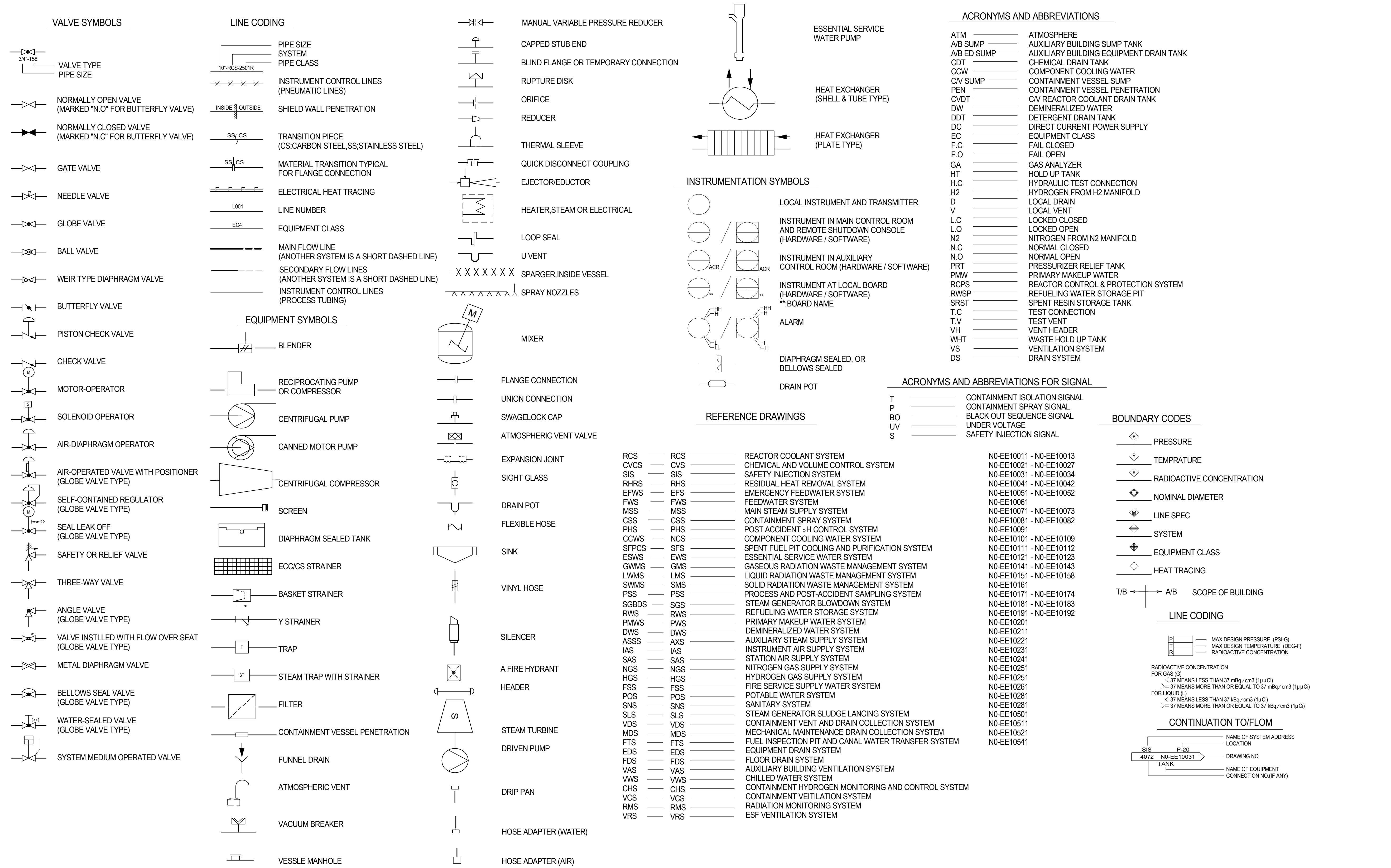

Figure 1.7-3 Legend for Piping and Instrumentation Diagrams of Primary System

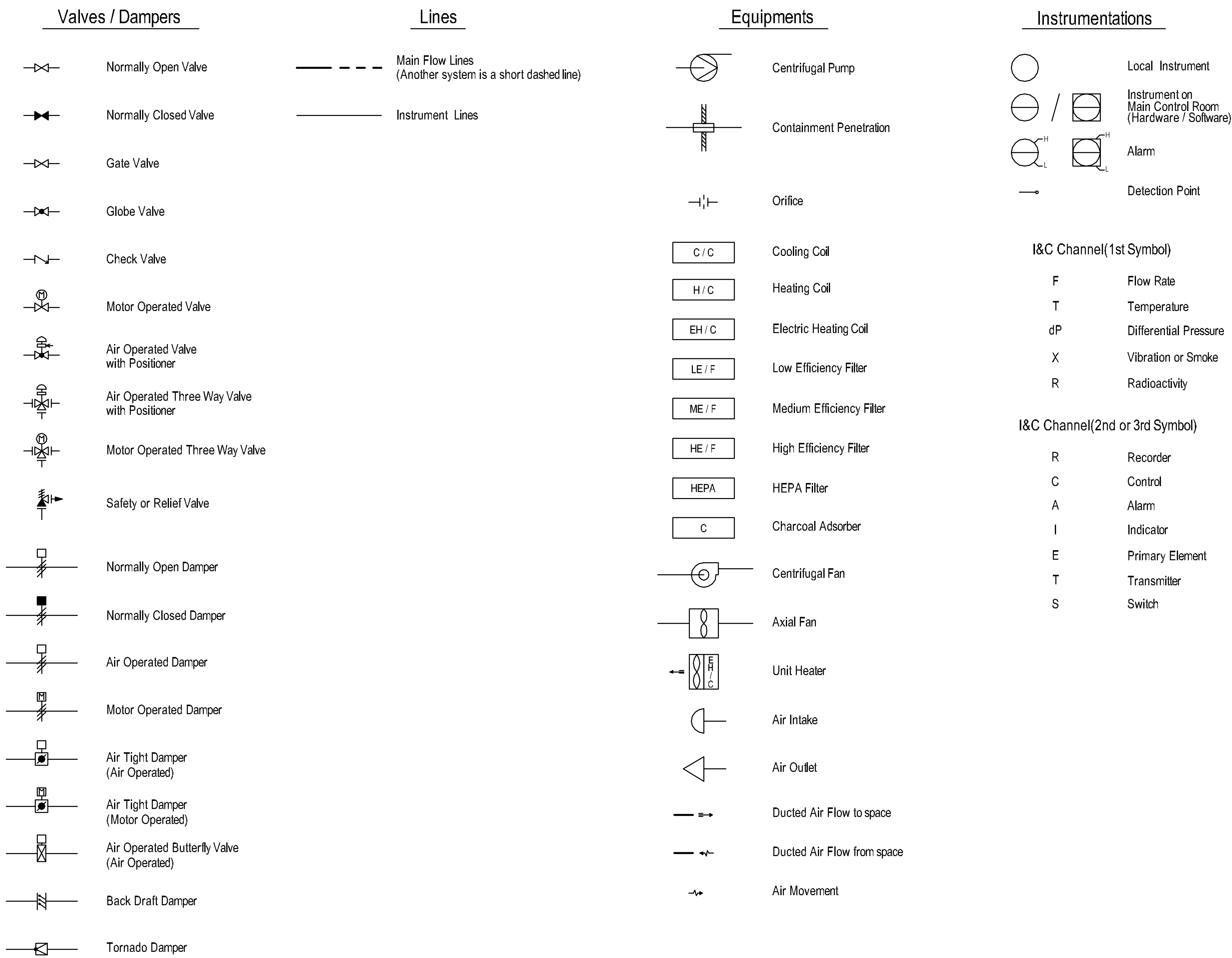

Figure 1.7-4 Legend for Piping and Instrumentation Diagrams of HVAC System

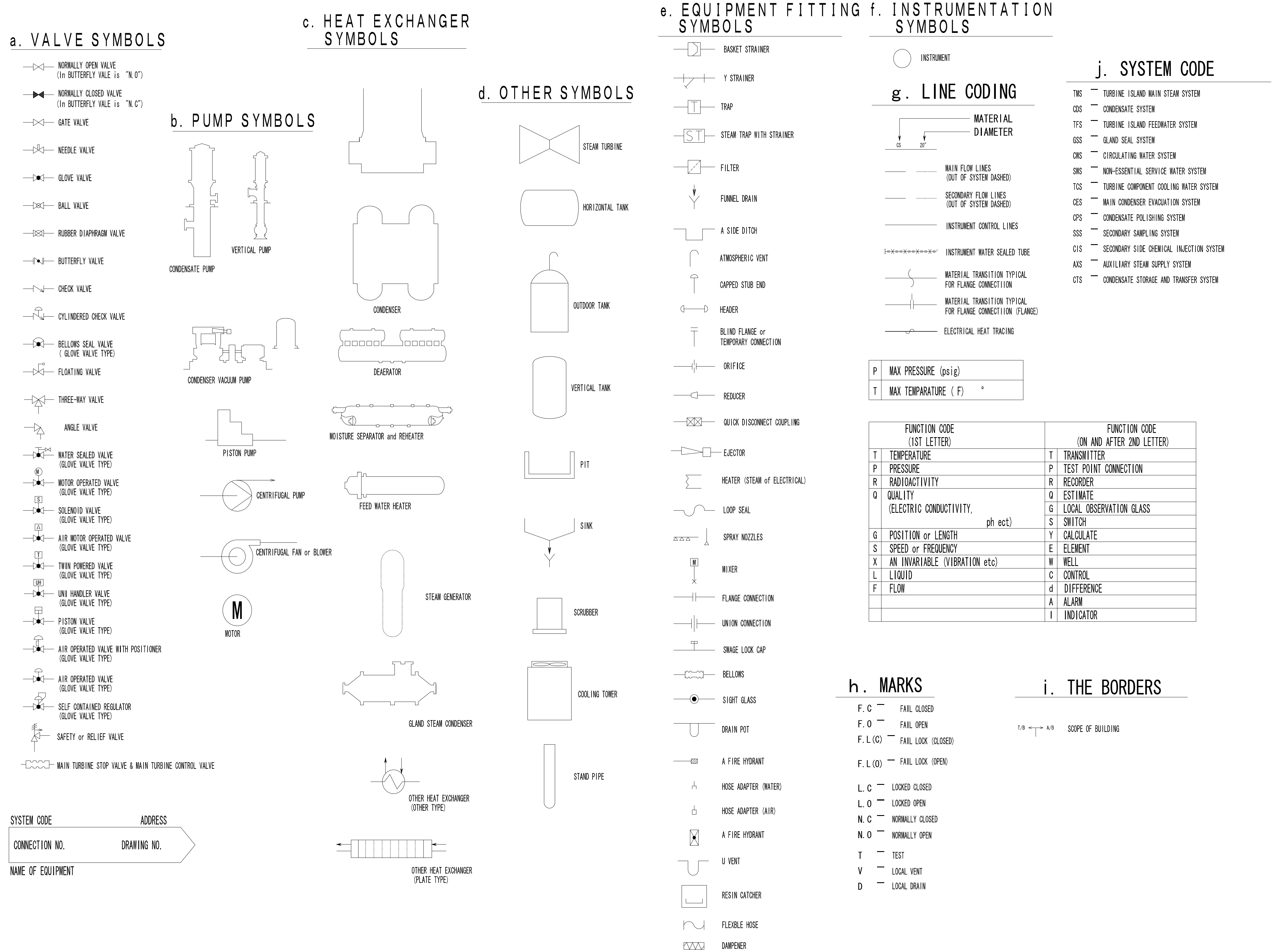

Figure 1.7-5 Legend for Piping and Instrumentation Diagrams of Secondary System

1.8 Interfaces for Standard Design

10 CFR 52.47, "Contents of applications: technical information", requires identification of the interface requirements to be met by those portions of the plant for which the application does not seek certification. As allowed by the regulations, conceptual designs for systems that are not part of the US-APWR standard design are included in the DCD for purposes of allowing the NRC to evaluate the overall acceptability of the design. However, the final details of these conceptual designs are subject to change due to site-specific conditions.

This section identifies the significant interfaces between the US-APWR standard plant design and the Combined License applicant.

The US-APWR standard plant design consists of several buildings as discussed in subsection 1.2.1.7, the equipment located in those buildings, and associated yard structures such as electrical equipment and tanks. This standard scope of design includes the entire nuclear island and all safety systems that would be required for construction of the US-APWR at a nuclear power plant site. The standard site plan for the US-APWR design included in this application for design certification is shown in Figure 1.2-1.

Items in the "Description" column of Table 1.8-1 are site-specific and are outside the scope of the US-APWR standard plant design. The table includes a description of each interface and the DCD section in which it is discussed. Combined License applicants referencing the US-APWR certified design are to be required to demonstrate that interface requirements are satisfied. The interface items are divided into two types:

- System Interface – Portion of a system that must be added to the standard design package to complete the implementation of the US-APWR at a specific site. Generally, a system interface can be accomplished via and intertie between the standard and site-specific portions of a system, such as by piping or electrical cable.

- Site Feature Interface – Construction of a non-system feature that must be added to the standard design package to complete the implementation of the US-APWR at a specific site. Examples would include general site improvements, or a building that is not included in the standard design, such as the administrative building.

Per the language of 10CFR52, the tabulated items are significant interfaces, not a comprehensive inventory of all features outside the standard scope of supply. Additionally, the reader can refer to Table 1.8-2 for a master listing of all COL items in the DCD.

1.8.1 Summary of Combined License Information Items

1.8.1.1 Consolidated Combined License Information Items for the Entire Design Control Document

Table 1.8-2 presents an accumulation of all COL items from all of the chapters of this DCD, including a description of each item and the section in which it can be found.

The COL Applicant is to provide the cross-reference identifying specific FSAR sections that address each COL information item from the DCD. The cross-reference includes the status of each COL information item that can be resolved and that can not be resolved as part of the COLA.

In addition, the COL Applicant is to identify a plant specific departures from the DCD (if necessary) and to provide the summary of conformance with site parameters and advancement of conceptual designs since the DCD.

1.8.2 Combined License Information

COL 1.8(1) *The COL Applicant is to demonstrate that the interface requirements established for the design have been met.*

COL 1.8(2) *The COL Applicant is to provide the cross-reference identifying specific FSAR sections that address each COL information item from the DCD*

COL 1.8(3) *The COL Applicant is to provide a summary of plant specific departures from the DCD, and conformance with site parameters.*

Table 1.8-1 Significant Site Specific Interfaces with the Standard US-APWR Design (sheet 1 of 2)

Interface Number	Interface	Interface Type	Description of Items Considered to be Outside the Standard Scope of Design	DCD Section
1	Circulating Water System	Site Feature Interface	The site-specific final system configuration and system design parameters. A typical design of the circulating water system is presented in the DCD.	10.4.5
2	Essential Service Water System and Ultimate Heat Sink	System Interface	Portions of the ESWS outside the US-APWR buildings. The UHS, safety-related system is provided to remove the heat transferred from the ESWS during normal operation, design basis event and safe shutdown. The UHS is the safety-related water source to ESWS.	9.2.1 9.2.5
3	Deleted	-	-	-
4	Electric Power	System Interface	Offsite power transmission system outside the high voltage terminals of the main and reserve transformers. Location and design of the main switchyard area and the equipment located therein, as well as design details such as voltage level for the main step-up transformers. A site-specific interface between the certified design and the local electrical grid is addressed in the DCD.	8.1, 8.2
5	Deleted			
6	Potable and Sanitary Water Systems	System Interface	Portions of Potable and Sanitary Water Systems (PSWS) outside the standard US-APWR buildings. The potable water system provides water supply and distribution fit for human consumption, and the sanitary drain system provides collection of sanitary wastewater.	9.2.4
7	Communications Systems	System Interface	Communications systems and equipment outside the buildings identified in the standard US-APWR design. This DCD is based upon the COL applicant providing adequate external communications, including interfaces with the local telecommunications provider, and communication links between the on-site system and other on-site and offsite facilities such as the Emergency Operations Facility and the training simulator.	9.5.2

Table 1.8-1 Significant Site Specific Interfaces with the Standard US-APWR Design (sheet 2 of 2)

Interface Number	Interface	Interface Type	Description of Items Considered to be Outside the Standard Scope of Design	DCD Section
8	Administrative, Emergency Response and Training Facilities	Site Feature Interface	Location and design of the COL applicant’s administrative building, training structures including the training simulator, and the Emergency Response Facility.	7.5.1 9.5.2 13.3
9	Security Systems	System Interface	Site security/surveillance systems, such as surveillance cameras, video displays, security detection sensors, communications, access control, etc.	13.6
10	General Site Improvements	Site Feature Interface	Landscaping features, roadways, walkways, security fences and barricades, traffic control barriers, etc., that are not part of the standard US-APWR building designs.	1.2 13.6
11	Fire Protection	System Interface	Fire protection features such as fire water supply facilities, sprinkler systems, smoke and fire detection devices, and fire alarm systems. A safety-related water source supplied to the seismic standpipe system.	9.5.1
12	Effluent Monitoring and Sampling	Site Feature Interface	Effluent monitoring and sampling systems and features required to monitor levels of activity in plant effluent released to the environment.	11.5
13	Deleted			
14	Compressed Gases	System Interface	Supply portions of oxygen, hydrogen, nitrogen and carbon dioxide systems. Supply lines from yard area connect to distribution lines in US-APWR buildings necessary for operation of components.	9.3.1

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 1 of 44)

COL ITEM NO.	COL ITEM
COL 1.1(1)	*The COL Applicant is to provide scheduled completion date and estimated commercial operation date of nuclear power plants referencing the US-APWR standard design.*
COL 1.1(2)	*The Combined License (COL) Applicant is to identify the actual plant location.*
COL 1.2(1)	*The COL Applicant is to develop a complete and detailed site plan in the site-specific licensing process.*
COL 1.4(1)	*The COL Applicant is to identify major agents, contractors, and participants for the COL application development, construction, and operation.*
COL 1.8(1)	*The COL Applicant is to demonstrate that the interface requirements established for the design have been met.*
COL 1.8(2)	*The COL Applicant is to provide the cross-reference identifying specific FSAR sections that address each COL information item from the DCD*
COL 1.8(3)	*The COL Applicant is to provide a summary of plant specific departures from the DCD, and conformance with site parameters.*
COL 1.9(1)	*The COL Applicant is to address an evaluation of the applicable RG, SRP, Generic Issues including Three Mile Island (TMI) requirements, and operational experience for the site-specific portion and operational aspect of the facility.*
COL 2.1(1)	*The COL Applicant is to describe the site geography and demography including the specified site parameters.*
COL 2.2(1)	*The COL Applicant is to describe nearby industrial, transportation, and military facilities within 5 miles of the site, or at greater distances as appropriate based on their significance. The COL Applicant is to establish the presence of potential hazards, determine whether these accidents are to be considered as DBEs, and the design parameters related to the accidents determined as DBEs.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 2 of 44)

COL ITEM NO.	COL ITEM
COL 2.3(1)	*The COL Applicant, whether the plant is to be sited inside or outside the continental US, is to provide site-specific pre-operational and operational programs for meteorological measurements, and is to verify the site-specific regional climatology and local meteorology are bounded by the site parameters for the standard US-APWR design or demonstrate by some other means that the proposed facility and associated site-specific characteristics are acceptable at the proposed site.*
COL 2.3(2)	*The COL Applicant is to provide conservative factors as described in SRP 2.3.4 (Reference 2.3-2). If a selected site will cause excess to the bounding* χ*/Q values, then the COL Applicant is to demonstrate how the dose reference values in 10 CFR 52.79(a)(1)(vi) (Reference 2.3-3) and the control room dose limits in 10 CFR 50, Appendix A, General Design Criteria 19 (Reference 2.3-4) are met using site-specific* χ*/Q values.*
COL 2.3(3)	*The COL Applicant is to characterize the atmospheric transport and diffusion conditions necessary for estimating radiological consequences of the routine release of radioactive materials to the atmosphere, and provide realistic estimates of annual average* χ*/Q values and D/Q values as described in SRP 2.3.5 (Reference 2.3-5).*
COL 2.4(1)	*The COL Applicant is to provide sufficient information to verify that hydrologic-related events will not affect the safety-basis for the US-APWR.*
COL 2.5(1)	*The COL Applicant is to provide sufficient information regarding the seismic and geologic characteristics of the site and the region surrounding the site.*
COL 3.1(1)	*The COL Applicant is to provide a design that allows for the appropriate inspections and layout features of the ESWS.*
COL 3.2(1)	*Deleted*
COL 3.2(2)	*Deleted*
COL 3.2(3)	*Deleted*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 3 of 44)

COL ITEM NO.	COL ITEM
COL 3.2(4)	*The COL Applicant is to identify the site-specific, safety-related systems and components that are designed to withstand the effects of earthquakes without loss of capability to perform their safety function; and those site-specific, safety-related fluid systems or portions thereof; as well as the applicable industry codes and standards for pressure-retaining components.*
COL 3.2(5)	*The COL Applicant is to identify the equipment class and seismic category of the site-specific, safety-related and non safety-related fluid systems, components (including pressure retaining), and equipment as well as the applicable industry codes and standards.*
COL 3.3(1)	*The COL Applicant is responsible for verifying the site-specific basic wind speed is enveloped by the determinations in this section.*
COL 3.3(2)	*These requirements also apply to seismic category I structures provided by the COL Applicant. Similarly, it is the responsibility of the COL Applicant to establish the methods for qualification of tornado effects to preclude damage to safety-related SSCs.*
COL 3.3(3)	*It is the responsibility of the COL Applicant to assure that site-specific structures and components not designed for tornado loads will not impact either the function or integrity of adjacent safety-related SSCs, or generate missiles having more severe effects than those discussed in Subsection 3.5.1.4.*
COL 3.3(4)	*The COL Applicant is to provide the wind load design method and importance factor for site-specific category I and category II buildings and structures.*
COL 3.3(5)	*The COL Applicant is to note the vented and unvented requirements of this subsection to the site-specific category I buildings and structures.*
COL 3.4(1)	*The COL Applicant is to address the site-specific design of plant grading and drainage.*
COL 3.4(2)	*The COL Applicant is to demonstrate the DBFL bounds their specific site, or is to identify and address applicable site conditions where static flood level exceed the DBFL and/or generate dynamic flooding forces.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 4 of 44)

COL ITEM NO.	COL ITEM
COL 3.4(3)	*Site-specific flooding hazards from engineered features, such as from cooling water system piping, is to be addressed by the COL Applicant.*
COL 3.4(4)	*The COL Applicant is to address any additional measures below grade to protect against exterior flooding and the intrusion of ground water into seismic category I buildings and structures.*
COL 3.4(5)	*The COL Applicant is to identify and design, if necessary, any site-specific flood protection measures such as levees, seawalls, floodwalls, site bulkheads, revetments, or breakwaters per the guidelines of RG 1.102 (Reference 3.4-3), or dewatering system if the plant is not built above the DBFL.*
COL 3.4(6)	*The COL Applicant is to identify any site-specific physical models used to predict prototype performance of hydraulic structures and systems.*
COL 3.5(1)	*The COL Applicant is to have plant procedures in place prior to fuel load that specify unsecured equipment, including portable pressurized gas cylinders, located inside or outside containment and required for maintenance or undergoing maintenance is to be removed from containment prior to operation, moved to a location where it is not a potential hazard to SSCs important to safety, or seismically restrained to prevent it from becoming a missile.*
COL 3.5(2)	*The COL Applicant is to commit to actions to maintain P1 within this acceptable limit as outlined in RG 1.115,"Pritectuib Against Low-Tragectory Turbine Missiles" (Reference 3.5-6) and SRP Section 3.5.1.3, "Turbine Missiles" (Reference 3.5-7).*
COL 3.5(3)	*As described in DCD, Section 2.2, the COL Applicant is to establish the presence of potential hazards, except aircraft, which is reviewed in Subsection 3.5.1.6, and the effects of potential accidents in the vicinity of the site.*
COL 3.5(4)	*It is the responsibility of the COL Applicant to verify the site interface parameters with respect to aircraft crashes and air transportation accidents as described in Section 2.2.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 5 of 44)

COL ITEM NO.	COL ITEM
COL 3.5(5)	*The COL Applicant is responsible to evaluate site-specific hazards for external events that may produce missiles more energetic than tornado missiles, and assure that the design of seismic category I and II structures meet these loads.*
COL 3.5(6)	*The COL Applicant is responsible to assess the orientation of the T/G of this and other unit(s) at multi-unit site for the probability of missile generation using the evaluation of Subsection 3.5.1.3.2.*
COL 3.6(1)	*The COL Applicant is to identify the site-specific systems or components that are safety-related or required for safe shutdown that are located near high-energy or moderate-energy piping systems, and are susceptible to the consequences of these piping failures. The COL Applicant is to provide a list of site-specific high-energy and moderate-energy piping systems, which includes a description of the layout of all piping systems where physical arrangement of the piping systems provides the required protection, the design basis of structures and compartments used to protect nearby essential systems or components, or the arrangements to assure the operability of safety-related features where neither separation nor protective enclosures are practical. Additionally, the COL Applicant is to provide the failure modes and effect analyses that verifies the consequences of failures in site-specific high-energy and moderate-energy piping does not affect the ability to safely shut down the plant.*
COL 3.6(2)	*Deleted*
COL 3.6(3)	*Deleted*
COL 3.6(4)	*The COL Applicant is to implement the criteria for defining break and crack locations and configurations for site-specific high-energy and moderate-energy piping systems. The COL Applicant is to identify the postulated break location for site-specific high-energy and moderate-energy piping systems. The COL Applicant is to implement the appropriate methods to assure that as-built configuration of site-specific high-energy and moderate-energy piping systems is consistent with the design intent and provide as-built drawings showing component locations and support locations and types that confirms this consistency.*
COL 3.6(5)	*Deleted*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 6 of 44)

COL ITEM NO.	COL ITEM
COL 3.6(6)	*Deleted*
COL 3.6(7)	*Deleted*
COL 3.6(8)	*Deleted*
COL 3.6(9)	*Deleted*
COL 3.7(1)	*The COL Applicant is to confirm that the site-specific PGA at the basemat level control point of the CSDRS is less than or equal to 0.3 g.*
COL 3.7(2)	*The COL Applicant is to perform an analysis of the US-APWR standard plant seismic category I design to verify that the site-specific FIRS at the basemat level control point of the CSDRS are enveloped by the site-independent CSDRS.*
COL 3.7(3)	*It is the responsibility of the COL Applicant to develop analytical models appropriate for the seismic analysis of buildings and structures that are designed on a site-specific basis including, but not limited to, the following:* • *PSFSVs (seismic category I)* • *ESWPT (seismic category I)* • *UHSRS (seismic category I)*
COL 3.7(4)	*To prevent non-conservative results, the COL Applicant is to review the resulting level of seismic response and determine appropriate damping values for the site-specific calculations of ISRS that serve as input for the seismic analysis of seismic category I and seismic category II subsystems.*
COL 3.7(5)	*The COL Applicant is to assure that the horizontal FIRS defining the site-specific SSE ground motion at the bottom of seismic category I or II basemats envelope the minimum response spectra required by 10 CFR 50, Appendix S, and the site-specific response spectra obtained from the response analysis.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 7 of 44)

COL ITEM NO.	COL ITEM
COL 3.7(6)	*The COL Applicant is to develop site-specific GMRS and FIRS by an analysis methodology, which accounts for the upward propagation of the GMRS. The FIRS are compared to the CSDRS to assure that the US-APWR standard plant seismic design is valid for a particular site. If the FIRS are not enveloped by the CSDRS, the US-APWR standard plant seismic design is modified as part of the COLA in order to validate the US-APWR for installation at that site.*
COL 3.7(7)	*The COL Applicant is to determine the allowable dynamic bearing capacity based on site conditions, and to evaluate the bearing load to this capacity.*
COL 3.7(8)	*The soil properties nay be considered strain-independent for subgrade materials with initial shear wave velocities of 3.500 ft/s or higher, to be confirmed by the COL Applicant as part of the site-specific subsurface material investigations discussed in Section 2.5.4. However, the COL Applicant must institute dynamic testing to evaluate the strain-dependent variation of the material dynamic properties for site materials with initial shear wave velocities below 3,500 ft/s.*
COL 3.7(9)	*The COL Applicant is to assure that the design or location of any site-specific seismic category I SSCs, for example pipe or duct banks, will not expose those SSCs to possible impact due to the failure or collapse of non-seismic category I structures, or with any other SSCs that could potentially impact, such as heavy haul route loads, transmission towers, non safety-related storage tanks, etc.*
COL 3.7(10)	*It is the responsibility of the COL Applicant to further address structure-to-structure interaction if the specific site conditions can be important for the seismic response of particular US-APWR seismic category I structures, or may result in exceedance of assumed pressure distributions used for the US-APWR standard plant design.*
COL 3.7(11)	*Deleted*
COL 3.7(12)	*It is the responsibility of the COL Applicant to design seismic category I below- or above-ground liquid-retaining metal tanks such that they are enclosed by a tornado missile protecting concrete vault or wall, in order to confine the emergency gas turbine fuel supply.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 8 of 44)

COL ITEM NO.	COL ITEM
COL 3.7(13)	*The COL Applicant is to set the value of the OBE that serves as the basis for defining the criteria for shutdown of the plant, according to the site specific conditions.*
COL 3.7(14)	*The COL Applicant is to determine from the site-specific geological and seismological conditions if multiple US-APWR units at a site will have essentially the same seismic response, and based on that determination, choose if more than one unit is provided with seismic instrumentation at a multiple-unit site.*
COL 3.7(15)	*Deleted*
COL 3.7(16)	*The COL Applicant is to verify the site-specific applicability of these monitors, and determine if there is a need for the installation of additional instrumentation for the measurement of the free-field ground motion based on conditions and requirements specific to the site.*
COL 3.7(17)	*Deleted*
COL 3.7(18)	*Deleted*
COL 3.7(19)	*The COL Applicant is to identify the implementation milestone for the seismic instrumentation implementation program based on the discussion in Subsections 3.7.4.1 through 3.7.4.5.*
COL 3.7(20)	*The COL Applicant is to validate the site-independent seismic design of the standard plant for site-specific conditions, including geological, seismological, and geophysical characteristics, and to develop the site-specific GMRS as free-field outcrop motions on the uppermost in-situ competent material.*
COL 3.7(21)	*The COL Applicant is responsible for the seismic design of those seismic category I and seismic category II SSCs that are not part of the US-APWR standard plant.*
COL 3.7(22)	*The COL Applicant is required to perform site-specific seismic analyses, including SSI analysis which considers seismic wave transmission incoherence and analysis of the CAV of the seismic input motion, in order to determine if high-frequency exceedances of the CSDRS could be transmitted to SSCs in the plant superstructure with potentially damaging effects.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 9 of 44)

COL ITEM NO.	COL ITEM
COL 3.7(23)	*The COL Applicant is to verify that the results of the site-specific SSI analysis for the broadened ISRS and basement walls lateral soil pressures are enveloped by the US-APWR standard design.*
COL 3.7(24)	*The COL Applicant is to verify that the site-specific ratios V/A and AD/V^2 (A, V, D, are PGA, ground velocity, and ground displacement, respectively) are consistent with characteristic values for the magnitude and distance of the appropriate controlling events defining the site-specific uniform hazard response spectra.*
COL 3.7(25)	*The COL Applicant referencing the US-APWR standard design is required to perform a site-specific SSI analysis for the R/B-PCCV-containment internal structure utilizing the program ACS-SASSI SSI Version 2.2 (Reference 3.7-17) which contains time history input incoherence function capability. The SSI analysis using SASSI is required in order to confirm that site-specific effects are enveloped by the standard design. After the SASSI analysis is first performed for a specific unit, subsequent COLAs for other units may be able to forego SASSI analyses if the FIRS and GMRS derived for those subsequent units are much smaller than the US-APWR standard plant CSDRS, and if the subsequent unit can also provide justification through comparison of site-specific geological and seismological characteristics.*
COL 3.7(26)	*SSI effects are also considered by the COL Applicant in site-specific seismic design of any seismic category I and II structures that are not included in the US-APWR standard plant. Consideration of structure-to-structure interaction is discussed in Subsection 3.7.2.8. The site-specific SSI analysis is performed for buildings and structures including, but not limited to, to the following:* • *Seismic category I ESWPT* • *Seismic category I PSFSV* • *Seismic category I UHSRS*
COL 3.7(27)	*It is the responsibility of the COL Applicant to perform any site-specific seismic analysis for dams that may be required.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 10 of 44)

COL ITEM NO.	COL ITEM
COL 3.7(28)	*The overall basemat dimensions, basemat embedment depths, and maximum height of the US-APWR R/B, PCCV, and containment internal structure on their common basemat are given in Table 3.7.1-3 and as updated by the COL Applicant to include site-specific seismic category I structures.*
COL 3.7(29)	*Table 3.7.2-1, as updated by the COL Applicant to include site-specific seismic category I structures, presents a summary of dynamic analysis and combination techniques including types of models and computer programs used, seismic analysis methods, and method of combination for the three directional components for the seismic analysis of the US-APWR standard plant seismic category I buildings and structures.*
COL 3.7(30)	*The COL Applicant is to provide site-specific design ground motion time histories and durations of motion.*
COL 3.8(1)	*Deleted*
COL 3.8(2)	*Deleted*
COL 3.8(3)	*It is the responsibility of the COL Applicant to assure that any material changes based on site-specific material selection for construction of the PCCV meet the requirements specified in ASME Code, Section III, Article CC-2000 of the code and supplementary requirements of RG 1.136 as well as SRP 3.8.1.*
COL 3.8(4)	*Deleted*
COL 3.8(5)	*Deleted*
COL 3.8(6)	*Deleted*
COL 3.8(7)	*It is the responsibility of the COL Applicant to determine the site-specific aggressivity of the ground water/soil and accommodate this parameter into the concrete mix design as well as into the site-specific structural surveillance program.*
COL 3.8(8)	*Deleted*
COL 3.8(9)	*Deleted*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 11 of 44)

COL ITEM NO.	COL ITEM
COL 3.8(10)	*The prestressing system is designed as a strand system, however the system material may be switched to a wire system at the choice of the COL Applicant. If this is done, the COL Applicant is to adjust the US-APWR standard plant tendon system design and details on a site-specific basis.*
COL 3.8(11)	*Deleted*
COL 3.8(12)	*Deleted*
COL 3.8(13)	*Deleted*
COL 3.8(14)	*It is the responsibility of the COL Applicant to establish programs for testing and ISI of the PCCV, including periodic inservice surveillance and inspection of the PCCV liner and prestressing tendons in accordance with ASME Code Section XI, Subsection IWL.*
COL 3.8(15)	*The COL Applicant is responsible for the seismic design of those seismic category I and seismic category II SSCs not seismically designed as part of the US-APWR standard plant, including the following seismic category I structures:* • *ESWPT* • *UHSRS* • *PSFSVs*
COL 3.8(16)	*Deleted*
COL 3.8(17)	*Deleted*
COL 3.8(18)	*Deleted*
COL 3.8(19)	*The design and analysis of the ESWPT, UHSRS, PSFSVs, and other site-specific structures are to be provided by the COL Applicant based on site-specific seismic criteria.*
COL 3.8(20)	*The COL Applicant is to identify any applicable externally generated loads. Such site-specific loads include those induced by floods, potential non-terrorism related aircraft crashes, explosive hazards in proximity to the site, and projectiles and missiles generated from activities of nearby military installations.*
COL 3.8(21)	*Deleted*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 12 of 44)

COL ITEM NO.	COL ITEM
COL 3.8(22)	*The COL Applicant is to establish a site-specific program for monitoring and maintenance of seismic category I structures in accordance with the requirements of NUMARC 93-01 (Reference 3.8-28) and 10 CFR 50.65 (Reference 3.8-29) as detailed in RG 1.160 (Reference 3.8-30). For seismic category I structures, monitoring is to include base settlements and differential displacements.*
COL 3.8(23)	*The COL Applicant is to determine if the site-specific zone of maximum frost penetration extends below the depth of the basemats for the standard plant, and to pour fill concrete under any basemat above the frost line so that the bottom of fill concrete is below the maximum frost penetration level.*
COL 3.8(24)	*Other non-standard seismic category I buildings and structures of the US-APWR are designed by the COL Applicant based on site-specific subgrade conditions.*
COL 3.8(25)	*The site-specific COL are to assure the design criteria listed in Chapter 2, Table 2.0-1, is met or exceeded.*
COL 3.8(26)	*Subsidence and differential displacement may therefore be reduced to less than 2 in. if justified by the COL Applicant based on site specific soil properties.*
COL 3.8(27)	*The COL Applicant is to specify normal operating thermal loads for site-specific structures, as applicable.*
COL 3.8(28)	*The COL Applicant is to specify concrete strength utilized in non-standard plant seismic category I structures.*
COL 3.8(29)	*The COL Applicant is to provide design and analysis procedures for the ESWPT, UHSRS, and PSFSVs.*
COL 3.9(1)	*The COL Applicant is to assure snubber functionality in harsh service conditions, including snubber materials (e.g., lubricants, hydraulic fluids, seals).*
COL 3.9(2)	*The first COL Applicant is to complete the vibration assessment program, including the vibration test results, consistent with guidance of RG 1.20. Subsequent COL Applicant need only provide information in accordance with the applicable portion of position C.3 of RG 1.20 for Non-Prototype internals.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 13 of 44)

COL ITEM NO.	COL ITEM
COL 3.9(3)	*Deleted*
COL 3.9(4)	*Deleted*
COL 3.9(5)	*Deleted*
COL 3.9(6)	*The COL Applicant is to provide the program plan for IST of dynamic restraints in accordance with Nonmandatory Appendix A of ASME OM Code.*
COL 3.9(7)	*Deleted*
COL 3.9(8)	*The COL Applicant is to administratively control the edition and addenda to be used for the IST program plan, and to provide a full description of their IST program plan for pumps, vavles, and dynamic restraints.*
COL 3.9(9)	*Deleted*
COL 3.9(10)	*The COL Applicant is to identify the site-specific active pumps.*
COL 3.9(11)	*The COL Applicant is to provide site-specific, safety-related pump IST parameters and frequency.*
COL 3.9(12)	*The COL Applicant is to provide type of testing and frequency of site-specific valves subject to IST in accordance with the ASME Code.*
COL 3.10(1)	*The COL Applicant is to document and implement an equipment qualification program for seismic category I equipment and provide milestones and completion dates.*
COL 3.10(2)	*Deleted*
COL 3.10(3)	*The COL Applicant is to develop and maintain an equipment qualification file that contains a list of systems, equipment, and equipment support structures and summary data sheets referred to as an equipment qualification summary data sheet (EQSDS) of the seismic qualification for each piece of safety-related seismic category I equipment (i.e., each mechanical and electrical component of each system), which summarize the component's qualification.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 14 of 44)

COL ITEM NO.	COL ITEM
COL 3.10(4)	*Deleted*
COL 3.10(5)	*Components that have been previously tested to IEEE Std 344-1971 prior to submittal of the DCD are reevaluated to justify the appropriateness of the input motion and requalify the equipment, if necessary. The COL Applicant is to requalify the component using biaxial test input motion unless the applicant provides justification for using a single-axis test input motion.*
COL 3.10(6)	*Deleted*
COL 3.10(7)	*Deleted*
COL 3.10(8)	*For design of seismic category I and II SSCs that are not part of the standard plant, the COL Applicant can similarly eliminate the OBE, or optionally set the OBE higher than 1/3 SSE, provided the design of the non-standard plant's SSCs are analyzed for the chosen OBE.*
COL 3.10(9)	*The COL Applicant is to investigate if site-specific in-structure response spectra generated for the COL application may exceed the standard US-APWR design's in-structure response spectra in the high-frequency range. Accordingly, the COL Applicant is to consider the functional performance of vibration-sensitive components, such as relays and other instrument and control devices whose output could be affected by high frequency excitation.*
COL 3.10(10)	*Deleted*
COL 3.11(1)	*The COL Applicant is responsible for assembling and maintaining the environmental qualification document, which summarizes the qualification results for all equipment identified in Appendix 3D, for the life of the plant.*
COL 3.11(2)	*The COL Applicant is to describe how the results of the qualification tests are to be recorded in an auditable file in accordance with requirements of 10 CFR 50.49 (j).*
COL 3.11(3)	*The COL Applicant is to provide a schedule showing the EQ Program proposed implementation milestones.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 15 of 44)

COL ITEM NO.	COL ITEM
COL 3.11(4)	*The COL Applicant is to describe periodic tests, calibrations, and inspections to be performed during the life of the plant, which verify the identified equipment remains capable of fulfilling its intended function.*
COL 3.11(5)	*The COL Applicant is to identify the site-specific equipment to be addressed in the EQ Program, including locations and environmental conditions.*
COL 3.11(6)	*The COL Applicant is to qualify site-specific electrical and mechanical equipment (including instrumentation and control, and certain accident monitoring equipment) using an equivalent qualification process to that delineated for the US-APWR Standard Plant.*
COL 3.11(7)	*The COL Applicant is to identify chemical and radiation environmental requirements for site-specific qualification of electrical and mechanical equipment (including instrumentation and control, and certain accident monitoring equipment).*
COL 3.11(8)	*The COL Applicant is to provide the site-specific mechanical equipment requirements.*
COL 3.11(9)	*Optionally, the COL Applicant may revise the parameters based on site-specific considerations.*
COL 3.12(1)	*Deleted*
COL 3.12(2)	*If any piping is routed in tunnels or treches in the yard, the COL Applicant is to generate site-specific seismic response spectra, which may be used for the design of these piping systems.*
COL 3.12(3)	*If the COL Applicant finds it necessary to lay ASME Code, Section III (Reference 3.12-2), Class 2 or 3 piping exposed to wind or tornado loads, then such piping must be designed to the plant design basis loads.*
COL 3.12(4)	*The COL Applicant is to screen piping systems that are sensitive to high frequency modes for further evaluation.*
COL 3.13(1)	*Deleted*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 16 of 44)

COL ITEM NO.	COL ITEM
COL 3.13(2)	*Deleted*
COL 3.13(3)	*The COL Applicant is to retain quality records including certified material test reports for all property test and analytical work performed on nuclear threaded fasteners in accordance with the requirements of 10 CFR 50.71.*
COL 3.13(4)	*The COL Applicant is to address compliance with ISI requirements as summarized in Subsection 3.13.2.*
COL 3.13(5)	*The COL Applicant is to commit to complying with the requirements of ASME Code, Section XI, IWA-5000 (Reference 3.13-14), and the requirements of 10 CFR 50.55a(b)(2)(xxvi) (Reference 3.13-11), Pressure Testing Class 1, 2, and 3 Mechanical Joints, and Paragraph (xxvii) Removal of Insulation.*
COL 4.4(1)	*Deleted*
COL 5.2(1)	*ASME Code Cases that are approved in Regulatory Guide 1.84; The COL applicant addresses the addition of ASME Code Cases that are approved in Regulatory Guide 1.84.*
COL 5.2(2)	*ASME Code Cases that are approved in Regulatory Guide 1.147; The COL applicant addresses Code Cases invoked in connection with the inservice inspection program that are in compliance with Regulatory Guide 1.147.*
COL 5.2(3)	*ASME Code Cases that are approved in Regulatory Guide 1.192; The COL applicant addresses Code cases invoked in connection with the operation and maintenance that are in compliance with Regulatory Guide 1.192.*
COL 5.2(4)	*Inservice inspection and testing program for the RCPB* *The COL applicant addresses and develops the implementation milestone of the inservice inspection and testing program for the RCPB, in accordance with Section XI of the ASME Code and 10 CFR 50.55a.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 17 of 44)

COL ITEM NO.	COL ITEM
COL 5.2(5)	*Preservice inspection and testing program for the RCPB* *The COL applicant addresses and develops the implementation milestone of tge preservice inspection and testing program for the RCPB in accordance with Article NB-5280 of Section III, Division I of the ASME Code.*
COL 5.2(6)	*Deleted*
COL 5.2(7)	*Deleted*
COL 5.2(8)	*Deleted*
COL 5.2(9)	*Deleted*
COL 5.2(10)	*Deleted*
COL 5.2(11)	*ASME Code Edition and Addenda* *The COL applicant addresses whether the ASME Code editions or addenda other than specified in Table 5.2.1-1 or 10 CFR 10 CFR 50.55a is used.*
COL 5.2(12)	*EPRI Primary Water Chemistry Guideline* *The COL applicant should specify the applicable version of the EPRI "Primary Water Chemistry Guideline" that will be implemented.*
COL 5.2(13)	*ISI accessibility* *The COL applicant addresses the discussion of the provisions to preserve accessibility to perform ISI for Class 1 components provided design of US-APWR Class 1 component is changed from the DCD design.*
COL 5.2(14)	*Procedures for conversion into common leakage rate* *The COL applicant addresses and develops a milestone schedule for preparation and implementation of the procedure.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 18 of 44)

COL ITEM NO.	COL ITEM
COL 5.2(15)	*Procedures for operator response to prolonged low-level leakage* *The COL applicant addresses and develops a milestone schedule for preparation and implementation of the procedure.*
COL 5.3(1)	*Pressure-Temperature Limit Curves; The COL applicant addresses the use of plant-specific reactor vessel P-T limit curves. Generic P-T limit curves for the US-APWR reactor vessel are shown in Figures 5.3-2 and 5.3-3, which are based on the conditions described in Subsection 5.3.2. However, for a specific US-APWR plant, these limit curves are plotted based on actual material composition requirements and the COL applicant addresses the use of these plant-specific curves.*
COL 5.3(2)	*Reactor Vessel Material Surveillance Program; The COL applicant provides a reactor vessel material surveillance program based on information in Subsection 5.3.1.6.*
COL 5.3(3)	*Surveillance Capsule Orientation and Lead Factors; The COL applicant addresses the orientation and resulting lead factors for the surveillance capsules of a particular US-APWR plant.*
COL 5.3(4)	*Reactor Vessel Material Properties Verification; The COL applicant verifies the USE and RT_{NDT} at EOL, including a PTS evaluation based on actual material property requirements of the reactor vessel material and the projected neutron fluence for the design-life objective of 60 years.*
COL 5.3(5)	*Preservice and Inservice Inspection; The COL applicant provides the information for preservice and inservice inspection described in Subsection 5.2.4.*
COL 5.4(1)	*Deleted*
COL 5.4(2)	*Deleted*
COL 5.4(3)	*Deleted*
COL 5.4(4)	*Deleted*
COL 5.4(5)	*Deleted*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 19 of 44)

COL ITEM NO.	COL ITEM
COL 5.4(6)	*Deleted*
COL 5.4(7)	*Deleted*
COL 6.1(1)	*Deleted*
COL 6.1(2)	*Deleted*
COL 6.1(3)	*Deleted*
COL 6.1(4)	*Deleted*
COL 6.1(5)	*Deleted*
COL 6.1(6)	*Deleted*
COL 6.1(7)	*The COL Applicant is responsible for identifying the implementation milestones for the coatings program.*
COL 6.2(1)	*Deleted*
COL 6.2(2)	*Deleted*
COL 6.2(3)	*Deleted*
COL 6.2(4)	*Deleted*
COL 6.2(5)	*Preparation of a cleanliness, housekeeping and foreign materials exclusion program is the responsibility of the COL applicant. This program addresses other debris sources such as latent debris inside containment. This program minimizes foreign materials in the containment.*
COL 6.2(6)	*Deleted*
COL 6.2(7)	*Deleted*
COL 6.2(8)	*The COL applicant is responsible for identifying the implementation milestone for the containment leakage rate testing program described under 10CFR50, Appendix J.*
COL 6.2(9)	*Deleted*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 20 of 44)

COL ITEM NO.	COL ITEM
COL 6.2(10)	*Deleted*
COL 6.3(1)	*Deleted*
COL 6.3(2)	*Deleted*
COL 6.3(3)	*Deleted*
COL 6.3(4)	*Deleted*
COL 6.3(5)	*Deleted*
COL 6.3(6)	*Deleted*
COL 6.4(1)	*The COL Applicant is responsible to provide details of specific toxic chemicals of mobile and stationary sources within the requirements of RG 1.78 (Ref 6.4-4) and evaluate the control room habitability based on the recommendation of RG 1.78 (Ref 6.4-4).*
COL 6.4(2)	*The COL Applicant is responsible to discuss the automatic actions and manual actions for the MCR HVAC system in the event of postulated toxic gas release.*
COL 6.4(3)	*Deleted*
COL 6.4(4)	*Deleted*
COL 6.4(5)	*The number, locations, sensitivity, range, type, and design of the toxic gas detectors are COL items. Depending on proximity to nearby industrial, transportation, and military facilities, and the nature of the activities in the surrounding area, as well as specific chemicals onsite, the COL Applicant is responsible to specify the toxic gas detection requirements necessary to protect the CRE.*
COL 6.5(1)	*Deleted*
COL 6.5(2)	*Deleted*
COL 6.5(3)	*Deleted*
COL 6.5(4)	*Deleted*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 21 of 44)

COL ITEM NO.	COL ITEM
COL 6.6(1)	*The COL Applicant is responsible for identifying the implementation milestone for ASME Section XI inservice inspection program for ASME Code Section III Class 2 and 3 systems, components (pums and valves), piping, and supports, consistent with the requirements of 10 CFR 50.55a (g).*
COL 6.6(2)	*The COL Applicant is responsible for identifying the implementation milestone for the augmented inservice insection program.*
COL 7.3(1)	*Deleted*
COL 7.4(1)	*The COL applicant is to provide a description of component controls and indications required for safe shutdown related to the UHS.*
COL 7.5(1)	*The COL applicant is to provide a description of site-specific PAM variables.*
COL 7.5(2)	*The COL applicant is to provide a description of the site-specific EOF.*
COL 7.9(1)	*Deleted*
COL 8.2(1)	*The COL applicant is to address transmission system of the utility power grid and its interconnection to other grids.*
COL 8.2(2)	*Deleted*
COL 8.2(3)	*The COL applicant is to address the plant switchyard which includes layout, control system and characteristics of circuit breakers and buses, and lighting and grounding protection equipment.*
COL 8.2(4)	*The COL applicant is to provide detail description of normal preferred power.*
COL 8.2(5)	*The COL applicant is to provide detail description of alternate preferred power.*
COL 8.2(6)	*Deleted*
COL 8.2(7)	*The COL applicant is to address protective relaying for each circuit such as lines and buses.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 22 of 44)

COL ITEM NO.	COL ITEM
COL 8.2(8)	*The COL applicant is to address switchyard dc power as part of switchyard design description.*
COL 8.2(9)	*The COL applicant is to address switchyard ac power as part of switchyard design description.*
COL 8.2(10)	*The COL applicant is to address transformer protection corresponded to site-specific scheme.*
COL 8.2(11)	*The COL applicant is to address the stability and reliability study of the offsite power system. Stability study is to be addressed in accordance with BTP 8-3 (Reference 8.2-17). The study addresses the loss of the unit, loss of the largest unit, loss of the largest load, or loss of the most critical transmission line including operating range, for maintaining transient stability. A failure modes and effects analysis (FMEA) is to be provided.*
COL 8.2(12)	*Deleted*
COL 8.3(1)	*The COL applicant is to provide transmission voltages. This includes also MT and RAT voltage ratings.*
COL 8.3(2)	*The COL applicant is to provide ground grid and lightning protection.*
COL 8.3(3)	*The COL applicant is to provide short circuit analysis for ac power system, since the system contribution is site specific.*
COL 8.3(4)	*Deleted*
COL 8.3(5)	*Deleted*
COL 8.3(6)	*Deleted*
COL 8.3(7)	*Deleted*
COL 8.3(8)	*The COL applicant is to provide short circuit analysis for dc power system.*
COL 8.3(9)	*Deleted*
COL 8.3(10)	*The COL applicant is to provide protective devise coordination.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 23 of 44)

COL ITEM NO.	COL ITEM
COL 8.3(11)	*The COL applicant is to provide insulation coordination (surge and lightning).*
COL 9.1(1)	*Deleted*
COL 9.1(2)	*Deleted*
COL 9.1(3)	*Deleted*
COL 9.1(4)	*Deleted*
COL 9.1(5)	*Deleted*
COL 9.1(6)	*To assure proper handling of heavy loads during the plant life, the COL Applicant is to establish a heavy load handling program, including associated procedural and administrative controls, that satisfies commitments made in Subsection 9.1.5 of the DCD, and that meets the guidance of ANSI/ASME B30.2, ANSI/ASME B30.9, ANSI N14.6, ASME NOG-1, CMAA Specification 70-2000, NUREG-0554, NUREG-0612, and NUREG-0800, Section 9.1.5. During the operating life of the plant, it is anticipated that temporarily installed hoists and mobile cranes will also be used for plant maintenance. The heavy load handling program will include temporary cranes and hoists. The heavy load handling program will adopt a defense-in-depth strategy to enhance safety when handling heavy loads. For instance, the program will restrict lift heights to practical minimums and limit lifting activities as much as practical to plant modes in which load drops have the smallest potential for adverse consequences, particularly when critical loads are being handled. Further, prior to the lifting of heavy loads after initial fuel loading, the program will institute any additional reviews as necessary to assure that potential drops of these loads due to inadvertent operations or equipment malfunctions, separately or in combination, will not jeopardize safe shutdown functions, cause a significant release of radioactivity, a criticality accident, or inability to cool fuel within the reactor vessel or spent fuel pool.*
COL 9.1(7)	*Deleted*
COL 9.1(8)	*Deleted*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 24 of 44)

COL ITEM NO.	COL ITEM
COL 9.1(9)	*The COL allpicant is to create a procedure that will instruct operators to perform formal inspection of the integrity of the spent fuel racks.*
COL 9.2(1)	*The COL Applicant is to provide the evaluation of the ESWP at the lowest probable water level of the UHS. The COL Application is to develop recovery procedures in the event of approaching low water level of UHS.*
COL 9.2(2)	*The COL Applicant is to provide the protection against adverse environmental, operating, and accident conditions that can occur, such as freezing, thermal overpressurization. The COL Applicant is to provide the preventive measures for protection against adverse environmental conditions.*
COL 9.2(3)	*The COL Applicant is to determine source and location of the UHS.*
COL 9.2(4)	*The COL Applicant is to determine location and design of the ESW intake structure.*
COL 9.2(5)	*The COL Applicant is to determine location and design of the ESW discharge structure.*
COL 9.2(6)	*The COL Applicant is to provide ESWP design details – required total dynamic head, NPSH available etc.*
COL 9.2(7)	*The COL Applicant is to provide the piping, valves, including those at the boundary between the safety-related and nonsafety-related portions, and other design of the ESWS related to the site specific conditions, including the safety evaluation.*
COL 9.2(8)	*The COL Applicant is to specify the following ESW chemistry requirements* • *A chemical injection system to provide non-corrosive, non-scale forming conditions to limit biological film formation.* • *Type of biocide, algaecide, pH adjuster, corrosion inhibitor, scale inhibitor and silt dispersant based on the site conditions.*
COL 9.2(9)	*COL Applicant is to confirm the storage capacity and usage of the potable water.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 25 of 44)

COL ITEM NO.	COL ITEM
COL 9.2(10)	*COL Applicant is to confirm that all State and Local Department of Health and Environmental Protection Standards are applied and followed.*
COL 9.2(11)	*The COL Applicant is to confirm the source of potable water to the site and the necessary required treatment.*
COL 9.2(12)	*COL Applicant is to confirm that the sanitary waste is sent to the onsite plant treatment area or they will use the city sewage system.*
COL 9.2(13)	*COL Applicant is to identify thepotable water supply and describe the system operation.*
COL 9.2(14)	*COL Applicant is to confirm Table 9.2.4-1 for required components and their values.*
COL 9.2(15)	*The COL Applicant is to determine the total number of people at the site and identify the usage capacity. Based on these numbers the COL Applicant is to size the potable water tank and associated pumps.*
COL 9.2(16)	*The COL Applicant is to provide values to the component Table 9.2.4-1 based on system and component descriptions from Section 9.2.4.2.1 and 9.2.4.2.2 respectively.*
COL 9.2(17)	*The COL Applicant is to determine the total number of sanitary lift stations and is to size the appropriate interfaces.*
COL 9.2(18)	*The COL Applicant is to determine the type of the UHS based on specific site conditions and meteorological data.*
COL 9.2(19)	*The COL Applicant is to design the UHS to receive its electrical power supply, if required by the UHS design, from safety busses so that the safety functions are maintained during LOOP. The UHS also receives its standby electrical power from the onsite emergency power supplies during a LOOP.*
COL 9.2(20)	*The COL Applicant is to provide a detailed description and drawings of the UHS, including water inventory, temperature limits, heat rejection capabilities, instrumentation, and alarms.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 26 of 44)

COL ITEM NO.	COL ITEM
COL 9.2(21)	*The COL Applicant is to determine the source of makeup water to the UHS inventory and the blowdown discharge location based on specific site conditions.*
COL 9.2(22)	*The COL Applicant is to provide results of UHS capability and safety evaluation of the UHS based on specific site conditions and meteorological data. The COL Applicant is to use at least 30 years site specific meteorological data and heat loads data for UHS performance analysis.*
COL 9.2(23)	*The COL Applicant is to provide test and inspection requirements of the UHS. These is to include inspection and testing requirements necessary to demonstrate that fouling and degradation mechanisms are adequately managed to maintain acceptable UHS performance and integrity.*
COL 9.2(24)	*The COL Applicant is to provide the required alarms, instrumentation and controls details based on the type of UHS to be provided.*
COL 9.2(25)	*The COL applicant will develop operating and maintenance procedures for the ESWS to address water hammer issues in accordance with NUREG-0927.*
COL 9.2(26)	*The COL applicant is to develop maintenance and test procedures to monitor debris build up and flush out debris.*
COL 9.2(27)	*The COL Applicant is to develop a milestone schedule for implementation of the operating and maintenance procedures for water hammer prevention.*
COL 9.3(1)	*The COL Applicant is to provide the high pressure nitrogen gas, low pressure nitrogen gas, the hydrogen gas, carbon dioxide, and oxygen supply systems.*
COL 9.3(2)	*Deleted*
COL 9.3(3)	*Deleted*
COL 9.3(4)	*Deleted*
COL 9.3(5)	*Deleted*
COL 9.3(6)	*Deleted*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 27 of 44)

COL ITEM NO.	COL ITEM
COL 9.3(7)	*Deleted*
COL 9.4(1)	*Deleted*
COL 9.4(2)	*Deleted*
COL 9.4(3)	*Deleted*
COL 9.4(4)	*The COL Applicant is to determine the capacity of cooling and heating coils provided in the air handling units that are affected by site specific conditions.*
COL 9.4(5)	*Deleted*
COL 9.4(6)	*The COL Applicant is to provide a system information and flow diagram of ESW pump area ventilation system if the ESW pump area requires the heating, ventilating and air conditioning.*
COL 9.5(1)	*The COL applicant establishes 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 addresses the design and fire protection aspects of the facilities, buildings and equipments, such as cooling towers and a fire protection water supply system, which are site specific and/or are not a standard feature of the US-APWR.*
COL 9.5(3)	*The COL Applicant provides apparatus for plant personnel and fire brigades such as portable fire extinguishers and self contained breathing apparatus.*
COL 9.5(4)	*The COL Applicant addresses all communication system interfaces external to the plant (offsite locations). These include interfaces to utility private networks, commercial carriers and the federal telephone system. The configuration of these connections will include consideration of the concerns raised in IE Bulletin 80-15.*
COL 9.5(5)	*The COL Applicant addresses the emergency offsite communications including the crisis management radio system.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 28 of 44)

COL ITEM NO.	COL ITEM
COL 9.5(6)	*The COL Applicant addresses connections to the Technical Support Center from where communications networks are provided to transmit information pursuant to the requirements delineated in 10 CFR 50 Appendix E, Part IV.E.9.*
COL 9.5(7)	*The COL Applicant addresses a continuously manned alarm station required by 10 CFR 73.46(e)(5) and the communications requirements delineated in 10 CFR 73.45(g)(4)(i) and (ii). The COL Applicant addresses notification of an attempted unauthorized or unconfirmed removal of strategic special nuclear material in accordance with 10 CFR 73.45(e)(2)(iii).*
COL 9.5(8)	*The COL Applicant addresses offsite communications for the onsite operations support center.*
COL 9.5(9)	*The COL Applicant addresses the emergency communication system requirements delineate in 10 CFR 73.55(f) such that a single act cannot remove onsite capability of calling for assistance and also as redundant system during onsite emergency crisis.*
COL 9.5(10)	*Deleted*
COL 9.5(11)	*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 10.2(1)	*Inservice Inspection* *The Combined License Applicant is to establish a turbine maintenance and inspection procedure prior to fuel load.*
COL 10.3(1)	*FAC monitoring program* *The Combined License Applicant will provide a description of the FAC monitoring program for carbon steel portions of the steam and power conversion systems that contain water or wet steamand are susceptible erosion-corrosion damage. The description will address consistency with Generic Letter 89-08 and NSAC-202L-R2 and will provide a milestone schedule for implementation of the program.*
COL 10.3(2)	*Delete*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 29 of 44)

COL ITEM NO.	COL ITEM
COL 10.3(3)	*Operating and maintenance procedures for water hammer prevention* *The Combined License Applicant is to provide operating and maintenance procedures including adequate precautions to prevent water (steam) hammer, relief valve discharge loads and wqater entrainment effects in accordance with UNREG-0927 and a milestone schedule for implementation of the procedure.*
COL 10.4(1)	*Circulating Water System; The Combined License Applicant is to determine the site specific final system configuration and system design parameters for the CWS including makeup water and blowdown.*
COL 10.4(2)	*Steam Generator Blowdown System; The Combined License applicant is to address the discharge to Waste Water System including site specific requirements.*
COL 10.4(3)	*Deleted*
COL 10.4(4)	*Deleted*
COL 10.4(5)	*System Design for Steam Generator Drain; The Combined License applicant is to address the nitrogen or equivalent system design for Steam Generator Drain Mode. (This is dependent on Waste water system design)*
COL 10.4(6)	*Operating and maintenance procedures for water hammer prevention* *The combined License Applicant is to provide operating and maintenance procedures in accordance with NUREG-0927 and a milestone schedule for implementation of the procedure.*
COL 11.2(1)	*The COL applicant is responsible for ensuring that mobile and temporary liquid radwaste processing equipment and its interconnection to plant systems conforms to regulatory requirements and guidance such as 10 CFR 50.34a (Ref. 11.2-5), 10 CFR 20.1406 (Ref.11.2-7) and RG 1.143 (Ref. 11.2-3), respectively.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 30 of 44)

COL ITEM NO.	COL ITEM
COL 11.2(2)	*Site-specific information of the LWMS, e.g., radioactive release points, effluent temperature, shape of flow orifices, etc., is provided in the COLA.*
COL 11.2(3)	*The COL applicant is responsible for providing site-specific hydrogeological data (such as contaminant migration time), and analysis to demonstrate that the potential groundwater contamination resulting from radioactive release due to liquid containing tank failure is bounded by the analysis discussed in Subsection 11.2.3.2.*
COL 11.2(4)	*The COL applicant is to calculate doses to members of the public following the guidance of RG 1.109* (Ref 11.2-15) *and RG 1.113 using site-specific parameters, and compares the doses due to the liquid effluents with the numerical design objectives of Appendix I to 10 CFR 50* (Ref 11.2-10) *and compliance with requirements of 10 CFR 20.1302, 40 CFR 190.*
COL 11.2(5)	*The COL applicant is to perform a site-specific cost benefit analysis to demonstrate compliance with the regulatory requirements.*
COL 11.2(6)	*The COL applicant is to provide piping and instrumentation diagrams (P&IDs).*
COL 11.3(1)	*Deleted*
COL 11.3(2)	*Deleted*
COL 11.3(3)	*The COL applicant is to provide a discussion of the onsite vent stack design parameters and release point specific characteristics.*
COL 11.3(4)	*Deleted*
COL 11.3(5)	*Deleted*
COL 11.3(6)	*The COL applicant is to calculate doses to members of the public following the guidance of RG 1.109(Ref. 11.3-19) and RG 1.111(Ref. 11.3-22), and compare the doses due to the gaseous effluents with the numerical design objectives of 10 CFR 50, Appendix I (Ref. 11.3-3) and compliance with requirements of 10 CFR 20.1302(Ref. 11.3-24), 40 CFR 190(Ref. 11.3-25).*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 31 of 44)

COL ITEM NO.	COL ITEM
COL 11.3(7)	*Deleted*
COL 11.3(8)	*The COL applicant is to perform a site-specific cost benefit analysis to demonstrate compliance with the regulatory requirements.*
COL 11.3(9)	*The COL applicant is to provide piping and instrumentation diagrams (P&IDs).*
COL 11.4(1)	*The current design meets the waste storage requirements in accordance with ANSI/ANS-55.1. When the COL applicant desires additional storage capability beyond that which is discussed in this Tier 2 document, the COL applicant will identify plant-specific needs for on-site waste storage and provide a discussion of on-site storage of low-level waste.*
COL 11.4(2)	*Deleted*
COL 11.4(3)	*The COL applicant is to prepare a plan for the process control program describing the process and effluent monitoring and sampling program. The plan should include the proposed implementation milestones.*
COL 11.4(4)	*The COL applicant is to describe mobile/portable SWMS connections that are considered non-radioactive but later may become radioactive through contact or contamination with radioactive systems (i.e., a non-radioactive system becomes contaminated due to leakage, valving errors, or other operating conditions in the radioactive systems), and operational procedures of the mobile/portable SWMS connections.*
COL 11.4(5)	*The current design provides collection and packaging of potentially contaminated clothing for offsite shipment and/or disposal. Depending on site-specific requirements, the COL applicant can send the wastes to an offsite laundry facility processing and/or bring in a mobile compaction unit for volume reduction. The laundry services, including contracted service and/or a temporary mobile compaction subsystem are COL items.*
COL 11.4(6)	*The COL applicant is required to perform a site-specific cost benefit analysis to demonstrate compliance with the regulatory requirements.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 32 of 44)

COL ITEM NO.	COL ITEM
COL 11.4(7)	*The SWMS design does no include solid waste processing facility (e.g. de-watering system, compactor for reducing waste volume) but provides the flexibility for the site-specific utilities to add compaction equipment or to adopt contract services from specialized facilities. This is the responsibility of the COL applicant.*
COL 11.4(8)	*The COL applicant is to provide piping and instrumentation diagrams (P&IDs).*
COL 11.5 (1)	*The COL applicant is responsible for the additional site-specific aspects of the process and effluent monitoring and sampling system beyond the standard design, in accordance with RGs 1.21, 1.33 and 4.15 (Ref. 11.5-12, 11.5-17, 11.5-14). Furthermore, the COL applicant is responsible for assuring the fulfillment of the guidelines issued in 10 CFR 50, Appendix I (Ref. 11.5-3) regarding the offsite doses released through gaseous and liquid effluent streams.*
COL 11.5(2)	*The COL applicant is to prepare an offsite dose calculation manual to provide specific administrative controls and liquid and gaseous effluent source terms to limit the releases to site-specific requirements containing a description of the methods and parameters that drive to arrive radiation instrumentation alarm setpoint. The COL applicant is to commit to follow the NEI generic template 07-09A (Ref. 11.5-30) as an alternative to providing the offsite dose calculation manual at the time of application.*
COL 11.5(3)	*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 program shall take into account associated radioactive materials present in liquid and gaseous effluents and direct external radiation from SSCs. The COL applicant is to follow the guidance outlined in NUREG-1301(Ref. 11.5-21), and NUREG-0133 (Ref. 11.5-18) when developing the radiological effluent monitoring program. The COL applicant is to commit to follow the NEI generic template 07-09A (Ref. 11.5-30) as an alternative to providing the radiological effluent monitoring program at the time of application.*
COL 11.5(4)	*The COL applicant is to develop procedures which are of inspection, decontamination, and replacement related to radiation monitoring instruments.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 33 of 44)

COL ITEM NO.	COL ITEM
COL 11.5(5)	*The COL applicant is to provide analytical procedures and sensitivity for selected radioanalytical methods and type of sampling media for site-specific matter.*
COL 11.5(6)	*The COL applicant is to perform a site-specific cost benefit analysis to demonstrate compliance with the regulatory requirements.*
COL 12.1(1)	*The COL Applicant is to demonstrate that the policy considerations regarding plant operations are compliance with RG 1.8, 8.8 and 8.10 (Subsection 12.1.1.3).*
COL 12.1(2)	*Deleted*
COL 12.1(3)	*The COL Applicant is to describe how the plant follows the guidance of RG 8.2, 8.4, 8.6, 8.7, 8.9, 8.13, 8.15, 8.25, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36 and 8.38.*
COL 12.1(4)	*Deleted*
COL 12.1(5)	*The COL Applicant is to provide the operational radiation protection program for ensuring that occupational radiation exposures are ALARA.*
COL 12.1(6)	*The COL applicant is to performe periodic review of operational practices to ensere configuration management, personnel training and qualification update, and procedure adherence.*
COL 12.1(7)	*The COL applicant is to track implementation of requirements for record retention according to 10 CFR50.75(g) and 10 CFR70.25(g) as applicable.*
COL 12.2(1)	*The COL Applicant is to list any additional contained radiation sources that are not identified in Subsection 12.2.1, including radiation sources used for instrument calibration or radiography.*
COL 12.2(2)	*The COL Applicant is to address the radiation protection aspects associated with additional storage space for radwaste and/or additional radwaste facilities for dry active waste.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 34 of 44)

COL ITEM NO.	COL ITEM
COL 12.3(1)	*The COL Applicant is responsible for the use of portable instruments, and the associated training and procedures, to accurately determine the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident, in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737.*
COL 12.3(2)	*Deleted*
COL 12.3(3)	*Deleted*
COL 12.3(4)	*The COL Applicant is to provide the site radiation zones that is shown on the site-specific plant arrangement plan.*
COL 12.3(5)	*The COL Applicant is to discuss the administrative control of the fuel transfer tube inspection and the access control of the area near the seismic gap below the fuel transfer tube.*
COL 12.4(1)	*For multiunit plants, the COL Applicant is to provide estimated annual doses to construction workers in a new unit construction area, as a result of radiation from onsite radiation sources from the existing operating plant(s).*
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 qualifications of personnel. The COL Applicant directs attention to activities that include 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 past experience in the design, construction, and operation of nuclear power plants and past 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 that reflect the added functional responsibilities with the nuclear power plant.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 35 of 44)

COL ITEM NO.	COL ITEM
COL 13.1(4)	*The COL Applicant is to develop a description of the organizational arrangement. This description shows how the added functional responsibilities associated with the addition of the nuclear power plant to the Applicant's power generation capacity are delegated and assigned (or expected to be assigned to) each of the working or performance-level organizational units to implement these responsibilities. The description includes organizational charts reflecting the current corporate structure and the specific 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 qualification requirements in terms of educational background and experience for positions or classes of positions depicted in the organizational arrangement.*
COL 13.1(6)	*The COL Applicant is to develop the organizational structure for the plant organization, its personnel responsibilities and authorities, and operating shift crews.*
COL 13.1(7)	*The COL Applicant is to develop the description of education, training, and experience requirements established for management, operating, technical, and maintenance positions for the operating organization.*
COL 13.2(1)	*The COL Applicant is to develop the training program description.*
COL 13.2(2)	*The COL Applicant is to develop training programs for reactor operators in accordance with NUREG-0800, Section 13.2.1.I.3 (Ref. 13.2-4).*
COL 13.2(3)	*The COL Applicant is to develop training programs for non-licensed plant staff In accordance with NUREG-0800, Section 13.2.2.I.3 (Ref. 13.2-4).*
COL 13.2(4)	*The COL Applicant is to develop training programs. These programs include a chart, which 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.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 36 of 44)

COL ITEM NO.	COL ITEM
COL 13.2(5)	*The COL Applicant is to determine the extent to which portions of applicable NRC guidance is used in the facility training program or the justification of exceptions.*
COL 13.3(1)	*The COL Applicant is to develop 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 as a physically separate document.*
COL 13.3(3)	*The COL Applicant is to develop an emergency classification and action level scheme.*
COL 13.3(4)	*The COL Applicant is to develop the security-related aspects of emergency planning.*
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 an emergency planning inspections, tests, analyses, and acceptance criteria.*
COL 13.3(7)	*The COL Applicant is to develop the description of the operation support center.*
COL 13.4(1)	*The COL Applicant is to develop a description and schedule for the implementation of operational programs. The COL Applicant is to "fully describe" the operational programs as defined in SECY-05-0197 (Ref. 13.4-1) and provide commitments for the implementation of operational programs required by regulation. In some instances, programs may be implemented in phases. The COL Applicant is to include the phased implementation milestones in their submittal.*
COL 13.5(1)	*The COL Applicant is to develop administrative procedures describing administrative controls over activities that are important to safety for the operation of a facility.*
COL 13.5(2)	*Deleted*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 37 of 44)

COL ITEM NO.	COL ITEM
COL 13.5(3)	*The COL Applicant is to develop procedures performed by licensed operators in the main control room. Operating procedures that are used by the operating organization to ensure routine operating, off-normal, and emergency activities are conducted in a safe manner are described. The plan includes the implementation of these procedures (Ref. 13.5-3).*
COL 13.5(4)	*The COL Applicant is to describe the different classifications of procedures the operators will use in the main control room and locally in the plant for operations, the operating organization responsible for maintaining the procedures, and the general format and content of the different classifications.*
COL 13.5(5)	*The COL Applicant is to describe the program for developing operating procedures.*
COL 13.5(6)	*The COL Applicant is to describe the program for developing and implementing emergency operating procedures.*
COL 13.5(7)	*The COL Applicant is to describe the classifications of maintenance and other operating procedures, the operating organization group or groups responsible for following each class of procedure, and the general objectives and character of each class and subclass.*
COL 13.6(1)	*The COL Applicant is to develop and provide plant overall security plan (consisting of the physical security plan, safeguards contingency plan, and the guard training and qualification plan) and the cyber security plan and the implementation schedule for security programs.*
COL 13.6(2)	*The COL applicant is to develop and provide as part of its physical security plan site specific physical security features and capabilities, such as (i) the physical barrier surrounding the protected area boundary; (ii) the isolation zone in areas adjacent to the protected area boundary, (iii) security lighting, or use of low-light technology, for the isolation zone and protected area; (iv) the vehicle barrier system, (v) controlled access points to control entry of personnel, vehicles and materials into the protected area, (vi) the intrusion detection system, and (vii) the closed circuit television camera and video assessment systems to provide monitoring and assessment of the protected area perimeter.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 38 of 44)

COL ITEM NO.	COL ITEM
COL 13.6(3)	*The COL applicant is to revise the non-standard plant vital area and vital equipment information contained in the US-APWR Design Certification, Physical Element Review to be consistent with its site-specific design.*
COL 13.6(4)	*The COL applicant is to make provision for the secondary alarm station in accordance with the requirements of 10 CFR 73.55(i)(4).*
COL 13.6(5)	*The COL applicant's physical security plan is to make provision for radio or microwave transmitted two-way voice communication) to communicate with the local law enforcement agencies.*
COL 13.7(1)	*The COL Applicant is to develop the description of the operating and construction plant fitness-for-duty programs.*
COL 14.2(1)	*Deleted*
COL 14.2(2)	*The COL Applicant reconciles the site-specific organization, organizational titles, organizational responsibilities, and reporting relationships to be consistent with US-APWR Test Program Description Technical Report, MUAP-08009 (Reference 14.2-29) [14.2.2].*
COL 14.2(3)	*Deleted*
COL 14.2(4)	*Deleted*
COL 14.2(5)	*Deleted*
COL 14.2(6)	*Deleted*
COL 14.2(7)	*The COL applicant provides an event-based schedule, relative to fuel loading, for conucting each major phase of the test program, and a schedule for the development of plant procedures that assures required procedures are available for use during the preparation, review and performance of preoperational and startup testing. For multiunit sites, the COL applicant discusses the effects of overlapping initial test program schedules on organizations and personnel participating in each ITP. The COL applicant identifies and cross-references each test or portion of a test required to be completed prior to fuel load which safisfies ITAAC requirements. [14.2.9] [14.2.11]*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 39 of 44)

COL ITEM NO.	COL ITEM
COL 14.2(8)	*Deleted*
COL 14.2(9)	*Deleted*
COL 14.2(10)	*The COL applicant is responsible for the testing outside scope of the certified design in accordance with the test criteria described in subsection 14.2.1. [14.2.12]*
COL 14.2(11)	*The COL holder for the first plant is to perform the first plant only test and prototype test. For subsequent plants, either these tests are performed, or the COL applicant provides a justification that the results of the first-plant only tests are applicable to the subsequent plant and are not required to be repeated. [14.2.8]*
COL 14.2(12)	*The COL holder makes available approved test procedures for satisfying testing requirements described in Section 14.2 to the NRC approximately 60 days prior to their intended use. [14.2.3, 14.2.11, 14.2.12.1]*
COL 14.3(1)	*The COL applicant provides the ITAAC for the site specific portion of the plant systems specified in Subsection 14.3.5, Interface Requirements. [14.3.4.6,14.3.4.7]*
COL 14.3(2)	*The COL applicant provides proposed ITAAC for the facility's emergency planning not addressed in the DCD in accordance with RG 1.206 (Reference 14.3-1) as appropriate. [14.3.4.10]*
COL 14.3(3)	*The COL applicant provides proposed ITAAC for the facility's physical security hardware not addressed in the DCD in accordance with RG 1.206 (Reference 14.3-1) as appropriate. [14.3.4.12]*
COL 15.0(1)	*In the COLA, if the site-specific χ/Q values exceed DCD χ/Q values, then the COL Applicant is to demonstrate how the dose reference values in 10 CFR 50.34 and 10 CFR 52.79 and the control room dose limits in 10 CFR 50, Appendix A, General Design Criterion 19 are met for affected events using site-specific χ/Q values. Additionally, the Technical Support Center (TSC) dose should be evaluated against the habitability requirements in Paragraph IV.E. 8 to 10 CFR Part 50, Appendix E, and 10 CFR 50.47(b)(8) and (b)(11).*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 40 of 44)

COL ITEM NO.	COL ITEM
COL 16.1(1)	*Adoption of RMTS is to be confirmed and the relevant descriptions are to be fixed.*
COL 16.1(2)	*Adoption of SFCP is to be confirmed and the relevant descriptions are to be fixed.*
COL 16.1(3)	*Deleted*
COL 16.1_3.3.1(1)	*The trip setpoints and allowable values in Table 3.3.1-1 are to be confirmed after completion of a plant specific setpoint study following selection of the plant specific instrumentation.*
COL 16.1_3.3.2(1)	*The trip setpoints, allowable values and time delay value in Table 3.3.2-1 are to be confirmed after completion of a plant specific setpoint study following selection of the plant specific instrumentation.*
COL 16.1_3.3.5(1)	*The trip setpoints and time delay values in SR 3.3.5.3 are to be confirmed after completion of a plant specific setpoint study following selection of the plant specific instrumentation.*
COL 16.1_3.3.6(1)	*The trip setpoints and allowable values in Table 3.3.6-1 are to be confirmed after completion of a plant specific setpoint study following selection of the plant specific instrumentation.*
COL 16.1_3.4.17(1)	*The site specific information for tube repair is to be provided.*
COL 16.1_3.7.9(1)	*LCO 3.7.9 and associated Bases for the Ultimate Heat Sink based on plant specific design, including required UHS water volume, lowest water level for ESW pumps and maximum water temperature of the UHS, are to be developed.*
COL 16.1_3.7.10(1)	*LCO 3.7.10 and associated Bases for hazardous chemical are to be confirmed by the evaluation with site-specific condition.*
COL 16.1_3.8.4(1)	*The battery float current values in required action A.2 is to be confirmed after selection of the plant batteries.*
COL 16.1_3.8.5(1)	*The battery float current values in required action A.2 is to be confirmed after selection of the plant batteries.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 41 of 44)

COL ITEM NO.	COL ITEM
COL 16.1_3.8.6(1)	*The battery float current values in condition B, required action B.2, and SR 3.8.6.1 are to be confirmed after selection of the plant batteries.*
COL 16.1_4.1(1)	*The site specific information for site location is to be provided.*
COL 16.1_4.3.1(1)	*The site specific boron concentration is to be provided.*
COL 16.1_5.1.1(1)	*The titles for members of the unit staff are to be specified .*
COL 16.1_5.1.2(1)	*The titles for members of the unit staff are to be specified .*
COL 16.1_5.2.1(1)	*The titles for members of the unit staff are to be specified.*
COL 16.1_5.2.2(1)	*The titles and number for members of the unit staff are to be specified.*
COL 16.1_5.3.1(1)	*Minimum qualification for unit staff is to be specified.*
COL 16.1_5.5.1(1)	*The titles for members of the unit staff that approve the Offsite Dose Calculation Manual are to be specified.*
COL 16.1_5.5.9(1)	*The site specific information for tube repair is to be provided.*
COL 16.1_5.5.20(1)	*Control Room Envelope Habitability Program for hazardous chemical are to be confirmed by the evaluation with site-specific condition.*
COL 16.1_5.6.1(1)	*In case of multiple unit site, the additional information for submittal of report is to be added.*
COL 16.1_5.6.1(2)	*The format of the Annual Radiological Environmental Operating Report is to be specified based on “the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979” or another format.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 42 of 44)

COL ITEM NO.	COL ITEM
COL 16.1_5.6.2(1)	*In case of multiple unit site, the additional information for submittal of report is to be added.*
COL 16.1_5.6.7(1)	*The site specific information for tube repair is to be provided.*
COL 16.1_5.7(1)	*The site specific information about High Radiation Area is to be provided.*
COL 17.4(1)	*The COL Applicant shall be responsible for the development and implementation of the Phases II and III of the D-RAP,including QA requirements. In the Phase II, the plant's site-specific information should be introduced to the D-RAP process and the site-specific risk-significant SSCs should be combined with the US-APWR design risk-significant SSCs into a list for the specific plant. Phase II is performed during the COL application phase and updated/maintained during the COL license holder phase. In the Phase III, procurement, fabrication, construction, and test specifications for the SSCs within the scope of the RAP should ensure that significant assumptions, such as equipment reliability, are realistic and achievable. The QA requirements should be implemented during the procurement, fabrication, construction, and pre-operation testing of the SSCs within the scope of the RAP. Phase III is performed during the COL license holder phase and prior to initial fuel loading. The COL applicant will propose a method by which it will incorporate the objectives of the reliability assurance program into other programs for design or operational errors that degrade nonsafety-related, risk-significant SSCs.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 43 of 44)

COL ITEM NO.	COL ITEM
COL 17.4(2)	*The COL Applicant shall be responsible for the development and implementation of the O-RAP, in which the RAP activities should be integrated into the existing operational program (i.e., Maintenance Rule, surveillance testing, in-service inspection, in-service testing, and QA). The O-RAP should also include 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 performed during the COL application phase. The development/integration of the operational 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 D-RAP should be categorized as high-safety-significant (HSS) within the scope of initial Maintenance Rule.*
COL 17.5(1)	*The COL applicant shall develop and implement a Quality Assurance Program Description for site-specific design activities and for plant construction and operation.*
COL 17.6(1)	*The COL applicant must provide in its FSAR a description of the maintenance rule program, and its for implementation, for monitoring the effectiveness of maintenance necessary to meet the requirements of 10 CFR 50.65.*
COL 18.1(1)	*Deleted*
COL 18.1(2)	*Deleted*
COL 18.3(1)	*Deleted*
COL 18.3(2)	*Deleted*
COL 18.4(1)	*Deleted*
COL 18.4(2)	*Deleted*
COL 18.4(3)	*Deleted*
COL 18.5(1)	*Deleted*
COL 18.5(2)	*Deleted*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 44 of 44)

COL ITEM NO.	COL ITEM
COL 18.6(1)	*Deleted*
COL 18.6(2)	*Deleted*
COL 18.7(1)	*Deleted*
COL 18.8(1)	*Deleted*
COL 18.9(1)	*Deleted*
COL 18.10(1)	*Deleted*
COL 18.10(2)	*Deleted*
COL 18.11(1)	*Deleted*
COL 18.11(2)	*Deleted*
COL 18.12(1)	*Deleted*
COL 19.3(1)	*The COL Applicant who intends to implement risk-managed technical specifications continues to update Probabilistic Risk Assessment and Severe Accident Evaluation to provide PRA input for risk-managed technical specifications.*
COL 19.3(2)	*Deleted*
COL 19.3(3)	*Deleted*
COL 19.3(4)	*The Probabilistic Risk Assessment and Severe Accident Evaluation is updated as necessary to assess specific site information and associated site-specific external events (high winds and tornadoes, external floods, transportation, and nearby facility accidents).*
COL 19.3(5)	*Deleted*
COL 19.3(6)	*The COL applicant develops an accident management program which includes severe accident management procedures that capture important operator actions. Training requirements are also included as part of the accident management program.*

1.9 Conformance with Regulatory Criteria

In keeping with the requirements of Sections C.I.1.9.1 through C.I.1.9.5 of RG 1.206, and section I.9 of the NUREG 0800, Standard Review Plan, this section provides the following information:

- Section 1.9.1 - Conformance with Regulatory Guides
- Section 1.9.2 - Conformance with Standard Review Plan (SRP)
- Section 1.9.3 - Generic Issues
- Section 1.9.4 - Operational Experience (Generic Communications)
- Section 1.9.5 - Advanced and Evolutionary Light-Water Reactor Design Issues

The COL Applicant is to address an evaluation of the applicable RG, SRP, Generic Issues including Three Mile Island (TMI) requirements, and operational experience for the site-specific portion and operational aspect of the facility.

1.9.1 Conformance with Regulatory Guides

RG 1.206, Section C.I.1.9.1, "Conformance with Regulatory Guides," specifies that the following groups of Regulatory Guides shall be evaluated for purposes of determining US-APWR conformance:

- Division 1, Power Reactors
- Division 4, Environmental and Siting
- Division 5, Materials and Plant Protection
- Division 8, Occupational Health

From NUREG 0800, Standard Review Plan, Section 1.0, "Introduction and Interfaces", section I (Areas of Review) subsection 9 (Conformance with Regulatory Criteria), the following requirement is drawn: "Regulatory Guides - A table of conformance with the NRC's regulatory guides that are applicable to the application is reviewed. The table should also include an identification and description of deviations from the guidance contained in the NRC's regulatory guides, as well as suitable justifications for any alternative approaches proposed by the COL Applicant with appropriate references to the Final Safety Analysis Report (FSAR) sections where they are addressed."

Each of Tables 1.9.1-1 through 1.9.1-4 contains an conformance evaluation to the group of RGs contained in one of the four required RG divisions. The tables show the RG numbers, titles, status, and chapter, section, subsection of the US-APWR DCD which corresponds the particular RG.

The status of each item is reported as "Conformance with no exceptions identified," "Conformance with exceptions," or "Not applicable." For RGs evaluated to be not applicable, a brief reason for non-applicability is provided in the status column of the table. Also included in the status column of each table are any exceptions or alternative approaches proposed for the US-APWR, with technical justification provided.

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 1 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Rev. 0, November 1970)	Not applicable. SIP and CS/RHRP are designed so that adequate NPSH are provided to system pumps in accordance with Regulatory Guide 1.82 Rev.3.	N/A
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors (Rev. 2, June 1974)	Not applicable. The guidance of Regulatory Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents at Nuclear Power Reactors" is applied instead of Regulatory Guide 1.4.	N/A
1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (Rev. 0, March 1971)	Conformance with no exceptions identified.	8.1.5.3
1.7	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident (Rev. 3, March 2007)	Conformance with no exceptions identified.	6.2.5.1, 19.2
1.8	Qualification and Training of Personnel for Nuclear Power Plants (Revision 3, May 2000)	Not applicable. RG applies to a site-specific operational program.	12.1.1.3.1, 12.1.4
1.9	Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants (Rev. 4, March 2007)	Conformance with no exceptions identified. US-APWR has no diesel generators, but will use gas turbine generators for emergency power in the standard design.	8.1.5.3, 14.2.12 (Note: MHI has generated a position on the use of gas turbine generators for emergency power that meets the intent of RG)
1.11	Instrument Lines Penetrating Primary Reactor Containment (Rev. 0, March 1971)	Conformance with exceptions. Isolation valve is not adopted to instrument lines for containment pressure.	6.2.4.1
1.12	Nuclear Power Plant Instrumentation for Earthquakes (Rev. 2, March 1997)	Conformance with exception. Programmatic/operational aspect is not applicable to US-APWR design certification.	3.7.4
1.13	Spent Fuel Storage Facility Design Basis (Rev. 2, March 2007)	Conformance with no exceptions identified.	9.1.1, 9.1.2, 9.1.3, 9.1.4
1.14	Reactor Coolant Pump Flywheel Integrity (for Comment) (Rev. 1, August 1975)	Conformance with no exceptions identified.	5.4.1.1

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 2 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.16	Reporting of Operating Information – Appendix A Technical Specifications (Rev. 4, August 1975)	Conformance with exception. Programmatic/operational aspect is not applicable to US-APWR design certification.	Chapter 16, 14.2.6, 14.2.7
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing (Rev. 3, March 2007)	Conformance with exceptions. The measurement at startup test for SG's internals is not planned.	3.9.2.3, 3.9.2.4, 3.9.2.6, 5.4.2.1.2.10, 14.2,
1.21	Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants (Rev. 1, June 1974)	Conformance with exceptions. To be conformed by COL Applicant with site-specific information.	3.1.6, 11.5.1, 12.3.4
1.22	Periodic Testing of Protection System Actuation Functions (Rev. 0, February 1972)	Conformance with no exceptions identified.	7.1.3.11, 7.1.3.14, 8.1.5.3
1.23	Meteorological Monitoring Programs for Nuclear Power Plants (Rev. 1, March 2007)	Not applicable. To be conformed by COL Applicant with site-specific characterization information.	N/A
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure (Rev. 0, March 1972)	Conformance with exceptions. To be conformed by COL Applicant with site-specific characterization information.	11.3.3
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 (Rev. 0, March 1972)	Not applicable. The guidance of Regulatory Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents at Nuclear Power Reactors" is applied instead of Regulatory Guide 1.25.	N/A
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (Rev. 4, March 2007)	Conformance with no exceptions identified.	3.2.2, 5.2.1.1, 5.2.2.1, 5.2.4.1
1.27	Ultimate Heat Sink for Nuclear Power Plants (Rev. 2, January 1976)	Conformance with exceptions. US-APWR is designed in accordance with the functional requirements for a UHS as described in this RG, however design of the UHS is site-specific and will be the responsibility of the COL Applicant.	9.2.1.3, 9.2.5
1.28	Quality Assurance Program Requirements (Design and Construction) (Rev. 3, August 1985)	Conformance with no exceptions identified.	14.2.7, 17.5

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 3 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.29	Seismic Design Classification (Rev. 4, March 2007)	Conformance with no exceptions identified.	3.2.1, 5.2.5, 5.2.2.1, 5.4.11.1, 7.1.3.7, 8.1.5.3, 9.1.1, 9.1.2, 9.3.1
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Rev. 0, August 1972)	Conformance with exceptions. Installation is not included in Design Certification phase.	14.2.7, 17.5
1.31	Control of Ferrite Content in Stainless Steel Weld Metal (Rev. 3, April 1978)	Conformance with no exceptions identified.	4.5.2, 5.2.3.4.4, 5.3.1.4, 6.1.1
1.32	Criteria for Power Systems for Nuclear Power Plants (Rev. 3, March 2004)	Conformance with no exceptions identified.	8.1.5.3, 16.3
1.33	Quality Assurance Program Requirements (Operation) (Rev. 2, February 1978)	Conformance with exceptions. Implementation of RG applies to a site-specific operational program for which COL Applicant will be responsible.	12.1.3 13.5
1.34	Control of Electroslag Weld Properties (Rev. 0, December 1972)	Not applicable. Electroslag welding is not employed in structural welds of low alloy steel. Electroslag welding is only applied for cladding.	5.2.3.3.2, 5.2.3.4.4, 5.3.1.4
1.35	In-Service Inspection (ISI) of Ungrouted Tendons in Prestressed Concrete Containments (Rev. 3, July 1990)	Conformance with no exceptions identified.	3.8.1.2, 3.8.1.7, 14.2.7
1.35.1	Determining Prestressing Forces for Inspection of Prestressed Concrete Containments (Rev. 0, July 1990)	Conformance with no exceptions identified.	3.8.1.2, 3.8.1.7, 14.27
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel (Rev. 0, February 1973)	Conformance with no exceptions identified.	5.2.3.2, 6.1.1.2
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (Rev. 1, March 2007)	Conformance with exception. Programmatic/operational aspect is not applicable to US-APWR design certification.	3.13.1, 4.5.1, 5.2.3, 5.3.1, 6.1.1, 14.2.7
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants (Rev. 2, May 1977)	Not applicable. RG applies to a site-specific operational program.	N/A
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants (Rev. 2, September 1977)	Not applicable. RG applies to a site-specific operational program.	N/A
1.40	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants (Rev. 0, March 1973)	Not applicable. US-APWR has no Class 1 continuous-duty motors in the containment.	N/A

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 4 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.41	Preoperational Testing of Redundant On-Site Electric Power Systems To Verify Proper Load Group Assignments (Rev. 0, March 1973)	Conformance with no exceptions identified.	8.1.5.3, 14.2.7
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components (Rev. 0, May 1973)	Conformance with no exceptions identified.	5.2.3.3.2, 5.3.1.4
1.44	Control of the Use of Sensitized Stainless Steel (Rev. 0, May 1973)	Conformance with no exceptions identified.	3.6.3.3.4, 5.2.3.4.1, 5.2.3.4.2, 6.1.1
1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems (Rev. 1, May 2008)	Conformance with no exceptions Identified.	5.2.5
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems (Rev. 0, May 1973)	Conformance with no exceptions identified.	8.1.5.3, table 8.1-1, 18.7.3.2, table 18.7-1
1.49	Power Levels of Nuclear Power Plants (Rev. 1, December 1973)	This RG has been withdrawn by NRC.	N/A
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel (Rev. 0, May 1973)	Conformance with no exceptions identified.	5.3.1.2, 5.3.1.4, 5.2.3.3.2, 6.1.1
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. 3, June 2001)	Conformance with no exceptions identified.	6.4.2, 6.4.6, Table 6.4-2, 6.5.1, Table 6.5-3, 9.4.1, 9.4.5, 12.3.3, 14.2.7
1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems (Rev. 2, November 2003)	Conformance with no exceptions identified.	7.1.3. 2, 7.1.3.3, 8.1.5.3
1.54	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants (Rev. 1, July 2000)	Conformance with exceptions. Programmatic/operational and site-specific aspects are not applicable to US-APWR design certification. ASTM standard revision levels may differ from RG 1.54 as specifically referenced in the “Corresponding Chapter/Section/Subsection”	6.1.2 11.2 11.4
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (Rev. 1, March 2007)	Not applicable. US-APWR has a concrete containment.	N/A
1.59	Design Basis Floods for Nuclear Power Plants (Rev. 2, August 1977)	Conformance with exceptions. RG applies to a site-specific characterization for flooding.	2.4, 3.4.1.2
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants (Rev. 1, December 1973)	Conformance with no exceptions identified. Note: COL Applicant will verify site-specific data is bounded by data used in DCD analyses.	2.3, 2.5, 3.7
1.61	Damping Values for Seismic Design of Nuclear Power Plants (Rev. 1, March 2007)	Conformance with no exceptions identified.	3.7, 3.9.2, 3.12.3, 3.12.5.4, 3.12.6.8

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 5 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.62	Manual Initiation of Protective Actions (Rev. 0, October 1973)	Conformance with no exceptions identified.	8.1.5.3, 18.7.3.2, Table 7.2-6,7, Table 7.3-5,6
1.63	Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants (Rev. 3, February 1987)	Conformance with no exceptions identified.	3.11, 8.1.5.3
1.65	Materials and Inspections for Reactor Vessel Closure Studs (Rev. 0, October 1973)	Conformance with no exceptions identified.	3.13.1.1, 3.13.1.2, 3.13.2, 5.2.3.6, 5.3.1.7, 5.3.3, 16.0
1.68	Initial Test Programs for Water-Cooled Nuclear Power Plants (Rev. 3, March 2007)	Conformance with exception. Programmatic/operational aspect is not applicable to US-APWR design certification.	14.2
1.68.2	Initial Startup Test Program To Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants (Rev. 1, July 1978)	Conformance with no exceptions identified.	14.2
1.68.3	Preoperational Testing of Instrument and Control Air Systems (Rev. 0, April 1982)	Conformance with exceptions. C.7: This criterion applies to instrument and control air system important safety. US-APWR instrument and control air system is not important to safety.	9.3.1.4, 14.2.7
1.69	Concrete Radiation Shields for Nuclear Power Plants (Rev. 0, December 1973)	Conformance with exceptions. Criterion 5 is not applicable to US-APWR design certification. There is no plan which uses metal for the aggregate of concrete shielding.	12.3.2
1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition) (Rev. 3, November 1978)	Not applicable. Format and content for new reactors established in RG 1.206.	N/A
1.71	Welder Qualification for Areas of Limited Accessibility (Rev. 1, March 2007)	Conformance with no exceptions identified.	5.2.3.3.2, 5.2.3.4.4, 5.3.1.4, 10.3.6.2, 6.1.1
1.72	Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin (Rev. 2, November 1978)	Not applicable. US-APWR design does not use Spray Pond. The spray water of US-APWR is supplied from RWSP in containment.	N/A
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants (Rev. 0, January 1974)	Conformance with no exceptions identified.	8.1.5.3
1.75	Physical Independence of Electric Systems (Rev. 3, February 2005)	Conformance with no exceptions identified.	7.1.3, 8.1.5.3

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 6 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.76	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants (Rev. 1, March 2007)	Conformance with no exceptions identified. Note: COL Applicant will verify site-specific data is bounded by data used in DCD analyses.	2.3, table 2.0-1, 3.3.2, 3.5.1.4
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors (Rev. 0, May 1974)	Conformance with no exceptions identified. Note: The newer criteria for fuel cladding failure, core coolability and fission product inventory contained in SRP, Section 4.2, Appendix B will be used in conjunction with the requirements of RG 1.77 for the US-APWR Rod Ejection analysis.	15.4.8.2, 16.0
1.78	Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release (Rev. 1, December 2001)	Conformance with exceptions. Full conformance by COL Applicant with site-specific consequence data.	6.4.4, 9.4.1
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors (Rev. 1, September 1975)	Conformance with no exceptions identified.	14.2.7
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants (Rev. 1, January 1975)	Not applicable. DCD describes a single reference plant design; RG applies to a site-specific multi-unit situation.	N/A
1.82	Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident (Rev. 3, November 2003)	Conformance with exceptions. Full conformance by COL Applicant with site-specific conditions.	6.2.2.1
1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes (Rev. 1, July 1975)	Not applicable. This RG is considered for withdrawal by NRC. Current NRC requirements for this area are shown in steam generator program guidelines (NEI-97-06 Rev.2) and Technical Specification task Force TSTF-449 Rev.4.	5.4.2.2
1.84	Design, Fabrication, and Materials Code Case Acceptability, ASME Section III (Rev. 33, August 2005)	Conformance with no exceptions identified.	3.12.2.2
1.86	Termination of Operating Licenses for Nuclear Reactors (Rev. 0, June 1974)	Not applicable. RG applies to a later phase site-specific operational program.	N/A
1.87	Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, and 1596) (Rev. 0, June 1975)	Not applicable.	N/A

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 7 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.89	Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants (Rev. 1, June 1984)	Conformance with no exceptions identified.	3.11, 7.1.3, 8.1.5.3
1.90	Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons (Rev. 1, August 1977)	Not applicable. US-APWR is not among the designs covered by this RG. US-APWR PCCV tendon type is Ungrouted.	N/A
1.91	Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants (Rev. 1, February 1978)	Not applicable. RG applies to a site-specific analysis.	N/A
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis (Rev. 2, July 2006)	Conformance with no exceptions identified.	3.7.2, 3.8.1, 3.12.3.2.4, 3.12.5.5
1.93	Availability of Electric Power Sources (Rev. 0, December 1974)	Conformance with no exceptions identified.	8.1.5.3
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (Rev. 1, April 1976)	Not applicable. RG applies to a site-specific operational program.	N/A
1.97	Criteria For Accident Monitoring Instrumentation For Nuclear Power Plants (Rev. 4, June 2006)	Conformance with no exceptions identified.	7.5.1.1, 7.5.2.1, 14.3.4
1.99	Radiation Embrittlement of Reactor Vessel Materials (Rev. 2, May 1988)	Conformance with no exceptions identified.	5.3.1, 5.3.2, 16.0
1.100	Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants (Rev. 2, June 1988)	Conformance with no exceptions identified.	3.10
1.101	Emergency Planning and Preparedness for Nuclear Power Reactors (Rev. 5, June 2005)	Conformance with exceptions. Full conformance by COL Applicant with site-specific EP data	13.3.5
1.102	Flood Protection for Nuclear Power Plants (Rev. 1, September 1976)	Conformance with exceptions. Full conformance by COL Applicant with site-specific flood data	3.4.1
1.105	Setpoints for Safety-Related Instrumentation (Rev. 3, December 1999)	Conformance with no exceptions identified.	7.2.2.7, 7.3.2.7
1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves (Rev. 1, March 1977)	Conformance with no exceptions identified.	8.1.5.3
1.107	Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures (Rev. 1, February 1977)	Not applicable. For the US-APWR, cement grout is not applied to prestressing tendons.	N/A
1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFRPart 50, Appendix I (Rev. 1, October 1977)	Conformance with exceptions. To be conformed by COL Applicant with site-specific meteorological information.	11.2.3, 11.2.4, 11.3.3, 11.3.7, 11.4.3, 11.5.2

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 8 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors (Rev. 0, March 1976)	Conformance with exceptions. The RG describes a licensee-specific evaluation based on reduction of radiological doses within 50 miles of a specific proposed site.	11.2.1, 11.3.1, 11.4.1, 11.5.2
1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors (Rev. 1, July 1977)	Not applicable. Full conformance by COL Applicant with site-specific dispersion data.	N/A
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors (Rev. 1, March 2007)	Conformance with no exceptions identified.	11.1.3
1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I (Rev. 1, April 1977)	Conformance with exceptions. To be conformed by COL Applicant with site-specific meteorological information.	11.2.4, 11.5.2
1.114	Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit (Rev. 2, May 1989)	Not applicable. RG applies to a site-specific operational program.	N/A
1.115	Protection Against Low-Trajectory Turbine Missiles (Rev. 1, July 1977)	Conformance with no exceptions identified	3.5.1.3
1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems (Rev. 0-R, May 1977)	Conformance with exceptions. Installation is not included in Design Certification phase.	17.5, 14.2.7
1.117	Tornado Design Classification (Rev. 1, April 1978)	Conformance with no exceptions identified.	3.3
1.118	Periodic Testing of Electric Power and Protection Systems (Rev. 3, April 1995)	Conformance with no exceptions identified.	7.1.3.14, 8.1.5.3, 14.2.7
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes (Rev. 0, August 1976	Conformance with no exceptions identified.	5.4.2.1.8, 5.4.2.2.2
1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components (Rev. 1, February 1978)	Conformance with no exceptions identified.	3.7.2, 3.12.3.2.1
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Supports (Rev. 2, February 2007)	Conformance with exceptions. Criterion 5:OBE seismic evaluation is not required in US-APWR.	3.9.3.4, 3.12.6.1
1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants (Rev. 1, October 1978)	Conformance with no exceptions identified.	2.4
1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification (Rev. 1, March 1978)	Conformance with no exceptions identified	4.2, 4.4

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 9 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants (Rev. 1, March 1978)	Not applicable. RG applies to a site-specific operational program.	N/A
1.128	Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants (Rev. 2, February 2007)	Conformance with exceptions. The hydrogen concentration limit required in RG 1.189 is appropriate for the fire protection scenario, over the RG 1.128.	8.1.5.3, 14.2.7
1.129	Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants (Rev. 2, February 2007)	Conformance with exceptions. Design certification applicability is to assure design features accommodate functions described in RG; full conformance in terms of program and activities will be the responsibility of the COL Applicant.	8.1.5.3
1.130	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports (Rev. 2, March 2007)	Conformance with no exceptions identified.	3.9.3.4, 3.12.6.1
1.131	Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants (Rev. 0, August 1977)	Conformance with no exceptions identified.	8.1.5.3
1.132	Site Investigations for Foundations of Nuclear Power Plants (Rev. 2, October 2003)	Not applicable. RG applies to site-specific operational program.	N/A
1.133	Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors (Rev. 1, May 1981)	Conformance with exceptions. C.3.a: Section 13.5 defines the responsibility for development of administrative and operating procedures. C.6: The COL applicant has the responsibility of this requirement.	4.4.6.3
1.134	Medical Evaluation of Licensed Personnel at Nuclear Power Plants (Rev. 3, March 1998))	Not applicable. RG applies to a site-specific operational program.	N/A
1.135	Normal Water Level and Discharge at Nuclear Power Plants (Rev. 0, September 1977)	Conformance with exception. Site-specific aspect is not applicable to US-APWR design certification.	2.4
1.136	Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments (Rev. 3, March 2007)	Conformance with no exceptions identified.	3.8.1.2, 14.2.7

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 10 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.137	Fuel-Oil Systems for Standby Diesel Generators (Rev. 1, October 1979)	Conformance with no exceptions identified. US-APWR has no diesel generators, but uses gas turbine generators for emergency power in the standard design.	8.1.5.3, 9.5.4 (Note: MHI has generated a position on the use of gas turbine generators for emergency power that meets the intent of RG.)
1.138	Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants (Revision 2, December 2003)	Not applicable. RG applies to site-specific operational program.	N/A
1.139	Guidance for Residual Heat Removal (for Comment (Rev. 0, May 1978) (Note: Cold shutdown requirements as related to environmental qualification of equipment)	Conformance with exceptions. Criterion 7 applies to a site-specific operational program.	5.4.7, 6.3.1.3, 7.4.1
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, June 2001)	Conformance with no exceptions identified.	9.4.3, 9.4.6, 12.3.3, 14.2.7
1.141	Containment Isolation Provisions for Fluid Systems (Rev. 0, April 1978)	Conformance with no exceptions identified.	6.2.4
1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments) (Rev. 2, November 2001)	Conformance with no exceptions identified.	3.5.3, 3.8.3, 3.8.4, 3.8.5
1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants (Rev. 2, November 2001)	Conformance with no exceptions identified.	3.2.2, 11.2 11.3, 11.4
1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants (Rev. 1, November 1982)	Not applicable. Full conformance by COL Applicant with site-specific dispersion data.	N/A
1.147	Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1 (Rev. 14, August 2005)	Conformance with no exceptions identified.	5.2.1.2, 5.2.4.1, 5.2.4.2, 6.6.3
1.148	Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants (Rev. 0, March 1981)	Conformance with no exceptions identified.	3.9.6, 3.10.2
1.149	Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations (Rev. 3, October 2001)	Not applicable. US-APWR reference design does not include a simulator.	N/A
1.150	Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations (Rev. 1, February 1983)	Withdrawn.	
1.151	Instrument Sensing Lines (Rev. 0, July 1983)	Conformance with no exceptions identified.	7.1.3.7

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 11 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.152	Criteria for Digital Computers in Safety Systems of Nuclear Power Plants (Rev. 2, January 2006)	Conformance with no exceptions identified.	7.9.2.6
1.153	Criteria for Safety Systems (Rev. 1, June 1996)	Conformance with no exceptions identified.	7.1.2, 8.1.5.3
1.154	Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors (Rev. 0, January 1987)	Not applicable.	N/A
1.155	Station Blackout (Rev. 0, August 1988)	Conformance with no exceptions identified.	8.1.5.3
1.156	Environmental Qualification of Connection Assemblies for Nuclear Power Plants (Rev. 0, November 1987)	Conformance with no exceptions identified.	8.1.5.3
1.157	Best-Estimate Calculations of Emergency Core Cooling System Performance (Rev. 0, May 1989)	Conformance with exceptions. C.3.13-14 applies to BWR only.	15.6.5
1.158	Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants (Rev. 0, February 1989)	Conformance with no exceptions identified.	8.1.5.3
1.159	Assuring the Availability of Funds for Decommissioning Nuclear Reactors (Rev. 1, October 2003)	Not applicable. RG applies to a site-specific operational program.	N/A
1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants (Rev. 2, March 1997)	Not applicable. RG applies to a site-specific operational program.	N/A
1.161	Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb (Rev. 0, June 1995)	Not applicable. RG applies to a site-specific operational program.	N/A
1.162	Format and Content of Report for Thermal Annealing of Reactor Pressure Vessel (Rev. 0, February 1996)	Not applicable. RG applies to a site-specific program to restore strength to the reactor vessel lost by irradiation.	N/A
1.163	Performance-Based Containment Leak-Test Program (Rev. 0, September 1995)	Conformance with no exceptions identified.	6.2.6, 14.2
1.165	Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion (Rev. 0, March 1997)	Not applicable. RG refers to a site-specific seismic characterization to be provided by COL Applicant.	N/A
1.166	Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions (Rev. 0, March 1997)	Not applicable. RG applies to a site-specific operational program.	N/A
1.167	Restart of a Nuclear Power Plant Shut Down by a Seismic Event (Rev. 0, March 1997)	Not applicable. RG applies to a site-specific operational program.	N/A

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 12 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.168	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 1, February 2004)	Conformance with no exceptions identified.	7.1.3.6
1.169	Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	Conformance with no exceptions identified.	7.1.3.17
1.170	Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	Conformance with no exceptions identified.	7.1.3.17
1.171	Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	Conformance with no exceptions identified.	7.1.3.17
1.172	Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	Conformance with no exceptions identified.	7.1.3.6, 7.1.3.17,
1.173	Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	Conformance with no exceptions identified.	7.1.3.17
1.174	An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis (Rev. 1, November 2002)	Not applicable. RG applies to a site-specific analysis.	N/A
1.175	An Approach for Plant-Specific, Risk-Informed Decision making: Inservice Testing (Rev. 0, August 1998)	Not applicable. RG applies to a site-specific analysis.	N/A
1.176	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance (Rev. 0, August 1998)	Withdrawn	
1.177	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications (Rev. 0, August 1998)	Not applicable. RG applies to a site-specific analysis.	N/A
1.178	An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping (Rev. 1, September 2003)	Not applicable. RG applies to a site-specific analysis.	N/A
1.179	Standard Format and Content of License Termination Plans for Nuclear Power Reactors (Rev. 0, January 1999)	Not applicable. RG applies to a site-specific operational program.	N/A
1.180	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems (Rev. 1, October 2003)	Conformance with no exceptions identified.	7.1.3.7, 8.1.5.3, 9.5.2
1.181	Content of the Updated Final Safety Analysis Report in Accordance with 10CFR50.71(e) (Rev. 0, September 1999)	Not applicable. RG applies to a site-specific operational program.	N/A
1.182	Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants (Rev. 0, May 2000)	Not applicable. RG applies to a site-specific operational program.	N/A

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 13 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.183	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors (Rev. 0, July 2000)	Conformance with exception. Site-specific aspect is not applicable to US-APWR design certification.	12.2.1.3, 12.3.2.2.7, 12.4.1.8, 12.3.1.2.2, 15.0.3, 15.1.5.5, 15.3.3.5, 15.4.8.5, 15.6.2, 15.6.3.5, 15.6.5.5, 15.7.4, Appendix 15A
1.184	Decommissioning of Nuclear Power Reactors (Rev. 0, July 2000)	Not applicable. RG applies to a site-specific operational program.	N/A
1.185	Standard Format and Content for Post-Shutdown Decommissioning Activities Report (Rev. 0, July 2000)	Not applicable. RG applies to a site-specific operational program.	N/A
1.186	Guidance and Examples for Identifying 10CFR50.2 Design Bases (Rev. 0, December 2000)	Not applicable.	N/A
1.187	Guidance for Implementation of 10CFR50.59, Changes, Tests, and Experiments (Rev. 0, November 2000)	Not applicable. RG applies to a site-specific operational program.	N/A
1.188	Standard Format and Content for Applications To Renew Nuclear Power Plant Operating Licenses (Rev. 1, September 2005)	Not applicable. RG applies to a site-specific operational program.	N/A
1.189	Fire Protection for Nuclear Power Plants (Rev. 1, March 2007)	Conformance with exceptions. See table 9.5-1 and DCD section 9.5.1 for a point-by-point discussion of conformance with the RG.	9.5.1, table 9.5-1
1.190	Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence (Rev. 0, March 2001)	Conformance with exceptions. Criterion 3: Not applicable. This criterion specifies requirements for descriptions of the report, neither calculation nor measurement methodology.	4.3.2.8, 5.3.1.4, 5.3.1.6
1.191	Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown (Rev. 0, May 2001)	Not applicable. RG applies to a site-specific operational program that occurs during plant decommissioning and permanent shutdown.	N/A
1.192	Operation and Maintenance Code Case Acceptability, ASME OM Code (Rev. 0, June 2003)	Conformance with exceptions. RG is referenced specifically for applicability of code case OMN-13 requirements for snubber inspection.	3.9.3.4

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 14 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.193	ASME Code Cases Not Approved for Use (Rev. 1, August 2005)	Not applicable. US-APWR design does not incorporate any of the identified ASME code cases.	N/A
1.194	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants (Rev. 0, June 2003)	Not applicable. Full conformance by COL Applicant with site-specific dispersion data	N/A
1.195	Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors (Rev. 0, May 2003)	Not applicable. Due to use of alternative source term, the guidance of RG 1.183 is applied instead of RG 1.195.	N/A
1.196	Control Room Habitability at Light-Water Nuclear Power Reactors (Rev. 1, January 2007)	Conformance with no exceptions identified.	6.4
1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors (Rev. 0, May 2003)	Conformance with no exceptions identified.	6.4.5
1.198	Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites (Rev. 0, November 2003)	Not applicable. RG applies to a site-specific analysis.	N/A
1.199	Anchoring Components and Structural Supports in Concrete (Rev. 0, November 2003)	Conformance with no exceptions identified.	3.8.4, 3.9.3.4, 3.12.6.4
1.200	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities (Rev. 1, January 2007)	Conformance with no exceptions identified.	19.1
1.201	Guidelines for Categorizing SSCs in Nuclear Power Plants According to Their Safety Significance (Rev. 1, May 2006)	Conformance with exceptions. Note: No.3 and No.9 are the NRC position to meet the requirement of 10CFR50.69, but design of US-APWR is in accordance with the standard method described in RG 1.206, section C.I.3.	17.4, 19.1.7
1.202	Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors (Rev. 0, February 2005)	Not applicable. RG applies to a site-specific operational program.	N/A
1.203	Transient and Accident Analysis Methods (Rev. 0, December 2005)	Not applicable. US-APWR design certification does not fall into any of the 3 categories described in the RG “Implementation” section; US-APWR follows the transient/accident analysis requirements of RG 1.206, section C.I.15.	N/A
1.204	Guidelines for Lightning Protection of Nuclear Power Plants (Rev. 0, November 2005)	Conformance with no exceptions identified.	8.1.5.3

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 15 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.205	Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants (Rev. 0, May 2006)	Not applicable. RG applies to a site-specific operational program.	N/A
1.206	Combined License Applications for Nuclear Power Plants (LWR Edition) (Rev. 0, June 2007)	Conformance with exceptions. Section C.II.2,CIII, and CIV are guidance for COL application referencing a certified design and/or an early site permit (ESP). These sections conform in the COL application. The US-APWR is not the passive-ALWR-plant. Therefore, section C.IV.9 is not applicable to the US-APWR.	All chapters and appendices
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 (Rev. 0, March 2007)	Conformance with no exceptions identified.	3.12.5
1.208	A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion (Rev. 0, March 2007)	Not applicable. RG applies to a site-specific analysis.	N/A
1.209	Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants (Rev. 0, March 2007)	Conformance with no exceptions identified.	7.1.3.7

Table 1.9.1-2 US-APWR Conformance with Division 4 Regulatory Guides (sheet 1 of 2)

Reg Guide Number	Title	Status	Corresponding Chapter/Section/ Subsection
4.1	Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants (Rev. 1, April 1975)	Not applicable. RG applies to a site-specific environmental monitoring activity.	N/A
4.2	Preparation of Environmental Reports for Nuclear Power Stations (Rev. 2, July 1976)	Not applicable. RG applies to a site-specific environmental evaluation.	N/A
4.2S1	Supplement 1 to Regulatory Guide 4.2, Preparation of Supplemental Environmental Reports for Applications To Renew Nuclear Power Plant Operating Licenses (Rev. 0, September 2000)	Not applicable. RG applies to license renewals.	N/A
4.4	Reporting Procedure for Mathematical Models Selected To Predict Heated Effluent Dispersion in Natural Water Bodies (Rev. 0, May 1974)	Not applicable. RG applies to site-specific environmental activity of modeling temperature impact of plant discharge on aquatic systems.	N/A
4.5	Measurements of Radionuclides in the Environment--Sampling and Analysis of Plutonium in Soil (Rev. 0, May 1974)	Not applicable. RG applies to a site-specific environmental monitoring activity.	N/A
4.6	Measurements of Radionuclides in the Environment-- Strontium-89 and Strontium-90 Analyses (Rev. 0, May 1974)	Not applicable. RG applies to a site-specific environmental monitoring activity.	N/A
4.7	General Site Suitability Criteria for Nuclear Power Stations (Revision 2, April 1998)	Not applicable. RG applies to a site-specific evaluation	N/A
4.8	Environmental Technical Specifications for Nuclear Power Plants (Rev. 0, December 1975)	Not applicable. RG applies to a site-specific operational controls resulting from environmental characterization and commitments.	N/A
4.9	Preparation of Environmental Reports for Commercial Uranium Enrichment Facilities (Rev. 1, October 1975)	Not applicable. RG applies to uranium enrichment facilities.	N/A
4.11	Terrestrial Environmental Studies for Nuclear Power Stations (Rev. 1, August 1977)	Not applicable. RG applies to a site-specific environmental evaluation.	N/A
4.13	Performance, Testing, and Procedural Specifications for Thermoluminescence Dosimetry: Environmental Applications (Rev. 1, July 1977)	Not applicable. RG applies to a site-specific dosimetry product.	N/A
4.14	Radiological Effluent and Environmental Monitoring at Uranium Mills (Rev. 1, April 1980)	Not applicable. RG applies to uranium mills.	N/A

Table 1.9.1-2 US-APWR Conformance with Division 4 Regulatory Guides (sheet 2 of 2)

Reg Guide Number	Title	Status	Corresponding Chapter/Section/ Subsection
4.15	Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) -- Effluent Streams and the Environment (Rev. I-2, March 2007)	Not applicable. RG applies to a site-specific operational program that will be the responsibility of the COL Applicant.	N/A
4.16	Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexafluoride Production Plants (Rev. 1, December 1985)	Not applicable. RG applies to non-reactor production facilities.	N/A
4.17	Standard Format and Content of Site Characterization Plans for High-Level-Waste Geologic Repositories (Rev. 1, March 1987)	Not applicable. RG applies to waste repositories.	N/A
4.18	Standard Format and Content of Environmental Reports for Near-Surface Disposal of Radioactive Waste (Rev. 0, June 1983)	Not applicable. RG applies to waste disposal sites.	N/A
4.19	Guidance for Selecting Sites for Near-Surface Disposal of Low-Level Radioactive Waste (Rev. 0, August 1988)	Not applicable. RG applies to waste disposal sites.	N/A
4.20	Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors (Rev. 0, December 1996)	Not applicable. RG applies to non-reactor facilities.	N/A

Table 1.9.1-3 US-APWR Conformance with Division 5 Regulatory Guides (sheet 1 of 6)

Reg Guide Number	Title	Status	Corresponding Chapter/Section/ Subsection
5.3	Statistical Terminology and Notation for Special Nuclear Materials Control and Accountability (Rev. 0, February 1973)	Not applicable. RG applies to fuel processing and fuel fabrication facilities.	N/A
5.4	Standard Analytical Methods for the Measurement of Uranium Tetrafluoride (UF4) and Uranium Hexafluoride (UF6) (Rev. 0, February 1973)	Not applicable. RG describe processes and procedures that would not be performed at a US-APWR.	N/A
5.5	Standard Methods for Chemical, Mass Spectrometric, and Spectrochemical Analysis of Nuclear-Grade Uranium Dioxide Powders and Pellets (Rev. 0, February 1973)	Not applicable. RG describe processes and procedures that would not be performed at a US-APWR.	N/A
5.7	Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas	Conformance with no exceptions identified.	13.6.2
5.8	Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Drying and Fluidized Bed Operations (Rev. 1, May 1974)	Not applicable. RG describe processes and procedures that would not be performed at a US-APWR	N/A
5.9	Guidelines for Germanium Spectroscopy Systems for Measurement of Special Nuclear Material (Rev. 2, December 1983)	Not applicable. RG describe processes and procedures that would not be performed at a US-APWR.	N/A
5.10	Selection and Use of Pressure-Sensitive Seals on Containers for Onsite Storage of Special Nuclear Material (Rev. 0, July 1973)	Not applicable. No containerized storage of special nuclear material is proposed in the reference US-APWR design.	N/A
5.11	Nondestructive Assay of Special Nuclear Material Contained in Scrap and Waste (Rev. 1, April 1984)	Not applicable. RG applies to facilities that process special nuclear material.	N/A
5.12	General Use of Locks in the Protection and Control of Facilities and Special Nuclear Material	Conformance with no exceptions identified.	13.6.2
5.13	Conduct of Nuclear Material Physical Inventories (Rev. 0, November 1973)	Not applicable. RG introduction specifically exempts nuclear reactor operating licenses from the requirements of the RG.	N/A

Table 1.9.1-3 US-APWR Conformance with Division 5 Regulatory Guides (sheet 2 of 6)

Reg Guide Number	Title	Status	Corresponding Chapter/Section/ Subsection
5.15	Tamper-Indicating Seals for the Protection and Control of Special Nuclear Material (Rev. 1, March 1997)	Not applicable. RG refers to a site-specific transportation requirement.	N/A
5.17	Truck Identification Markings (Rev. 0, January 1974)	Not applicable. RG applies to over the road shippers of special nuclear material.	N/A
5.18	Limit of Error Concepts and Principles of Calculation in Nuclear Materials Control (Rev. 0, January 1974)	Not applicable. RG applies to procedural controls that are outside scope of reference design.	N/A
5.20	Training, Equipping, and Qualifying of Guards and Watchmen (Rev. 0, January 1974)	Not applicable. Site-specific security programs are not addressed in reference US-APWR design.	N/A
5.21	Nondestructive Uranium-235 Enrichment Assay by Gamma Ray Spectrometry (Rev. 1, December 1983)	Not applicable. RG describes processes and procedures that would not be conducted at a US-APWR.	N/A
5.22	Assessment of the Assumption of Normality (Employing Individual Observed Values) (Rev. 0, April 1974)	Not applicable. RG describes processes and procedures that would not be conducted at a US-APWR.	N/A
5.23	In Situ Assay of Plutonium Residual Holdup (Rev. 1, February 1984)	Not applicable. RG describes processes and procedures that would not be conducted at a US-APWR.	N/A
5.25	Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Equipment for Wet Process Operations (Rev. 0, June 1974)	Not applicable. RG applies to processes performed on special nuclear materials at chemical conversion, fuel fabrication, scrap recovery, and fuel reprocessing facilities.	N/A

Table 1.9.1-3 US-APWR Conformance with Division 5 Regulatory Guides (sheet 3 of 6)

Reg Guide Number	Title	Status	Corresponding Chapter/Section/ Subsection
5.26	Selection of Material Balance Areas and Item Control Areas (Rev. 1, April 1975)	Not applicable. RG describes processes and procedures that would not be conducted at a US-APWR.	N/A
5.27	Special Nuclear Material Doorway Monitors (Rev. 0, June 1974)	Not applicable. Site-specific physical protection not addressed in reference US-APWR design.	N/A
5.28	Evaluation of Shipper-Receiver Differences in the Transfer of Special Nuclear Materials (Rev. 0, June 1974)	Not applicable. RG applies to procedural controls that are outside scope of reference design.	N/A
5.31	Specially Designed Vehicle with Armed Guards for Road Shipment of Special Nuclear Material (Rev. 1, April 1975)	Not applicable. RG applies to over-the-road shippers of special nuclear material.	N/A
5.32	Communication with Transport Vehicles (Rev. 1, May 1975)	Not applicable. Site-specific physical protection features and security programs are not addressed in reference US-APWR design.	N/A
5.33	Statistical Evaluation of Material Unaccounted For (Rev. 0, June 1974)	Not applicable. RG references sections 70.51(e) and 70.53(b)(1), which no longer exist.	N/A
5.34	Nondestructive Assay for Plutonium in Scrap Material by Spontaneous Fission Detection (Rev. 1, May 1984)	Not applicable. RG applies to fuel processing and fuel fabrication facilities.	N/A
5.36	Recommended Practice for Dealing with Outlying Observations (Rev. 0, June 1974)	Not applicable. Site-specific physical protection features and security programs are not addressed in reference US-APWR design.	N/A
5.37	In Situ Assay of Enriched Uranium Residual Holdup (Rev. 1, October 1983)	Not applicable. RG describes processes and procedures that would not be conducted at a US-APWR.	N/A

Table 1.9.1-3 US-APWR Conformance with Division 5 Regulatory Guides (sheet 4 of 6)

Reg Guide Number	Title	Status	Corresponding Chapter/Section/ Subsection
5.38	Nondestructive Assay of High-Enrichment Uranium Fuel Plates by Gamma Ray Spectrometry (Rev. 1, October 1983)	Not applicable. RG describes processes and procedures that would not be conducted at a US-APWR.	N/A
5.39	General Methods for the Analysis of Uranyl Nitrate Solutions for Assay, Isotopic Distribution, and Impurity Determinations (Rev. 0, December 1974)	Not applicable. RG describes processes and procedures that would not be conducted at a US-APWR.	N/A
5.42	Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Equipment for Dry Process Operations (Rev. 0, January 1975)	Not applicable. RG applies to process facilities.	N/A
5.43	Plant Security Force Duties (Rev. 0, January 1975)	Not applicable. RG implements 10CFR73.50, which does not apply to power reactor licensees.	N/A
5.44	Perimeter Intrusion Alarm Systems	Conformance with no exceptions identified.	13.6.2
5.48	Design Considerations--Systems for Measuring the Mass of Liquids (Rev. 0, February 1975)	Not applicable. RG describes processes and procedures that would not be conducted at a US-APWR.	N/A
5.49	Internal Transfers of Special Nuclear Material (Rev. 0, March 1975)	Not applicable. RG describes processes and procedures that would not be conducted at a US-APWR.	N/A
5.51	Management Review of Nuclear Material Control and Accounting Systems (Rev. 0, June 1975)	Not applicable. RG describes processes and procedures that would not be conducted at a US-APWR.	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, December 1994)	Not applicable. RG does not apply to power reactors.	N/A

Table 1.9.1-3 US-APWR Conformance with Division 5 Regulatory Guides (sheet 5 of 6)

Reg Guide Number	Title	Status	Corresponding Chapter/Section/ Subsection
5.53	Qualification, Calibration, and Error Estimation Methods for Nondestructive (Rev. 1, February 1984)	Not applicable. RG relates to a requirement 10CFR70.58(f), specific to measurement bias, that no longer exists.	N/A
5.54	Standard Format and Content of Safeguards Contingency Plans for Nuclear Power Plants (Rev. 0, March 1978)	Not applicable. Site-specific physical protection features and security programs are not addressed in reference US-APWR design.	N/A
5.55	Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities (Rev. 0, March 1978)	Not applicable. RG applies to fuel cycle facilities and does not include power reactors.	N/A
5.56	Standard Format and Content of Safeguards Contingency Plans for Transportation (Rev. 0, March 1978)	Not applicable. RG describes processes and procedures that would not be conducted at a US-APWR.	N/A
5.57	Shipping and Receiving Control of Strategic Special Nuclear Material (Rev. 1, June 1980)	Not applicable. Site-specific physical protection features and security programs are not addressed in reference US-APWR design.	N/A
5.58	Considerations for Establishing Traceability of Special Nuclear Material Accounting Measurements (Rev. 1, February 1980)	Not applicable. RG describes processes and procedures that would not be conducted at a US-APWR.	N/A
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, February 1983)	Not applicable. Site-specific physical protection features and security programs are not addressed in reference US-APWR design.	N/A
5.60	Standard Format and Content of a Licensee Physical Protection Plan for Strategic Special Nuclear Material in Transit (Rev. 0, April 1980)	Not applicable. RG describes processes and procedures that would not be conducted at a US-APWR.	N/A

Table 1.9.1-3 US-APWR Conformance with Division 5 Regulatory Guides (sheet 6 of 6)

Reg Guide Number	Title	Status	Corresponding Chapter/Section/ Subsection
5.61	Intent and Scope of the Physical Protection Upgrade Rule Requirements for Fixed Sites (Rev. 0, June 1980)	Not applicable. RG describes physical protection requirements for fuel cycle facilities and transportation, which do not apply to US-APWR.	N/A
5.62	Reporting of Safeguards Events (Rev. 1, November 1987)	Not applicable. Site-specific security programs are not addressed in reference US-APWR design.	N/A
5.63	Physical Protection for Transient Shipments (Rev. 0, July 1982)	Not applicable. RG describes a special nuclear material possession scenario not included in the US-APWR reference design.	N/A
5.65	Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls	Conformance with no exceptions identified.	13.6.2
5.66	Access Authorization Program for Nuclear Power Plants (Rev. 0, June 1991)	Not applicable. Site-specific security programs are not addressed in reference US-APWR design.	N/A
5.67	Material Control and Accounting for Uranium Enrichment Facilities Authorized To Produce Special Nuclear Material of Low Strategic Significance (Rev. 0, December 1993)	Not applicable. RG addresses enrichment facilities.	N/A
5.68	Protection Against Malevolent Use of Vehicles at Nuclear Power Plants in general Use of Locks in the protection and controls of Facilities and Special Nuclear Material	Conformance with no exceptions identified.	13.6.2
5.69	Guidance for the Implementation of the Radiological Sabotage Design-Basis Threat	Conformance with no exceptions identified.	13.6.2

Table 1.9.1-4 US-APWR Conformance with Division 8 Regulatory Guides (sheet 1 of 4)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
8.2	Guide for Administrative Practices in Radiation Monitoring (Rev. 0, February 1973)	Conformance with exceptions. To be conformed in COL.	12.1.4, 12.3.4
8.4	Direct-Reading and Indirect-Reading Pocket Dosimeters (Rev. 0, February 1973)	Conformance with exceptions. To be conformed in COL.	12.1.4
8.5	Criticality and Other Interior Evacuation Signals (Rev. 1, March 1981)	Not applicable. RG refers to site-specific procedures and/or equipment that are outside the reference US-APWR design.	N/A
8.6	Standard Test Procedure for Geiger-Muller Counters (Rev. 0, May 1973)	Conformance with exceptions. To be conformed in COL.	12.1.4
8.7	Instructions for Recording and Reporting Occupational Radiation Exposure Data (Rev. 2, November 2005)	Conformance with exceptions. To be conformed in COL.	12.1.4
8.8	Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable (Rev. 3, June 1978)	Conformance with exceptions. All design issues are addressed; site-specific policy considerations are outside scope of design certification.	3.7.4.2, 9.3.2, 11.3.1, 11.4.1, 11.4.2, , 12.1.1.3, 12.1.2, 12.1.4, 12.2.1.1.10, 12.3.1, 12.3.2.1, 12.3.2.2, 12.3.3.3, 12.3.4,
8.9	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program (Rev. 1, July 1993)	Not applicable. RG refers to site-specific procedures and/or equipment that are outside the reference US-APWR design.	N/A (RG is mentioned, however, in 12.1.4)
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable (Rev. 1-R, May 1977)	Conformance with exceptions. Programmatic/operational aspect is not applicable to US-APWR design certification.	12.1.1.3, 12.1.4, 12.2.1.1.10
8.11	Applications of Bioassay for Uranium (Rev. 0, June 1974)	Not applicable. RG applies to bioassay for uranium.	N/A

Table 1.9.1-4 US-APWR Conformance with Division 8 Regulatory Guides (sheet 2 of 4)

Reg Guide Number	Title	Status	Corresponding Chapter/Section/ Subsection
8.13	Instruction Concerning Prenatal Radiation Exposure (Rev. 3, June 1999)	Not applicable. RG refers to site-specific procedures and/or equipment that are outside the reference US-APWR design.	N/A
8.15	Acceptable Programs for Respiratory Protection (Rev. 1, October 1999)	Not applicable. RG refers to site-specific procedures and/or equipment that are outside the reference US-APWR design.	N/A
8.18	Information Relevant to Ensuring that Occupational Radiation Exposures at Medical Institutions Will Be as Low as Reasonably Achievable (Rev. 1, October 1982)	Not applicable. RG applies to medical institutions.	N/A
8.19	Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants -- Design Stage Man-Rem Estimates (Rev. 1, June 1979)	Conformance with no exceptions identified	12.4
8.20	Applications of Bioassay for I-125 and I-131 (Rev. 1, September 1979)	Not applicable. This RG is outdated according to NUREG-1736	N/A
8.21	Health Physics Surveys for Byproduct Material at NRC-Licensed Processing and Manufacturing Plants (Rev. 1, October 1979)	Not applicable. RG applies to processing and manufacturing plants.	N/A
8.22	Bioassay at Uranium Mills (Revision 1, August 1988)	Not applicable. RG applies to uranium mills.	N/A
8.23	Radiation Safety Surveys at Medical Institutions (Rev. 1, January 1981)	Not applicable. RG applies to medical institutions.	N/A
8.24	Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication (Rev. 1, October 1979)	Not applicable. RG applies to uranium processing and fuel fabrication.	N/A
8.25	Air Sampling in the Workplace (Rev. 1, June 1992)	Not applicable. RG refers to site-specific procedures and/or equipment that are outside the reference US-APWR design.	N/A
8.26	Applications of Bioassay for Fission and Activation Products (Rev. 0, September 1980)	Not applicable. This RG is outdated according to NUREG-1736	N/A

Table 1.9.1-4 US-APWR Conformance with Division 8 Regulatory Guides (sheet 3 of 4)

Reg Guide Number	Title	Status	Corresponding Chapter/Section/ Subsection
8.27	Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants (Rev. 0, March 1981)	Not applicable. RG applies to training about radiation protection.	N/A
8.28	Audible-Alarm Dosimeters (Rev. 0, August 1981)	Not applicable. RG refers to site-specific procedures and/or equipment that are outside the reference US-APWR design.	N/A
8.29	Instruction Concerning Risks from Occupational Radiation Exposure (Rev. 1, February 1996)	Not applicable. RG refers to site-specific procedures and/or equipment that are outside the reference US-APWR design.	N/A
8.30	Health Physics Surveys in Uranium Recovery Facilities (Rev. 1, May 2002)	Not applicable. RG applies to uranium recovery facilities.	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, May 2002)	Not applicable. RG applies to uranium recovery facilities.	N/A
8.32	Criteria for Establishing a Tritium Bioassay Program (Rev. 0, July 1988)	Not applicable. This RG is outdated according to NUREG-1736	N/A
8.33	Quality Management Program (Rev. 0, October 1991)	Not applicable. RG applies to medical use of by-product material.	N/A
8.34	Monitoring Criteria and Methods To Calculate Occupational Radiation Doses (Rev. 0, July 1992)	Not applicable. RG refers to site-specific procedures and/or equipment that are outside the reference US-APWR design.	N/A
8.35	Planned Special Exposures (Rev. 0, June 1992)	Not applicable. RG refers to site-specific procedures and/or equipment that are outside the reference US-APWR design.	N/A
8.36	Radiation Dose to the Embryo/Fetus (Rev. 0, July 1992)	Not applicable. RG refers to site-specific procedures and/or equipment that are outside the reference US-APWR design.	N/A

Table 1.9.1-4 US-APWR Conformance with Division 8 Regulatory Guides (sheet 4 of 4)

Reg Guide Number	Title	Status	Corresponding Chapter/Section/ Subsection
8.37	ALARA Levels for Effluents from Materials Facilities (Rev. 0, July 1993)	Not applicable. RG applies to material facilities.	N/A
8.38	Control of Access to High and Very High Radiation Areas of Nuclear Plants (Rev. 1, May 2006)	Conformance with exceptions. Conformance about the view of the high radiation zone of handling, but responsible for COL Applicant about actual employment.	12.3.1.2.1.2, 12.1.4
8.39	Release of Patients Administered Radioactive Materials (Rev. 0, April 1997)	Not applicable. RG applies to administration of radio-pharmaceuticals to medical patients.	N/A

1.9.2 Conformance with Standard Review Plan

Language cited from RG 1.206 Section C.I.1.9.2 and from Standard Review Plan 1.0, “Introduction and Interfaces,” Section I.9, states that applicants should evaluate their facilities against the NRC Standard Review Plan in effect 6 months before the docket date of the application. The SRP conformance evaluation presented in this section was performed using the revision of the SRP dated September 2007. This section of the US-APWR DCD is a series of SRP evaluation tables that corresponds to the 19 chapters of the DCD, as follows:

Table 1.9.2-1	Chapter 1	Introduction and General Description of the Plant
Table 1.9.2-2	Chapter 2	Site Characteristics
Table 1.9.2-3	Chapter 3	Design of Structures, Systems, Components, and Equipment
Table 1.9.2-4	Chapter 4	Reactor
Table 1.9.2-5	Chapter 5	Reactor Coolant and Connecting Systems
Table 1.9.2-6	Chapter 6	Engineered Safety Features
Table 1.9.2-7	Chapter 7	Instrumentation and Controls
Table 1.9.2-8	Chapter 8	Electrical Power
Table 1.9.2-9	Chapter 9	Auxiliary Systems
Table 1.9.2-10	Chapter 10	Steam and Power Conversion System
Table 1.9.2-11	Chapter 11	Radioactive Waste Management
Table 1.9.2-12	Chapter 12	Radiation Protection
Table 1.9.2-13	Chapter 13	Conduct of Operations
Table 1.9.2-14	Chapter 14	Verification Programs
Table 1.9.2-15	Chapter 15	Transient and Accident Analyses
Table 1.9.2-16	Chapter 16	Technical Specifications
Table 1.9.2-17	Chapter 17	Quality Assurance and Reliability Assurance
Table 1.9.2-18	Chapter 18	Human Factors Engineering
Table 1.9.2-19	Chapter 19	Probabilistic Risk Assessment and Severe Accident Evaluation.

Each of the table entries presents the section and title from the SRP (NUREG 0800), excerpts from the SRP text that describe the acceptance criteria for that section, a status column, and a column that indicates where the topic appears in the DCD.

The status of each item is reported as "Conformance with no exceptions", "Conformance with exceptions", or "Not applicable". Status is indicated for the whole SRP section whose number and title are indicated in the left hand column of the table. For SRP sections evaluated to be not applicable, a brief reason for non-applicability is provided in the status column of the table. Also included in the status column of each table are any exceptions or alternative approaches proposed for the US-APWR, with technical justification provided.

Table 1.9.2-1 US-APWR Conformance with Standard Review Plan Chapter 1 Introduction and General Description of the Plant

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
1.0 Introduction and Interfaces	1. There are no specific SRP acceptance criteria associated with these general requirements. 2. For the regulatory considerations, acceptance is based on addressing the regulatory requirements as discussed within this FSAR section or within the referenced FSAR section. The SRP acceptance criteria associated with the referenced section will be reviewed within the context of that review. 3. For performance of new safety features, the information is sufficient to provide a reasonable assurance that (1) these new safety features will perform as predicted in the applicant's FSAR, (2) the effects of system interactions are acceptable, and (3) the applicant provides sufficient data to validate analytical codes. The design qualification testing requirements may be met with either separate effects or integral system tests; prototype tests; or a combination of tests, analyses, and operating experience.	Conformance with no exceptions identified. There are no specific SRP acceptance criteria associated with general requirements. Section 1.9 of US-APWR DCD describes conformance with regulatory criteria.	Chapter 1

Table 1.9.2-2 US-APWR Conformance with Standard Review Plan Chapter 2 Site Characteristics (sheet 1 of 17)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
2.0 Site Characteristics and Site Parameters	For a DC application, the Chapter 2 review is focused on site-related design characteristics and postulated site parameters for the design. A subset of the site parameters will become part of the certified design. Previous certified designs have used the designations Tier 1 and Tier 2 for delineating the portion of design-related information that is approved and certified, and the portion that is approved but not certified, respectively. Site parameters are included in both Tier 1 and Tier 2 information. This section should summarize the complete set of site parameters and the subset of site parameters that will be included within the certified design – the top-level bounding site parameters used to define a suitable site for a facility referencing the certified design. Because site parameters were used in bounding evaluations of the certified design, they define the requirements for the design that must be met by a site. [Review guidance for Tier 1 site parameters was previously included in draft SRP Section 14.3.1, “Site Parameters (Tier 1).”] Examples of site-related design characteristics and site parameters that should be addressed are included in Tables 1 and 2 of Appendix A to this SRP section. The applicant has selected the site-related design characteristics and site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information), and the staff agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application. Accordingly, the staff concludes that the site parameters meet the requirements of 10CFR52.47(a)(1)(iii).	Conformance with no exceptions identified. Note: Per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA. Chapter 2.0 of the DCD contains specific site parameter requirements necessary to meet the engineering and design needs for safe construction and operation of the US-APWR.	2.0
2.1.1 Site Location and Description	3. Design Certification Reviews - There are no postulated site parameters for a DC related to this SRP section. The site location and description is site-specific and will be addressed by the COL Applicant.	Conformance with no exceptions identified. Note: Per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA. There are no postulated site parameters in the US-APWR DCD. The US-APWR design assumes that each site-specific COLA will include the detailed information as required in RG 1.206	2.1.1

Table 1.9.2-2 US-APWR Conformance with Standard Review Plan Chapter 2 Site Characteristics (sheet 2 of 17)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
2.1.2 Exclusion Area Authority and Control	3. Design Certification Reviews - Exclusion area authority and control is site-specific and will be addressed by the COL Applicant.	Conformance with no exceptions identified. Note: per expectations expressed in the SRP, site-specific characterization data will be contained in the COLA. Section 2.1.2 of the US-APWR DCD describes that the US-APWR COLA will include detailed information of site characteristics.	2.1.2
2.1.3 Population Distribution	3. Design Certification Reviews - The population distribution is site-specific and will be addressed by the COL Applicant.	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA. Section 2.1.3 of the US-APWR DCD describes that the US-APWR COLA will include detailed information of site characteristics.	2.1.3
2.2.1-2.2.2 Identification of Potential Hazards in Site Vicinity	3. Design Certification Reviews - The identification of potential hazards in the site vicinity is site-specific and will be addressed by a COL Applicant.	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site specific characterization data will be contained in the COLA. DC applications do not contain general descriptions of site characteristics because this information is site-specific and will be addressed by a COL applicant. Section 2.2.1 of the US-APWR DCD describes that the US-APWR COLA will include detailed information of site characteristics.	2.2.1, 2.2.2

Table 1.9.2-2 US-APWR Conformance with Standard Review Plan Chapter 2 Site Characteristics (sheet 3 of 17)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
2.2.3 Evaluation of Potential Accidents	3. Design Certification Reviews - The evaluation of potential accidents is site-specific and will be addressed by the COL Applicant.	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA. Section 2.2.3 of the US-APWR DCD describes that the US-APWR COLA will include detailed information of site characteristics.	2.2.3
2.3.1 Regional Climatology	1. Description of the general climate of the region 2. Data on severe weather phenomena 3. Tornado parameters should be based on Regulatory Guide 1.76 4. Basic (straight-line) 100-year return period 3-second gust wind speed 5. Maximum evaporation and drift loss of water and minimum water cooling 6. Winter precipitation loads 7. Ambient temperature and humidity 8. High air pollution potential information 9. All other meteorological and air quality conditions identified by the applicant as climate site characteristics for ESP applications or used as design and operating bases for CP, OL, or COL applications should be documented and substantiated. NRC desired conclusion for a DC review: “The applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information), and staff agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application. The regional climatology is site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the values of the actual site characteristics specified in a COL or CP application.”	Conformance with exceptions. Section 2.3.1 of US-APWR DCD describes that the regional climatology is site-specific and will be defined in the COLA. The site parameters postulated for the design is described in Table 2.01 of US-APWR DCD.	2.3.1

Table 1.9.2-2 US-APWR Conformance with Standard Review Plan Chapter 2 Site Characteristics (sheet 4 of 17)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
2.3.2 Local Meteorology	1. Local summaries of meteorological data 2. A complete topographical description of the site and environs out to a distance of 80 kilometers 3. Discussion and evaluation of the influence of the plant and its facilities on the local meteorological and air quality 4. Description of local site airflow should include wind roses and annual joint frequency distributions of wind speed and wind direction by atmospheric stability NRC desired conclusion for a DC review: "The NRC staff acknowledges that the applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information) and agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application. Technical specifications and emergency operations are site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the site parameters specified by the siting review."	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA. Section 2.3.2 of US-APWR DCD describes that the regional climatology is site-specific and will be defined in the COLA.	2.3.2
2.3.3 Onsite Meteorological Measurements Programs	3. Design Certification Reviews - There are no postulated site parameters for a DC related to this SRP section. The onsite meteorological monitoring program is site-specific and will be addressed by the COL Applicant.	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA. Section 2.3.3 of US-APWR DCD describes that the onsite meteorological measurements program is site-specific and will be supplied in the COLA.	2.3.3

Table 1.9.2-2 US-APWR Conformance with Standard Review Plan Chapter 2 Site Characteristics (sheet 5 of 17)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
2.3.4 Short Term Atmospheric Dispersion Estimates for Accident Releases	The staff should ensure that the DC applicant has included EAB, LPZ, and control room atmospheric dispersion factors for the appropriate time periods in the list of site parameters. The DC application should also contain figures and tables showing the design features that would be used by the COL Applicant to generate control room χ/Q values (e.g., intake heights, release heights, building cross sectional areas, distance to receptors). If any straight-line horizontal distances of less than 10 meters from a release location to the environment to a receptor have been proposed, the staff should attempt to impress upon the applicant that it is good engineering practice to design and maintain some distance of separation (e.g., more than 10 meters) between potential release pathways and potential intake pathways to the control room. NRC desired conclusion for a DC review: “The applicant has selected the short-term (post-accident) site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information), and the staff agrees that (except for the control room χ/Q values) they are representative of a reasonable number of sites that have been or may be considered for a COL application. The short-term atmospheric dispersion characteristics for accidental release are site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the values of the actual site characteristics specified in a COL or CP application.”	Conformance with no exceptions identified. Note: Per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA.	2.0, 2.3.4
2.3.5 Long-Term Atmospheric Dispersion Estimates for Routine Releases	The staff should ensure that the DC applicant has included the maximum annual average site boundary χ/Q value in the list of site parameters. NRC desired conclusion for a DC review: “The applicant has selected the long term (routine release) site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information),and staff agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application. The long term atmospheric dispersion and deposition characteristics are site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the values of the actual site characteristics specified in a COL or CP application.”	Conformance with no exceptions identified. Note: Per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA.	2.0, 2.3.5

Table 1.9.2-2 US-APWR Conformance with Standard Review Plan Chapter 2 Site Characteristics (sheet 6 of 17)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
2.4.1 Hydrologic Description	1. Interface of the Plant with the Hydrosphere 2. Hydrological Causal Mechanisms 3. Surface and Ground Water Uses 4. Data: The application should provide a complete description of all spatial and temporal datasets used by the applicant in support of its conclusions regarding safety of the plant. 5. Alternate Conceptual Models of site hydrology 6. Consideration of Other Site-Related Evaluation Criteria: The application should demonstrate that the potential effects of site-related proximity and of seismic and non-seismic information as they relate to hydrologic description in the vicinity of the proposed plant site and site regions are appropriately taken into account. NRC desired conclusion for a DC review: "The NRC staff acknowledges that the applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information), and agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application. Site hydrology descriptions are site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the site parameters specified by the siting review."	Conformance with no exceptions identified. Note: per expectations expressed in the SRP, site-specific characterization data will be contained in the COLA. Section 2.4.1 of the US-APWR DCD describes that the US-APWR COLA will describe site-specific flood.	2.4, table 2.0-1
2.4.2 Floods	1. Local Flooding on the Site and Drainage Design 2. Stream Flooding 3. Surges 4. Seiches 5. Tsunami 6. Seismically Induced Dam Failures (or Breaches 7. Flooding Caused by Landslides 8. Effects of Ice Formation in Water Bodies 9. Combined Events Criteria 10. Consideration of Other Site-Related Evaluation Criteria: The application should demonstrate that the potential effects of site-related proximity, seismic, and non-seismic information as they relate to hydrologic description in the vicinity of the proposed plant site and site regions are appropriately taken into account.	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA. Section 2.4.2 of the US-APWR DCD describes that the floods are site-specific and will be defined in the COLA.	2.4, table 2.0-1

Table 1.9.2-2 US-APWR Conformance with Standard Review Plan Chapter 2 Site Characteristics (sheet 7 of 17)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
2.4.2 Floods (continued)	NRC desired conclusion for a DC review: “The NRC staff acknowledges that the applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information), and agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL causal mechanisms, and the controlling flooding application. The local intense precipitation, flooding mechanism are site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the site parameters specified by the siting review. For DC and COL reviews, the findings will also summarize (to the extent that the review is not discussed in other SER sections) the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable, and interface requirements and combined license action items relevant to this SRP section.”		
2.4.3 Probable Maximum Flood (PMF) on Streams and Rivers	1. Design Bases for Flooding in Streams and Rivers 2. Design Bases for Site Drainage 3. Consideration of Other Site-Related Evaluation Criteria. To meet the requirements of GDC 2, 10CFR52.17 and 10CFRPart 100 information about the potential effects of site-related proximity, seismic, and non-seismic information as they relate to flooding in streams and rivers and local flooding adjacent to and on the plant site is needed. NRC desired conclusion for a DC review: “The NRC staff acknowledges that the applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information), but agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application. Probable maximum flood on streams and rivers and flooding of site drainage are site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the site parameters specified by the siting review.”	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA. Section 2.4.3 of the US-APWR DCD describes that the potential for floods on streams and rivers are site-specific and will be defined in the COLA.	2.4, table 2.01-1

Table 1.9.2-2 US-APWR Conformance with Standard Review Plan Chapter 2 Site Characteristics (sheet 8 of 17)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
2.4.4 Potential Dam Failures	1. Flood Waves from Severe Breaching of an Upstream Dam: 2. Domino-Type or Cascading Dam Failures 3. Dynamic Effects on Structures 4. Loss of Water Supply Due to Failure of a Downstream Dam 5. Effects of Sediment Deposition and Erosion 6. Failure of Onsite Water Control or Storage Structures 7. Consideration of Other Site-Related Evaluation Criteria: The potential effects of site-related proximity, seismic, and non-seismic information as they relate to flooding due to upstream dam failures and loss of safety-related water supply due to blockages and failures of downstream dam failures adjacent to and on the plant site and site regions are needed to meet the requirements of GDC 2, 10CFR52.17, and 10CFRPart 100. NRC desired conclusion for a DC review: "The NRC staff acknowledges that the applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information), but agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application. Dam failures are site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the site parameters specified by the siting review"	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA. Section 2.4.4 of the US-APWR DCD describes that the potential for seismically induced dam failures are site-specific and will be defined in the COLA.	2.4, table 2.0-1
2.4.5 Probable Maximum Surge and Seiche Flooding	1. Probable Maximum Hurricane 2. Probable Maximum Wind Storm 3. Seiche and Resonance 4. Wave Runup 5. Effects of Sediment Erosion and Deposition 6. Consideration of Other Site-Related Evaluation Criteria. The potential effects of site-related proximity, seismic, and non-seismic information as they relate to flooding and loss of safety-related water supply due to surge and seiche adjacent to the plant site and site regions are needed to meet the requirements of GDC 2, 10CFR52.17, and 10CFRPart 100.	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA. Section 2.4.5 of the US-APWR DCD describes that potential for surge and seiche effects on US-APWR safety-related facilities are site-specific and will be defined in the COLA.	2.4, table 2.0-1

Table 1.9.2-2 US-APWR Conformance with Standard Review Plan Chapter 2 Site Characteristics (sheet 9 of 17)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
2.4.5 Probable Maximum Surge and Seiche Flooding (continued)	NRC desired conclusion for a DC review: "The NRC staff acknowledges that the applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information) and agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application. Surge and seiche are site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the site parameters specified by the siting review."		
2.4.6 Probable Maximum Tsunami Flooding	1. Historical Tsunami Data 2. Probable Maximum Tsunami 3. Tsunami Propagation Models 4. Wave Runup, Inundation, and Drawdown 5. Hydrostatic and Hydrodynamic Forces 6. Debris and Water-Borne Projectiles 7. Effects of Sediment Erosion and Deposition 8. Consideration of Other Site-Related Evaluation Criteria. The application should provide an evaluation of the potential effects of site-related proximity, seismic, and non-seismic information as they affect tsunamis near the plant site and site regions. This assessment should be sufficient to demonstrate that the applicant's design bases appropriately account for these effects. NRC desired conclusion for a DC review: "The NRC staff acknowledges that the applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information), and agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application. Probable maximum tsunami hazards are site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the site parameters specified by the siting review."	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA. Section 2.4.6 of the US-APWR DCD describes that the potential for tsunami effects are site-specific and will be defined in the COLA.	2.4, table 2.0-1

Table 1.9.2-2 US-APWR Conformance with Standard Review Plan Chapter 2 Site Characteristics (sheet 10 of 17)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
2.4.7 Ice Effects	1. Safety-related Facilities Exposed to Flooding 2. Type of Flood Protection. 3. Emergency Procedures 4. Consideration of Other Site-Related Evaluation Criteria. To meet the requirements of GDC 2, 10CFR52.17, and 10CFRPart 100, an assessment regarding the potential effects of site-related proximity, seismic, and non-seismic information on the postulated flooding protection is needed. The assessment should be sufficient to demonstrate that the applicant's design bases appropriately account for these effects. NRC desired conclusion for a DC review: "The NRC staff acknowledges that the applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information) and that they are representative of a reasonable number of sites that have been or may be considered for a COL application. Icing effects are site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the site parameters specified by the siting review."	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA. Section 2.4.7 of the US-APWR DCD describes that the potential for ice effects are site-specific and will be defined in the COLA.	2.4, table 2.0-1
2.4.8 Cooling Water Canals and Reservoirs	1. Hydraulic Design Bases for Protection of Structures 2. Hydraulic Design Bases of Canals 3. Hydraulic Design Bases of Reservoirs 4. Consideration of Other Site-Related Evaluation Criteria: To meet the requirements of GDC 1, GDC 2, 10CFR52.17, and 10CFRPart 100, a complete description of the potential effects of site-related proximity, seismic, and non-seismic information on the postulated design bases of safety-related canals and reservoirs is needed. This description should be sufficient to demonstrate that the applicant's design bases appropriately account for these effects. NRC desired conclusion for a DC review: "The NRC staff acknowledges that the applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information) and agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application. The [identify applicable site parameter] is site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the values of the actual site characteristics specified in a COL or CP application."	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA. Section 2.4.8 of the US-APWR DCD describes conditions relative to and affecting cooling water canals and reservoirs are site-specific and will be defined in the COLA.	2.4, table 2.0-1

Table 1.9.2-2 US-APWR Conformance with Standard Review Plan Chapter 2 Site Characteristics (sheet 11 of 17)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
2.4.9 Channel Diversions	1. Historical Channel Diversions 2. Regional Topographic Evidence 3. Ice Causes. 4. Flooding of Site Due to Channel Diversions. 5. Human-Induced Causes of Channel Diversion 6 Alternate Water Sources. 7. Consideration of Other Site-Related Evaluation Criteria: To meet the requirements of GDC 1, GDC 2, 10CFR52.17, and 10CFRPart 100, a description of the potential effects of site-related proximity, seismic, and non-seismic information on the postulated worst-case channel diversion scenario for the proposed plant site is needed. This description should be sufficient to demonstrate that the applicant's design bases appropriately account for these effects. NRC desired conclusion for a DC review: "The NRC staff acknowledges that the applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information) and agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application. Channel diversion effects are site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the site parameters specified by the siting review."	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA. Section 2.4.9 of the US-APWR DCD describes conditions relative to channel diversions and effects on the safety basis are site-specific and will be defined in the COLA.	2.4, table 2.0-1
2.4.10 Flooding Protection Requirements	1. Safety-related Facilities Exposed to Flooding 2. Type of Flood Protection 3. Emergency Procedures 4. Consideration of Other Site-Related Evaluation Criteria. To meet the requirements of GDC 2, 10CFR52.17, and 10CFRPart 100, an assessment regarding the potential effects of site-related proximity, seismic, and non-seismic information on the postulated flooding protection is needed. The assessment should be sufficient to demonstrate that the applicant's design bases appropriately account for these effects.	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA.	2.4, table 2.0-1

Table 1.9.2-2 US-APWR Conformance with Standard Review Plan Chapter 2 Site Characteristics (sheet 12 of 17)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
2.4.10 Flooding Protection Requirements (continued)	NRC desired conclusion for a DC review: "The NRC staff acknowledges that the applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information) and agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application Flood protection measures are site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the site parameters specified by the siting review."	Section 2.4.10 of the US-APWR DCD describes flooding protection requirements are site-specific and will be defined in the COLA.	
2.4.11 Low Water Considerations	1. Low Water from Drought 2. Low Water from Other Phenomena 3. Effect of Low Water on Safety-Related Water Supply 4. Water Use Limits 5. Consideration of Other Site-Related Evaluation Criteria: To meet the requirements of GDC 2, 10CFR52.17, and 10CFRPart 100, the applicant should provide an assessment of the potential effects of site-related proximity, seismic, and non-seismic information on the postulated worst-case low-flow scenario for the proposed plant site. This assessment should be sufficient to demonstrate that the applicant's design bases appropriately account for these effects. NRC desired conclusion for a DC review: "The NRC staff acknowledges that the applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information) and agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application. Low water effects are site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the site parameters specified by the siting review."	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site specific characterization data will be contained in the COLA. The COLA will describe low water conditions and describe how this volume of cooling water will be available for safety basis events. However, numerical value of an average cooling water volume requirement for 30 days following design-basis earthquake (DBE) shutdown is described in the Tier 1 information and Table 2.0-2 as DCD bases.	2.4, table 2.0-1

Table 1.9.2-2 US-APWR Conformance with Standard Review Plan Chapter 2 Site Characteristics (sheet 13 of 17)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
2.4.12 Groundwater	1. Local and Regional Groundwater Characteristics and Use 2. Effects on Plant Foundations and other Safety-Related SSCs 3. Reliability of Groundwater Resources and Systems Used for Safety-Related Purposes 4. Reliability of Dewatering Systems 5. Consideration of Other Site-Related Evaluation Criteria: To meet the requirements of 10CFR50.55a, GDC 2, GDC 4, GDC 5, 10CFR100.20(c)(3), 10CFR100.23(d), and 10CFR100.10(c) or 100.20(c), the applicant's assessment of the potential effects of site-related proximity, seismic, and non-seismic information on the postulated worst case scenario related to groundwater effects for the proposed plant site is needed. This assessment should be sufficient to demonstrate that the applicant's design bases appropriately account for these effects. NRC desired conclusion for a DC review: "The NRC staff acknowledges that the applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information) and agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application. Effects of groundwater in the vicinity of the site are site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the site parameters specified by the siting review."	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA. Section 2.4.12 of the US-APWR DCD describes groundwater are site-specific and will be defined in the COLA.	2.4, table 2.0-1
2.4.13 Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	1. Alternate Conceptual Models 2. Pathways 3. Characteristics that Affect Transport 4. Consideration of Other Site-Related Evaluation Criteria: The applicant's assessment of the potential effects of site-proximity hazards, seismic, and non-seismic events on the radioactive concentration from the postulated tank failure related to accidental release of radioactive liquid effluents to ground and surface waters for the proposed plant site is needed. This assessment should be sufficient to demonstrate that the applicant's design bases appropriately account for these effects.	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA.	2.4, table 2.0-1

Table 1.9.2-2 US-APWR Conformance with Standard Review Plan Chapter 2 Site Characteristics (sheet 14 of 17)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
2.4.13 Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters (continued)	5. Branch Technical Position BTP 11-6 provides guidance in assessing a potential release of radioactive liquids following the postulated failure of a tank and its components, located outside of containment, and impacts of the release of radioactive materials at the nearest potable water supply, located in an unrestricted area, for direct human consumption or indirectly through animals, crops, and food processing. NRC desired conclusion for a DC review: "The NRC staff acknowledges that the applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information) and agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application. Effects of accidental releases of radioactive liquid effluents in ground and surface waters on existing users and known and likely future users of ground and surface water resources in the vicinity of the site are site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the site parameters specified by the siting review."	Section 2.4.13 of the US-APWR DCD describes pathways and transport characteristics of Liquid Effluents in Ground and Surface Waters issues relative to the US-APWR are site-specific. The COLA will address these issues. The COL Applicant will list the inventory of potential onsite radionuclides that could affect groundwater.	
2.4.14 Technical Specifications and Emergency Operation Requirements	1. Bases for Emergency Actions 2. Available Response Time 3. Technical Specifications. 4. Consideration of Other Site-Related Evaluation Criteria: To meet the requirements of 10CFR50.36, GDC 2, 10CFR52.17, and 10CFR100, the applicant's assessment of the potential effects of site-related proximity, seismic, and non-seismic information on the postulated technical specifications and emergency operations is needed. This assessment should be sufficient to demonstrate that the applicant's analyses appropriately account for these effects. 5. 10CFR50, Appendix A, General Design Criterion (GDC) 44, for CP and OL applications, as it relates to providing an ultimate heat sink for normal operating and accident conditions.	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA. Section 2.4.14 of the US-APWR DCD describes the COLA will define the requirements and technical specification needs for emergency operations requirements relative to hydrologic conditions.	2.4, table 2.0-1

Table 1.9.2-2 US-APWR Conformance with Standard Review Plan Chapter 2 Site Characteristics (sheet 15 of 17)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
2.4.14 Technical Specifications and Emergency Operation Requirements (continued)	NRC desired conclusion for a DC review: “The NRC staff acknowledges that the applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information) and agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application. Technical specifications and emergency operations are site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the site parameters specified by the siting review.”		
2.5.1 Basic Geologic and Seismic Information	There are no postulated site parameters for a DC related to this SRP section. Geologic and seismic information is site-specific and will be addressed by the COL Applicant.	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA. Section 2.5.1 of the US-APWR DCD describes the COL Applicant will define Site-specific geological and seismic information.	2.5, Table 2.0-1
2.5.2 Vibratory Ground Motion	1. Seismicity 2. Geologic and Tectonic Characteristics of Site and Region 3. Correlation of Earthquake Activity with Seismic Sources of seismic 4. Probabilistic Seismic Hazard Analysis and Controlling Earthquakes 5. Seismic Wave Transmission Characteristics of the Site 6. Ground Motion Response Spectra NRC desired conclusion for a DC review: “The applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information), and the staff agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application. Local and regional geologic and seismic parameters are specific to the site and region and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the values of the actual site characteristics specified in a COL or CP application.”	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA.	Table 2.0-1, 2.4, 2.5, 3.7.1

Table 1.9.2-2 US-APWR Conformance with Standard Review Plan Chapter 2 Site Characteristics (sheet 16 of 17)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
2.5.3 Surface Faulting	1. Geologic, Seismic, and Geophysical Investigations 2. Geologic Evidence, or Absence of Evidence, for Surface Tectonic Deformation 3. Correlation of Earthquakes with Capable Tectonic Sources 4. Ages of Most Recent Deformation 5. Relationship of Tectonic Structures in the Site Area to Regional Tectonic Structures. 6. Characterization of Capable Tectonic Sources 7. Designation of Zones of Quaternary Deformation in the Site Region 8. Potential for Surface Tectonic Deformation at the Site Location NRC desired conclusion for a DC review: "The applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information), and the staff agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application. Local and regional geologic and seismic parameters are specific to the site and region and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the values of the actual site characteristics specified in a COL or CP application.	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA. Section 2.5.3 of the US-APWR DCD describes the COLA will verify that no surface faulting is within the site-specific exclusion area.	Table 2.0-1
2.5.4 Stability of Subsurface Materials and Foundations	1. Geologic Features 2. Properties of Subsurface Materials 3. Foundation Interfaces 4. Geophysical Surveys 5. Excavation and Backfill 6. Ground Water Conditions 7. Response of Soil and Rock to Dynamic Loading 8. Liquefaction Potential 9. Earthquake Design Basis 10. Static Stability 11. Design Criteria 12. Techniques to Improve Subsurface Conditions	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site-specific characterization data will be contained in the COLA. Section 2.5.4 of the US-APWR DCD describes site envelope design criteria for subsurface materials and foundation conditions. The COLA will include a discussion to verify that site-specific conditions meet the COLA Items.	Table 2.0-1, 2.5.4, 3.8

Table 1.9.2-2 US-APWR Conformance with Standard Review Plan Chapter 2 Site Characteristics (sheet 17 of 17)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
2.5.4 Stability of Subsurface Materials and Foundations (continued)	NRC desired conclusion for a DC review: "The applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information), and the staff agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application. The stability of subsurface materials and foundations is site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the values of the actual site characteristics specified in a COL or CP application."		
2.5.5 Stability of Slopes	1. Slope Characteristics 2. Design Criteria and Analyses 3. Boring Logs 4. Compacted Fill NRC desired conclusion for a DC review: "The applicant has selected the site parameters referenced above for plant design inputs (a subset of which is included as Tier 1 information), and the staff agrees that they are representative of a reasonable number of sites that have been or may be considered for a COL application. The stability of slopes is site-specific and will be addressed by the COL Applicant. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the values of the actual site characteristics specified in a COL or CP application."	Not applicable. There are no postulated site parameters for a DC related to this SRP section. Stability of slopes is site-specific and will be addressed by the COL applicant.	Table 2.0-1

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 1 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.2.1 Seismic Classification	1. Requirements of GDC 2, 10CFRPart 100, Appendix A, and 10CFRPart 50, Appendix S, regarding seismic design classification, are met by using guidance provided in RG 1.29 "Seismic Design Classification." This guide describes acceptable method of identifying and classifying those plant features that should be designed to withstand the effects of the SSE. RG 1.151 provides guidance with regard to seismic design requirements and classification of safety-related instrumentation sensing lines. RG 1.143 provides guidance used to establish the seismic design requirements of radioactive waste management SSCs to meet the requirements of GDC 2 and 61 as they relate to designing these SSCs to withstand earthquakes. The guide identifies several radioactive waste SSCs requiring some level of seismic design consideration. RG 1.189 provides guidance used to establish the design requirements of fire protection to meet the requirements of GDC 2 as it relates to designing these SSCs to withstand earthquakes. This guide identifies portions of fire protection SSCs requiring some level of seismic design consideration.	Conformance with no exceptions identified.	2.5.2, 3.2.1, 3.2.2, 3.7, 3.8, 3.9, 3.10, 3.11, 3.12, 9.5.1, 14.3, 17.5
3.2.2 System Quality Group Classification	To meet the requirements of GDC 1 and 10CFR50.55a, the following regulatory guide is used: 1. RG 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants." This guide describes an acceptable method for determining quality standards for Quality Group B, C, and D water- and steam-containing components important to safety of water-cooled nuclear power plants.	Conformance with no exceptions identified.	3.2.2,3.9.6
3.3.1 Wind Loading	1. The wind used in the design shall be the most severe wind that has been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which historical data have been accumulated. 2. The acceptance criteria for the design wind speed, its recurrence interval, the speed variation with height, the applicable gust factors, and the bases for determining these site-related parameters, are stated in SRP Sections 2.3.1 and 2.3.2. The approved values of these parameters should serve as basic input to the review and evaluation of the structural design procedures.	Conformance with exceptions. COL Applicant is to verify that site specific wind speed is enveloped by DCD windspeed, and CAT I structures, systems and components are not adversely impacted by site specific SSCs.	3.3.1

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 2 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.3.1 Wind Loading (continued)	3. The procedures used to transform the wind speed into an equivalent pressure to be applied to structures and parts, or portions of structures, as delineated in American Society of Civil Engineers/Structural Engineering Institute (ASCE/SEI) 7-05, “Minimum Design Loads for Buildings and Other Structures,” are acceptable. In particular, the procedures used are acceptable if found in accordance with the following….		
3.3.2 Tornado Loads	1. The tornado wind and associated missiles generated by the tornado wind used in the design shall be the most severe wind that has been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which historical data have been accumulated. 2. The acceptance criteria for tornado parameters including maximum wind speed, translational speed, rotational speed, and atmospheric pressure change, and the bases for determining these parameters are defined in SRP Sections 2.3.1 and 2.3.2. Acceptance criteria for the spectrum of tornado-generated missiles and their characteristics, as well as the bases for determining these parameters, are defined in SRP Section 3.5.1.4. These parameters should serve as basic input to the review and evaluation for structural design. 3. The acceptance criteria for procedures used to transform tornado parameters into equivalent loads on structures are as follows: • Tornado Characteristics and Effects • Tornado Wind Effects • Atmospheric Pressure Change Effects • Tornado-Generated Missile Impact Effects • Combined Tornado Effects 4. The information provided to demonstrate that failure of any structure or component not designed for tornado loads will not affect the capability of other SSCs to perform necessary safety functions, is acceptable if found in accordance with either of the following: A. The postulated failure or collapse of structures and components not designed for tornado loads, including missiles, can be shown not to result in any structural or other damage to safety-related structures, systems, or components.	Conformance with exceptions. COL Applicant is to verify that site specific tornado wind speed is enveloped by DCD tornado windspeed, and CAT I structures, systems and components are not adversely impacted by site specific SSCs.	3.3.2

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 3 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.3.2 Tornado Loads (continued)	B. Safety-related structures are designed to resist the effects of the postulated structural failure, collapse, or generation of missiles from structures and components not designed for tornado loads.		
3.4.1 Internal Flood Protection for Onsite Equipment Failures	1. Guidance acceptable for meeting the seismic design and classification requirements of GDC 2 is found in Regulatory Guide (RG) 1.29, Position C.1 for safety-related SSCs and Position C.2 for nonsafety-related SSCs. 2. The requirements of GDC 4 are met if SSCs important to safety are designed to accommodate the effects of discharged fluid resulting from high and moderate energy line breaks that are postulated in SRP sections 3.6.1 and 3.6.2.	Conformance with no exceptions identified.	3.4.1
3.4.2 Analysis Procedures	The design of a structure that must withstand the effects of the highest flood and groundwater levels is acceptable if the relevant requirements of GDC 2, “Design Bases for Protection Against Natural Phenomena,” are complied with. The criteria necessary to meet the relevant requirements of GDC 2 are as follows: 1. The highest flood and groundwater levels and the associated static and dynamic effects, if any, used in the design shall be the most severe ones that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated. 2. In most situations, the highest flood level is below the proposed plant grade and only its hydrostatic effects need be considered. Unless the hydrostatic head associated with the highest flood and groundwater levels is relieved by utilizing a drainage or a pumping system around the foundations of structures, hydrostatic pressure has to be considered as a structural load on basement walls and the foundation slab of a structure. In consideration of any uplifting or floating of a structure, the total buoyancy force may be based on the highest flood level or the highest groundwater level excluding wave action. However, wave action should be included in the calculation for lateral and overturning movements of a structure.	Conformance with exceptions. COL Applicant is to address final site specific groundwater and external flood elevations.	3.4.2

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 4 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.4.2 Analysis Procedures (continued)	3. Where the flood level is above the proposed plant grade, the dynamic loads of wave action should be considered. Procedures for determining such dynamic loads are acceptable if they are in accordance with or equivalent to those delineated in the U.S. Army Coastal Engineering Research Center, "Shore Protection Manual" (Vol. I, June 2002, reprinted from 1973 edition and Vol. II, June 2002, reprinted from 1973 edition) or in EM 1110-2-1100, Coastal Engineering Manual, Part II, Chapter 1, "Water Wave Mechanics," U.S. Army Corps of Engineers, April 30, 2002 as applicable.		
3.5.1.1 Internally Generated Missiles (Outside Containment)	The design of the SSCs important to safety is acceptable if the integrated design affords protection from the internally generated missiles (outside containment) that may result from equipment failure, in order to maintain their safety functions in accordance with GDC 4. Acceptance is based on the design meeting the guidance as described in Regulatory Guide (RG) 1.115, as related to the protection of SSCs important to safety from the effects of turbine missiles. 1. The applicant's statistical significance of an identified missile can be evaluated by a probability analysis. Its statistical significance is determined by calculating the probability of missile occurrence. If this probability is less than 10-7 per year, the missile is not considered statistically significant. If the probability of occurrence is greater than 10-7 per year, the probability of impact on a significant target is determined. If the product of these two probabilities is less than 10-7 per year, the missile is not considered statistically significant. If the product is greater than 10-7 per year, the probability of significant damage is determined. If the combined probability (product of all three) is less than 10-7 per year, the missile is not considered statistically significant. If the combined probability is greater than 10-7 per year, missile protection of SSCs important to safety, and of nonsafety-related SSCs whose failure could affect an intended safety function of the safety related SSCs, should be provided by one or more of the six methods listed below.	Conformance with no exceptions identified.	3.5.1.1

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 5 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.5.1.1 Internally Generated Missiles (Outside Containment) (continued)	2. Missile protection for SSCs important to safety is adequate if provided by one or more of the following methods: (1) locating the system or component in a missile-proof structure, (2) separating redundant systems or components for the missile path or range, (3) providing local shields and barriers for systems and components, (4) designing the equipment to withstand the impact of the most damaging missile, (5) providing design features to prevent the generation of missiles, or (6) orienting missile sources to prevent missiles from striking equipment important in safety. RG 1.117 provides guidance on the SSCs that should be protected. Where barriers are used as a method of protection of SSCs from internal missiles, the design of the barriers is acceptable if it meets the guidance of RG 1.115 position C.3. Components within one train of a system with redundant trains need not be protected from missiles originating from the same train.		
3.5.1.2 Internally Generated Missiles (Inside Containment)	The design of the SSCs important to safety is acceptable if the integrated design affords protection from the internally generated missiles (inside containment) that may result from equipment failure, in order to maintain their safety functions in accordance with GDC 4. 1. The applicant's statistical significance of an identified missile can be evaluated by a probability analysis. The statistical significance for a potential missile is determined by calculating the probability of missile occurrence. If this probability is less than 10-7 per year, the missile is not considered significant. If the probability of occurrence is greater than 10-7 per year, the probability that it will impact a significant target is determined. If the product of these two probabilities is less than 10-7 per year, the missile is not considered significant. If the product is greater than 10-7 per year, the probability of significant damage is determined. If . the combined probability (product of all three) is less than 10-7 per year, the missile is not considered significant. If the combined probability is greater than 10-7 per year, missile protection of SSCs important to safety, and of nonsafety-related SSCs whose failure could affect an intended safety function of the safety related SSCs, should be provided by one or more of the six methods listed below.	Conformance with no exceptions identified.	3.5.1.2

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 6 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.5.1.2 Internally Generated Missiles (Inside Containment) (continued)	2. The missile protection for SSCs important to safety is adequate if provided by one or more of the following methods: (1) locating the system or component in a missile-proof structure, (2) separating redundant systems or components for the missile path or range, (3) providing shields and barriers for systems and components, (4) designing the equipment to withstand the impact of the most damaging missile, (5) providing design features to prevent the generation of missiles, or (6) orienting missile sources to prevent missiles from striking equipment important to safety. In summary, a Safety Analyses Report (SAR) statement that SSCs important to safety will be afforded protection by locating them in individual missile-proof structures, physically separating redundant systems or system components, or providing special protective shields or barriers is an acceptable method to meet this criterion		
3.5.1.3 Turbine Missiles	1. Probability Analysis 2. Operating experience on turbine rotor cracking, turbine stop and control valve failures, blade failures, and rotor ruptures resulting in the generation of high-energy missiles 3. Maintaining acceptably low missile generation probability by means of a suitable program of periodic testing and inspection 4. Appropriate orientation of the turbine 5. Turbine in-service inspection program 6. Missile Barriers	Conformance with no exceptions identified.	3.5.1.3
3.5.1.4 Missiles Generated by Tornadoes and Extreme Winds	1. Regulatory Guide (RG) 1.76 describes acceptable design-basis tornado-generated missile spectrum for the design of nuclear power plants. 2. The method of identifying appropriate design-basis missiles generated by natural phenomena shall be consistent with the acceptance criteria defined for the evaluation of potential accidents from external sources in SRP Section 2.2.3. Other methodologies used by licensees and applicants with appropriate rationale may be acceptable on a case-by-case basis.	Conformance with exceptions. COL Applicant is to verify that site specific windspeeds are bounded by DCD windspeeds.	3.5.1.4

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 7 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.5.1.5 Site Proximity Missiles (Except Aircraft)	1. To meet the requirements of 10CFR100 , the probability that site proximity missiles will impact the plant and cause radiological consequences greater than the 10CFR100 exposure guidelines must be less than an order of magnitude of 10-7 per year (see guidance in SRP Section 2.2.3). If the review indicates that the above criterion is not met, then the acceptance criterion described in item 2 below applies. 2. The plant will meet the relevant requirements of GDC 4 and will be considered appropriately protected against site proximity missiles' design if the SSCs important to safety are capable of withstanding the effects of the postulated missiles without loss of safe-shutdown capability and without causing a release of radioactivity in excess of the 10CFRPart 100 dose guidelines.	Conformance with exceptions. COL Applicant may be required to perform additional site specific missile analysis.	3.5.1.5
3.5.1.6 Aircraft Hazards	1. 10CFR100.10, 10CFR100.20, 10CFR100.21, 10CFR52.17 and 10CFR52.79 requirements are met if the probability of aircraft accidents resulting in radiological consequences greater than the 10CFRPart 100 exposure guidelines is less than an order of magnitude of 10-7 per year (see SRP Section 2.2.3). The probability is considered to be less than an order of magnitude of 10-7 per year by inspection if the distances from the plant meet all of the criteria listed below: (additional text follows on methodology) 2. If the above proximity criteria are not met, or if sufficiently hazardous military activities are identified (see item B above), a detailed review of aircraft hazards must be performed. Aircraft accidents that could lead to radiological consequences in excess of the exposure guidelines of 10CFRPart 100 with a probability of occurrence greater than an order of magnitude of 10-7 per year should be considered in the design of the plant. If the results of the review do not support a finding that the risk from aircraft activities is acceptably low, then the design-basis acceptance criteria outlined in GDC 4 applies.	Conformance with exceptions. COL Applicant may be required to perform additional site specific aircraft hazards analysis.	3.5.1.6

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 8 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.5.1.6 Aircraft Hazards (continued)	The plant meets the relevant requirements of GDC 3 and GDC 4, and is considered appropriately protected against design-basis aircraft impacts and fires, if the SSCs important to safety are capable of withstanding the effects of the postulated aircraft impacts and fires without loss of safe-shutdown capability and without causing a release of radioactivity that could exceed the 10CFRPart 100 dose guidelines. Regulatory Guide (RG) 1.117 provides acceptable methods for determining those SSCs that should be protected. The selection of SSCs to be protected is based upon not allowing offsite exposures to exceed an appropriate fraction of the offsite dose guidelines of 10CFRPart 100. Basing the limits upon an appropriate "fraction" ensures protection for those events that are not as severe as the design-basis event but have a higher probability of occurrence. Protecting those SSCs important to safety from the effects of externally generated missiles due to aircraft hazards prevents failure of those systems required for safe shutdown and prevents the release of radioactivity with the potential for causing exposures in excess of the 10CFRPart 100 guidelines. The expected rate of exposure identified in 10CFR50.34(a)(1) dose guideline as it relates to the requirements identified in 10CFR100.20(b) should be about an order of magnitude of 10-6 per year. If it can be shown with rigorous analysis, using realistic assumptions and reasonable arguments that the estimated probability could be lower, then, in accordance with the SRP Section 2.2.3, it is acceptable.		
3.5.2 SSCs to Be Protected From Externally-Generated Missiles	Acceptance is based on the design meeting the guidelines of Regulatory Guide (RG) 1.13 as to the capability of spent fuel pool systems and structures to withstand the effects of externally-generated missiles and to prevent missiles from contacting stored fuel assemblies; RG 1.27 as to the capability of the ultimate heat sink and connecting conduits to withstand the effects of externally-generated missiles; RG 1.115 as to the protection of important safety related SSCs from the effects of turbine missiles; and RG 1.117 as to the protection of important safety-related SSCs from the effects of tornado missiles.	Conformance with exceptions. COL Applicant is to be required to perform analysis of essential service water system.	3.5.2
3.5.3 Barrier Design Procedures	1. Local Damage Prediction A. Concrete B Steel C Composite Sections 2. Overall Damage Prediction	Conformance with no exception identified.	3.5.3

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 9 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.6.1 Plant Design for Protection against Postulated Piping Failures in Fluid Systems outside Containment	The application of GDC 2 to this section is to incorporate environmental effects of full circumferential ruptures of non-seismic moderate energy piping in areas where effects are not already bounded by failures of high energy piping. The application of GDC 4 to this section is that the design of SSCs important to safety will accommodate the effects of the environmental conditions associated with postulated pipe ruptures of high and moderate energy piping. Acceptance is based on conformance to BTP 3-3. 1. High and moderate energy fluid systems are separated from essential systems and components, as described in Appendix B to BTP 3-3. 2. High and moderate energy fluid systems, or portions thereof, are enclosed as described in item B.1.b of BTP 3-3. 3. For cases where neither physical separation nor protective enclosures are considered practical by the applicant, the reviewer will verify the following: A. The reasons for which the applicant judged both physical separation and system enclosure to be impractical as means of protection are consistent with item B.1.c. of BTP 3-3. B. Redundant design features or additional protections (assuming a single active failure in any required system) have been provided such that failure modes and effects analyses for all failure situations ensure the performance of safety features. These analyses are done under the criteria and assumptions of item B.3 of BTP 3-3. 4. Design Features are in accordance with item B.2 of BTP 3-3. 5. The effects of postulated failures on essential equipment and the ability of the plant to be safely shut down are analyzed in accordance with item B.3 of BTP 3-3.	Conformance with no exception identified.	3.6.1 Appendix 3E

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 10 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	1. Postulated Pipe Rupture Locations Inside Containment. Acceptable criteria to define postulated pipe rupture locations and configurations inside containment are specified in Branch Technical position (BTP) 3-4. 2. Postulated Pipe Rupture Locations Outside Containment. Acceptable criteria to define postulated rupture locations and plant layout considerations for protection against postulated pipe ruptures outside containment are specified in BTP 3-4. 3. Methods of Analysis. Detailed acceptance criteria covering pipe-whip dynamic analysis, including determination of the forcing functions of jet thrust and jet impingement, are included in subsection III, "Review Procedures," of this SRP section. The general bases and assumptions of the analysis are given in BTP 3-4, subsection 2.C.	Conformance with no exception identified.	3.6.2 Appendix 3E
3.6.3 Leak-Before-Break Evaluation Procedures	1. Compliance with GDC 4 requires that components important to safety be designed to accommodate the effects of, and be compatible with, environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. Safety-related components should be protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failure or events and conditions outside the nuclear power unit. Meeting the requirements of GDC 4 provides assurance that SSCs important to safety will be protected from the dynamic effects of pipe rupture and capable of performing their intended safety function. 2. LBB analyses should demonstrate that the probability of pipe rupture is extremely low under conditions consistent with the design basis for the piping. A deterministic evaluation of the piping system that demonstrates sufficient margins against failure, including verified design and fabrication and an adequate inservice inspection program, can be assumed to satisfy the extremely low probability criterion.	Conformance with no exception identified.	3.6.3 Appendix 3B

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 11 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.7.1 Seismic Design Parameters	1. Design Ground Motion A. Design Response Spectra (text follows on methodology) ii. Certified Standard Plant Design (CD). For a design certification (DC) application, the postulated seismic design response spectra need to bound the minimum required response spectrum anchored at 0.1g (as specified in Appendix S to 10CFRPart 50). These design response spectra are referred to as the CSDRS when the design is certified by the Commission under 10CFRPart 52. B. Design Time Histories 2. Percentage of Critical Damping Values 3. Supporting Media for Seismic Category I Structures 4. Review Considerations for DC and COL Applications (Note: all information in this section is related to COL applications and none to design certifications [DCs])	Conformance with exceptions. COL Applicant need to demonstrate that the chosen site is bounded by DCD parameters for peak ground acceleration, foundation response spectra and supporting media.	3.7.1, 3.8
3.7.2 Seismic System Analysis	1. Seismic Analysis Methods. The seismic analysis of all seismic Category I SSCs should use either a suitable dynamic analysis method or an equivalent static load analysis method, if justified. The SRP acceptance criteria primarily address linear elastic analysis coupled with allowable stresses near elastic limits of the structures. However, for certain special cases (e.g., evaluation of as-built structures), reliance on limited inelastic/nonlinear behavior when appropriate is acceptable to the staff. Analysis methods incorporating inelastic/nonlinear considerations and the analysis results are reviewed on a case-by-case basis. A. Dynamic Analysis Method B. Equivalent Static Load Method 2. Natural Frequencies and Responses. 3. Procedures Used for Analytical Modeling. A. Designation of Systems Versus Subsystems B. Decoupling Criteria for Subsystems. C. Modeling of Structures. D. Representation of Floor Loads, Live Loads, and Major Equipment in Dynamic Model In E. Special Consideration for Dynamic Modeling of Structures.	Conformance with exceptions. COL Applicant need to consider site-specific subgrade condition (materials, layers, etc.) in the SSI modeling and analysis, and in the evaluating for overturning and sliding effects, and also need to design seismic Category Ⅱ SSC based on the design criteria for seismic Category Ⅰ SSC.	3.7.2, 3.8

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 12 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.7.2 Seismic System Analysis (continued)	4. Soil-Structure Interaction A. Modeling of Structure. B. Modeling of Supporting Soil C. Input Ground Motion. 5. Development of In-Structure Response Spectra. 6. Three Components of Earthquake Motion. 7. Combination of Modal Responses. 8. Interaction of Non-Category I Structures with Category I SSCs 9. Effects of Parameter Variations on Floor Response Spectra. 10. Use of Equivalent Vertical Static Factors 11. Methods Used to Account for Torsional Effects. 12. Comparison of Responses. 13. Analysis Procedure for Damping. 14. Determination of Seismic Overturning Moments and Sliding Forces for Seismic Category I Structures		
3.7.3 Seismic Subsystem Analysis	1. Seismic Analysis Methods. 2. Determination of Number of Earthquake Cycles 3. Procedures Used for Analytical Modeling. The acceptance criteria provided in SRP Section 3.7.2, subsection II.3, are applicable. 4. Basis for Selection of Frequencies. To avoid resonance, the fundamental frequencies of components and equipment should preferably be selected to be less than ½ or more than twice the dominant frequencies of the support structure. Use of equipment frequencies within this range is acceptable if the equipment is adequately designed for the applicable loads. 5. Analysis Procedure for Damping. The acceptance criteria provided in SRP Section 3.7.2, subsection II.13, are applicable. 6. Three Components of Earthquake Motion. The acceptance criteria provided in SRP Section 3.7.2, subsection II.6, are applicable. 7. Combination of Modal Responses. The 8. Interaction of Other Systems with Seismic Category I Systems 9. Multiply-Supported Equipment and Components With Distinct Inputs 10. Use of Equivalent Vertical Static Factors. The acceptance criteria provided in SRP Section 3.7.2, subsection II.10, are applicable.	Conformance with exceptions. COL applicant is required to perform analysis of any site-specific dams.	3.7.3, 3.9,3.12

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 13 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.7.3 Seismic Subsystem Analysis (continued)	11. Torsional Effects of Eccentric Masses. For seismic Category I subsystems, when the torsional effect of an eccentric mass is judged to be significant, the eccentric mass and its eccentricity should be included in the mathematical model. The criteria for judging the significance will be reviewed on a case-by-case basis. 12. Seismic Category I Buried Piping, Conduits, and Tunnels. 13. Methods for Seismic Analysis of Seismic Category I Concrete Dams. For the analysis of all seismic Category I concrete dams, an appropriate approach that takes into consideration the dynamic nature of forces (due to both horizontal and vertical earthquake loadings), the behavior of the dam material under earthquake loadings, soil-structure interaction (SSI) effects, and nonlinear stress-strain relations for the soil, should be used. Analysis of earthen dams is reviewed under SRP Section 2.5.5, “Stability of Slopes.” 14. Methods for Seismic Analysis of Above-Ground Tanks		
3.7.4 Seismic Instrument-ation	The type, locations, operability, characteristics, installation, actuation, remote indication, and maintenance of seismic instrumentation should meet the guidance discussed below. Where an applicant proposes specific details different from these, acceptability should be based upon meeting applicable regulations, as stated above, consistent with current proven technologies and intended use of the recorded information. 1. Comparison with RG 1.12. 2. Comparison with RG 1.166	Conformance with no exceptions identified.	3.7.4
3.8.1 Concrete Containment	1. Description of the Containment. 2. Applicable Codes, Standards, and Specifications. 3. Loads and Loading Combinations. 4. Design and Analysis Procedures. 5. Structural Acceptance Criteria 6. Materials, Quality Control, and Special Construction Techniques 7. Testing and Inservice Surveillance Requirements	Conformance with no exceptions identified.	3.8.1

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 14 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.8.2 Steel Containment	1. Description of the Containment. 2. Applicable Codes, Standards, and Specifications. 3. Loads and Loading Combinations. 4. Design and Analysis Procedures. 5. Structural Acceptance Criteria 6. Materials, Quality Control, and Special Construction Techniques 7. Testing and Inservice Surveillance Requirements	Not applicable. US-APWR does not utilize a steel containment.	N/A
3.8.3 Concrete and Steel Internal Structures of Steel or Concrete Containments	1. Description of the Internal Structures. 2. Applicable Codes, Standards, and Specifications. 3. Loads and Loading Combinations. 4. Design and Analysis Procedures. 5. Structural Acceptance Criteria 6. Materials, Quality Control, and Special Construction Techniques 7. Testing and Inservice Surveillance Requirements	Conformance with no exceptions identified.	3.8.3
3.8.4 Other Seismic Category I Structures	1. Description of the Structures. 2. Applicable Codes, Standards, and Specifications. 3. Loads and Loading Combinations. 4. Design and Analysis Procedures. 5. Structural Acceptance Criteria 6. Materials, Quality Control, and Special Construction Techniques 7. Testing and Inservice Surveillance Requirements 8. Masonry Walls	Conformance with no exceptions identified.	3.8.4
3.8.5 Foundations	1. Description of the Foundation. 2. Applicable Codes, Standards, and Specifications. 3. Loads and Loading Combinations. 4. Design and Analysis Procedures. 5. Structural Acceptance Criteria 6. Materials, Quality Control, and Special Construction Techniques 7. Testing and Inservice Surveillance Requirements	Conformance with no exceptions identified,	3.8.5

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 15 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.9.1 Special Topics for Mechanical Components	1. To meet the requirements of GDCs 1, 2, 14, 15, and 10CFRPart 50, Appendix S, the applicant should provide a complete list of transients to be used in the design and fatigue analysis of all Code Class 1 and core support components, supports, and reactor internals within the reactor coolant pressure boundary. The number of events for each transient and the number of load and stress cycles per event and for events in combination should be included. All transients, such as startup and shutdown operations, power level changes, emergency and recovery conditions (including, for new applications, natural convection cooldown), switching operations (i.e., startup or shutdown of one or more coolant loops), control system or other system malfunctions, component malfunctions, transients from single operator errors, inservice hydrostatic tests, seismic events as determined from the criteria specified in Appendix S to 10CFRPart 50, and design-basis events contained in the Code-required "Design Specifications" for the components of the reactor coolant pressure boundary, should be specified. The section of the applicant's SAR on transients will be acceptable if the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure conditions caused by those transients. To a large extent the selection of these specific transient conditions is based upon engineering judgment and experience. Some guidance on the selection of these transients and combinations can be found in SRP Section 3.9.3. Transients and consequent loads and load combinations with appropriate specified design and service limits should provide a complete basis for design of the reactor coolant pressure boundary for all conditions and events expected over the service lifetime of the plant. The staff should consider the number of transients appropriate for the design life of the plant. Also, environmental conditions to which equipment important to safety will be exposed (e.g., chemistry of the coolant water) should be considered to minimize the degradation of materials due to corrosion.	Conformance with no exceptions identified,	3.9.1,5.1,5.2

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 16 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.9.1 Special Topics for Mechanical Components (continued)	2. To meet the requirements of 10CFRPart 50, Appendix B, and GDC 1, a list of computer programs to be used in dynamic and static analyses to determine the structural and functional integrity of seismic Category I Code and non-Code items and the analyses to determine stresses should be provided. For each program the following information should be provided to demonstrate applicability and validity: (additional text follows on methodology) 3. To meet the requirements of GDCs 1, 14, and 15, if experimental stress analysis methods are used in lieu of analytical methods for any seismic Category I Code or non-Code items, the section of the SAR addressing the experimental stress analysis methods is acceptable if the information meets the provisions of Appendix II to ASME Code, Section III, Division 1 and, as in the case of analytical methods, if the information is sufficiently detailed to show the design meeting the provisions of the Code-required "Design Specifications." 4. To meet the requirements of GDCs 1, 14, and 15 when Service Level D limits are specified by the applicant for Code Class 1 and core support components and for supports, reactor internals, and other non-Code items, the methods of analysis to calculate the stresses and deformations should conform to the methods outlined in Appendix F to ASME Code, Section III, Division 1, subject to the conditions addressed in subsection III.4 of this SRP section.		
3.9.2 Dynamic Testing and Analysis of Systems, Structures, and Components	1. Relevant requirements of GDCs 1, 2, 4, 14, and 15 are met if vibration, thermal expansion, and dynamic effects testing are conducted during startup functional testing for specified high- and moderate-energy piping and their supports and restraints. The purposes of these tests are to confirm that the piping, components, restraints, and supports have been designed to withstand the dynamic loadings and operational transient conditions encountered during service as required by the code and to confirm that no unacceptable restraint of normal thermal motion occurs. (Additional text follows on contents of test program) 2. To meet the requirements of GDC 2, acceptance criteria for the areas of review described in subsection I.2 of this SRP section are given below. Other approaches which can be justified as equivalent to or more conservative than the stated acceptance criteria may be used to confirm the ability of all Seismic Category I systems and components and their supports to function as needed during and after an earthquake. (Additional text follows on methodologies)	Conformance with exceptions. SRP 3.9.2 requires the measurement at startup test for the internals (such as the primary and secondary separator), that may potentially be adversely affected by the flow induced vibration. However, there has been no experience of degradation of the internals due to the excessive vibration. In addition, the internals configuration and operating conditions of 91TT-1 SG are similar to the operating SGs. Therefore, the measurement at startup test is not planed.	3.6.3, 3.7.1, 3.7.2, 3.7.3, 3.9.1, 3.9.3, 3.9.5, 3.10, 4.4, 5.4.2.1.2.10, 14.0, 14.3

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 17 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.9.2 Dynamic Testing and Analysis of Systems, Structures, and Components (continued)	3. To meet the requirements of GDCs 1 and 4, the following guidelines, in addition to RG 1.20 "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing", apply to the analytical solutions to predict vibrations of reactor internals for prototype plants. Generally, this analysis is required only for prototype designs and power uprate of existing plants; However, it is not required for non-prototypes except that segments of an analysis (in particular, assessments of any potential adverse flow effects) may be necessary if there are deviations from the prototype internals design or operating conditions or if the non-prototype is based on a conditional prototype which has experienced problems from adverse flow effects. If the reactor internal structures are a non-prototype design, the applicant should refer to the results of tests and analyses for the prototype reactor and give a brief summary of the results. A more detailed summary of results of assessment of the potential of any adverse flow effects also should be given. (Additional text follows on methodologies) 4. For requirements of GDCs 1 and 4, the preoperational vibration and stress test program for the internals of a prototype reactor, for existing reactors under consideration for power uprate, and for non-prototype reactors whose valid or conditional prototypes have experienced structural failures due to adverse flow effects in any plant (e.g., steam dryer cracking and valve failures) should conform to the requirements for a prototype test as specified in RG 1.20, including vibration prediction, vibration monitoring, adverse flow effects (flow-induced acoustic and structural resonances, data reduction, bias errors and uncertainty analysis, and walkdown and surface inspections. The test program to demonstrate design adequacy of the reactor internals should include, but not necessarily be limited to, the following: (additional text follows on methodologies) 5. For requirements of GDCs 2, 4, 14, and 15 dynamic system analyses should confirm the structural design adequacy of the reactor internals and the reactor coolant piping (unbroken loops) to withstand the dynamic loadings of the most severe LOCA in combination with the SSE. Where a substantial separation between the forcing frequencies of the LOCA (or SSE) loading and the natural frequencies of the internal structures can be demonstrated, the analysis may treat the loadings statically. (Additional text follows on methodologies)		

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 18 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.9.2 Dynamic Testing and Analysis of Systems, Structures, and Components (continued)	6. For requirements of GDC 1, as to the correlation of tests and analyses of reactor internals, the applicant should address the following items to ensure the adequacy and sufficiency of the test and analysis results. (Additional text follows on methodologies) 7. For new applications, test specifications should be in accordance with ASME OM-S/G-1990, "Standards and Guides For Operation of Nuclear Power Plants," Part 3, "Requirements for Preoperational and Initial Start-Up Vibration Testing of Nuclear Power Plant Piping Systems," and Part 7, "Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems."		
3.9.3 ASME Code Class 1, 2, and 3 Components, and Component Supports, and Core Support Structures	1. Loading Combinations, System Operating Transients, and Stress Limits. The design and service loading combinations, including system operating transients, and the associated design and service stress limits considered for each component and its supports should be sufficiently defined to provide the basis for design of Code Class 1, 2, and 3 components and component supports, and core support structures for all conditions. The acceptability of the combination of design and service loadings (including system operating transients), applicable to the design of Class 1, 2, and 3 components and component supports, and core support structures, and of the designation of the appropriate design or service stress limit for each loading combination, is judged by comparison with positions stated in Appendix A, and with appropriate standards acceptable to the staff, developed by professional societies and standards organizations. The design criteria for internal parts of components such as valve discs, seats, and pump shafting should comply with applicable Code or Code Case criteria. In those instances where no Code criteria exist, the design criteria are acceptable if they ensure the structural integrity of the part such that no safety-related functions are impaired. 2. Design and Installation of Pressure Relief Devices. The applicant should use design criteria for pressure relief installations specified in Appendix O, ASME Code, Section III, and Division 1, “Rules for the Design of Safety Valve Installations.” In addition, the following criteria are applicable: (additional text follows on requirements)	Conformance with no exceptions identified, Note: Analysis data will be contained in the COLA.	3.9.3.1,3.9.3.4, Appendix 3C

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 19 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.9.3 ASME Code Class 1, 2, and 3 Components, and Component Supports, and Core Support Structures (continued)	3. Component Supports. The component support designs should provide adequate margins of safety under all combinations of loadings. The combination of loadings (including system operating transients) considered for each component support within a system, including the designation of the appropriate service stress limit for each loading combination should meet the criteria in Appendix A, Regulatory Guides (RG) 1.124 and RG 1.130 and Subsection NF of the Code. (Additional text follows on requirements)		
3.9.4 Control Rod Drive Systems	1. The descriptive information is determined to be sufficient provided the minimum requirements for such information meet Section 3.9.4 of Regulatory Guide (RG) 1.29. 2. Construction (as defined in NCA-1110 of Section III of the ASME Code) should meet the following codes and standards utilized by the nuclear industry which have been reviewed and found acceptable: (additional text follows on requirements) 3. For the various design and service conditions defined in NB-3113 of Section III of the ASME Code, load combination sets are as given in SRP Section 3.9.3. The stress limits applicable to pressurized and nonpressurized portions of the control rod drive systems should be as given in SRP Section 3.9.3 for the response to each loading set. For BWRs, the CRDS design should adequately consider water hammer loads to assure that system safety functions can be achieved. 4. The operability assurance program will be acceptable provided the observed performance as to wear, functioning times, latching, and ability to overcome a stuck rod meets system design requirements.	Conformance with no exceptions identified.	3.9.4
3.9.5 Reactor Pressure Vessel Internals	1. Requirements for loads, loading combinations, and limits applicable to those portions of reactor internals constructed to Subsection NG of the ASME Code are presented in SRP Section 3.9.3. 2. The design and construction of the core support structures should comply with the requirements of Subsection NG, "Core Support Structures," of the ASME Code and SRP Section 3.9.3.	Conformance with exceptions. SRP 3.9.5 requires the measurement at startup test for the internals (such as the primary and secondary separator), that may potentially be adversely affected by the flow induced vibration. However, there has been no experience of degradation of the internals due to the excessive vibration. In addition, the internals	3.9.5, 5.4.2.1.2.10,

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 20 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.9.5 Reactor Pressure Vessel Internals (continued)	3. The design criteria, loading conditions, and analyses that provide the bases for the design of reactor internals other than the core support structures should meet the guidelines of NG-3000 and be constructed not to affect the integrity of the core support structures adversely (NG-1122). If other guidelines (e.g., manufacturer standards or empirical methods based on field experience and testing) are the bases for the stress, deformation, and fatigue criteria, those guidelines should be identified and their use justified. 4. Deformation limits for reactor internals should be established by the applicant and presented in the safety analysis report. The basis for these limits should be included. The stresses of these displacements should not exceed the specified limits. The requirements for dynamic analysis of these components are addressed in SRP Section 3.9.2. 5. The reactor internals should be designed to accommodate asymmetric blowdown loads from postulated pipe ruptures. The applicant's evaluation of such loads should demonstrate that they do not exceed the limits imposed by the applicable codes and standards. Where double-ended guillotine break of reactor coolant piping is postulated, criteria for evaluating loading transients and structural components are specified in NUREG-0609. 6. Potential adverse flow effects of flow-induced vibration (FIV) and acoustic resonances on reactor internals (including the steam dryer in BWRs) should be adequately addressed in accordance with relevant criteria stated in the Appendix to this SR Section. 7. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). 8. COL Action Items and Certification Requirements and Restrictions.	configuration and operating conditions of 91TT-1 SG are similar to the operating SGs. Therefore, the measurement at startup test is not planed.	
3.9.6 Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	1. Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints 2. Inservice Testing Program for Pumps 3. Inservice Testing Program for Valves 4. Inservice Testing Program for Dynamic Restraints 5. Relief Requests and Proposed Alternatives 6. Inspection, Test, Analysis, and Acceptance Criteria (ITAAC) 7. COL Action Items and Certification Requirements and Restrictions 8. Operational Program Description and Implementation	Conformance with exceptions. 1.C,1.G,3.B,3.C.i.(4),3.C.iii.(1).(C),. &(d),3.C. iii ,3.C.. iv .(3).3. C .v.(2)&(3).3 C . vi .(3),(5)&(6),3.C. vii ,3.C viii,4 and 6 are to be provided by COL applicant.	3.9.6

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 21 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.9.7 Risk-Informed Inservice Testing of Pumps and Valves	Fundamental elements of acceptance criteria are a description of changes from an existing ISI program, risk–informed engineering analysis methods, and programs for implementation and monitoring.	Not applicable. Only applies to licensees proposing to use risk-informed methods for establishing in-service testing requirements in favor of traditional methods.	N/A
3.9.8 Risk-Informed Inservice Inspection of Piping	Fundamental elements of acceptance criteria are a description of changes from an existing ISI program, risk–informed engineering analysis methods, and programs for implementation and monitoring.	Not applicable. Only applies to licensees proposing to use risk-informed methods for establishing in-service testing requirements in favor of traditional methods.	N/A
3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	Paragraphs 1 and 2 below define acceptable testing and analysis procedures for confirming the functionality of equipment for the defined load condition. These criteria, when satisfied, will fulfill the requirements of GDC 2 and 4, as discussed above, 10CFRPart 50, Appendix B, Criterion XI, and 10CFRPart 50, Appendix S as they relate to the qualification of equipment. 1. The qualification of electrical equipment and its supports should meet the requirements and recommendations of American National Standards Institute/Institute of Electrical and Electronics Engineers (ANSI/IEEE) Std 344-1987 as endorsed by RG 1.100. (Subsequent revision to RG 1.100 will provide guidance with exceptions for use of Appendix QR-A of ASME QME-1-2007 for seismic qualification of active mechanical equipment and other qualifications of mechanical components, and IEEE Std 344-2004 for seismic qualification for Class 1E equipment.) These documents are generally applicable to all types of equipment and should be used to the extent practicable for the qualification of mechanical equipment. Specifically, conformance to the following criteria should be demonstrated. (Additional text follows on requirements.) 2. Instrumentation described in RG 1.97, including associated mountings, should be tested under appropriate seismic and dynamic loadings as described in the regulatory guide, thereby ensuring that the instruments will continue to monitor plant variables and systems after a seismic event and/or accident.	Conformance with no exceptions identified.	3.7,3.9,3.10

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 22 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment (continued)	3. If the applicant proposes qualification by an experience-based approach, the details of the experience database, including applicable implementation methods and procedures to ensure structural integrity and functionality of the in-scope mechanical and electrical equipment, must meet the functionality of equipment for the defined load condition as presented in paragraphs 1 and 2 above. Supporting documentation for equipment identified in the database should confirm that such equipment remained functional during and after an SSE and the equivalent effect of five postulated occurrences of OBE in combination with other relevant static and dynamic loads. 4. GDC 1 and 10CFRPart 50, Appendix B, Criteria XVII establish requirements for records concerning the qualification of equipment. To satisfy these requirements, complete and auditable records must be available, and the applicant must maintain them, for the life of the plant, at a central location. These files should describe the qualification method used for all equipment in sufficient detail to document the degree of compliance with the criteria of this SRP section. These records should be updated and kept current as equipment is replaced, further tested, or otherwise further qualified. The equipment qualification file should contain a list of all systems, equipment, and the equipment support structures, as defined in the second paragraph of subsection I of this SRP Section. The equipment list should identify which equipment is supplied by the NSSS and which equipment is supplied by the BOP. The equipment qualification file should also include qualification summary data sheets for each piece of equipment (i.e., each mechanical and electrical component of each system) which summarize the component’s qualification. These data sheets should include the following information: (Additional text follows on requirements.)		

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 23 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment (continued)	5. GDC 14 requires, in part, that the RCPB shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage. 10CFR50, Appendix A, GDC 30 further requires, in part, that components which are part of the RCPB shall be designed, fabricated, erected, and tested to the highest quality standards practical. As discussed under acceptance criteria in SRP Section 3.9.6, to satisfy these requirements, the qualification program for valves that are part of the RCPB should include testing or testing and analyses demonstrating that these valves will not experience any leakage, or increase in leakage, as a result of any loading or combination of loadings for which the valves must be qualified. 6. The implementation of the qualification program described above should be documented in the following ways: (Additional text follows on requirements.)		
3.11 Environmental Qualification of Mechanical and Electrical Equipment	1. NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment," Revision 1, July 1981 provides staff positions applicable to existing plants for assessing the compliance of an environmental qualification program with 10CFR50.49. For future plants, Regulatory Guide 1.89 provides the principal guidance for implementing the requirements and criteria of 10CFR50.49 for environmental qualification of electrical equipment that is important to safety and located in a harsh environment. However, certain NUREG-0588 Category I guidance may be used if relevant guidance is not provided in Regulatory Guide 1.89. NUREG-0588 includes two sets of qualification criteria, Category I and Category II. Category I refers to IEEE Std 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations." Category I applies to plants whose CP SERs were dated after July 1, 1974. Category II refers to IEEE Std 323-1971, and is not applicable to any future plants.	Conformance with exceptions. Criterion 13:The effect of chemical exposure will be studied in COL application. Criterion 16: This criteria will be studied in COL application.	3.11

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 24 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.11 Environmental Qualification of Mechanical and Electrical Equipment (continued)	2. IEEE Std 323 contains the principles and criteria that are generic to the environmental qualification process. The following clarification related to the criteria in IEEE Std 323 should be considered. IEEE Std 323 requires that the service environment, including the installed configuration of the equipment, be considered as part of the qualification process. In meeting this requirement, the potential for flooding of electrical equipment that are installed above the flood level, but are subject to water and moisture intrusion, should be considered as part of environmental qualification. Operating experience (e.g., Information Notice 89-63) shows that electrical enclosures that are located above the flood level and are subject to water and moisture intrusion could result in submergence of electrical components inside the enclosures, if the enclosures do not have drainage holes. The reviewer should confirm that equipment in such locations, whose design is such that water accumulation is possible, should have measures to preclude such accumulation (e.g., enclosure drain holes) or the affected equipment should be qualified for the anticipated submergence. 3. RG 1.40, “Qualification Tests of Continuous-Duty Motors Installed inside the Containment of Water-Cooled Nuclear Power Plants,” endorses IEEE Std 334, "IEEE Trial Use Guide for Type Tests of Continuous-Duty Class 1 Motors Installed inside the Containment of Nuclear Power Generating Stations." These documents contain guidance acceptable to the staff for the environmental design and qualification of Class 1E motors, and should be used in conjunction with NUREG-0588 and Regulatory Guide 1.89, as appropriate, for evaluating the environmental design and qualification of Continuous-Duty Class 1E Motors. 4. Regulatory Guide 1.63, “Electrical Penetration Assemblies in Containment Structures for Nuclear Power Plants,” endorses IEEE Std 317, "IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations." These documents contain general guidance that is acceptable to the staff for the environmental design and qualification of electrical penetration assemblies, and should be used in conjunction with NUREG-0588 and Regulatory Guide 1.89, as appropriate, for evaluating the environmental design and qualification of electrical penetration assemblies.		

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 25 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.11 Environmental Qualification of Mechanical and Electrical Equipment (continued)	5. RG 1.73, “Qualification Tests of Electric Valve Operators Installed inside the Containment of Nuclear Power Plants,” endorses IEEE Std 382, "IEEE Trial Use Guide for Type Test of Class 1E Electric Valve Operators for Nuclear Power Generating Stations." These documents contain guidance acceptable to the staff for the environmental design and qualification of Class 1E electric valve operators, and should 3.11-8 Revision 3 - March 2007 be used in conjunction with NUREG-0588 and Regulatory Guide 1.89, as appropriate, for evaluating the environmental design and qualification of Class 1E electric valve operators. 6. RG 1.89, “Environmental Qualification of Certain Electric Equipment Important to Safety in Nuclear Power Plants,” provides guidance for implementing the requirements and criteria of 10CFR50.49 for environmental qualification of electrical equipment that is important to safety and located in a harsh environment. RG 1.89 endorses the provisions of IEEE Std 323 as being acceptable to the staff, and provides supplementary guidance for satisfying the Commission's regulations regarding the environmental qualification of electrical equipment located in a harsh environment. 7. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," provides guidance acceptable to the staff for the environmental qualification of the post-accident monitoring equipment described in Subsection I, Item 1(f), of this SRP section, as well as instruments and controls for the equipment described in Subsection I, Items 1(a) to 1(e), of this SRP section. These criteria, as supplemented by those of RG 1.89, should be used to evaluate the environmental qualification of the I&C equipment.		

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 26 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.11 Environmental Qualification of Mechanical and Electrical Equipment (continued)	8. Draft Regulatory Guide 1.131, “Qualification Tests of Electric Cables and Field Splices for Light-Water-Cooled Nuclear Power Plants,” endorses IEEE Std 383, "Standard for Type Test of Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations." These documents contain guidance acceptable to the staff for the environmental qualification of Class 1E electric cables and field splices, and should be used in conjunction with NUREG-0588 and Regulatory Guide 1.89, as appropriate, for evaluating the environmental qualification of Class 1E electric cables and field splices. Pending issuance of the “Final” version, the Draft version of RG 1.131 may be used as guidance. 9. Regulatory Guide 1.156, “Environmental Qualification of Connection Assemblies for Nuclear Power Plants,” endorses IEEE Std 572, "IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations." These documents contain guidance acceptable to the staff for the environmental qualification of Class 1E connection assemblies, and should be used in conjunction with NUREG-0588 and Regulatory Guide 1.89, as appropriate, for evaluating the environmental qualification of Class 1E connection assemblies. 10. Regulatory Guide 1.158, "Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants," endorses IEEE Std 535, "IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations." These documents contain guidance acceptable to the staff for the environmental qualification of Class 1E lead storage batteries, and should be used in conjunction with NUREG-0588 and Regulatory Guide 1.89, as appropriate, for evaluating the environmental qualification of lead storage batteries.		

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 27 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.11 Environmental Qualification of Mechanical and Electrical Equipment (continued)	11. Regulatory Guide 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," provides guidance acceptable to the staff for determining electromagnetic compatibility for I&C equipment during service. These criteria, as supplemented by those of 3.11-9. Regulatory Guide 1.89, should be used to evaluate the environmental design and qualification of safety-related I&C equipment. New digital systems and new advanced analog systems may require susceptibility testing for electromagnetic interference/radio-frequency interference (EMI/RFI) and power surges, if the environments are significant to the equipment being qualified. The functional descriptions of I&C equipment are provided in SRP Chapter 7. 12. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," provides guidance acceptable to the staff for determining the radiation dose and dose rate for equipment during postulated accident conditions. These criteria, as supplemented by those of Regulatory Guide 1.89, should be used to evaluate the accident source term used in the environmental design and qualification of equipment important to safety. 10CFR50.67, “Accident Source Term,” provides the requirements for licensees to revise the accident source term used in design basis radiological analyses for plants licensed prior to January 10, 1997. Radiation dose and dose rate used to determine the radiation environment for qualification of electrical and mechanical equipment must be based on an NRC staff-approved source term and methodology, as discussed in NUREG-0588 and as supplemented by Section II.B.2 of NUREG-0737, “Clarification of TMI Action Plan Requirements,” and NUREG-0718, “Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License,” or as discussed in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." The radiation environment must be based on the integrated effects of the normally expected radiation environment over the equipment's installed life, plus the effects associated with the most severe design basis event during or following which the equipment is required to remain functional.		

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 28 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.11 Environmental Qualification of Mechanical and Electrical Equipment (continued)	The effects of beta radiation must also be considered in the qualification process. The effects of radiation exposure due to recirculatory fluid must be considered for equipment located outside the containment. The staff's definition of what constitutes a mild radiation environment for electronic components, such as semiconductors or electronic components containing organic material, differs from that for other equipment. The staff's position, as stated in NUREG-1503, “Final SER ABWR, Chapter 3, Design of Structures, Components, Equipment, and Systems,” and NUREG-1793, “Final SER AP1000, Chapter 3, Design of Structures, Components, Equipment, and Systems,” is that a mild radiation environment for electronic equipment is a total integrated dose less than 10 Gy (1E3 rad), and a mild radiation environment for other equipment is less than 100 Gy (1E4 rad). Environmental qualification for electrical equipment located in a “Radiation harsh” environment (i.e., locations where radiation is the only harsh environmental condition) can be accomplished in accordance with 10CFR50.49(f)(4) using analysis of test data (from identical materials) combined with radiation test information (i.e., partial test data), and appropriate consideration of margin and aging effects for nonmetallic components/materials when sufficient documentation is available to preclude the need for a type test. 13. The effects of chemical exposure must be addressed in the environmental qualification process. The concentration of chemicals used for qualification must be equivalent to, or more severe than that resulting from the most limiting mode of plant operation (e.g., containment spray, emergency core cooling system initiation, or recirculation phase). If the chemical composition of the chemical spray can be affected by equipment malfunctions, the most severe chemical environment that results from a single failure in the spray system must be assumed. If only demineralized water spray is used, then the effect of the demineralized water spray must be included in the equipment qualification.		

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 29 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.11 Environmental Qualification of Mechanical and Electrical Equipment (continued)	14. Mechanical components must be designed to be compatible with postulated environmental conditions, including those associated with LOCAs. A process must be established to determine the suitability of materials, parts, and equipment needed for safety-related functions, and to verify that the design of such materials, parts, and equipment is adequate. Also, equipment records must be maintained, and these records must include the results of tests and material analyses used as part of the environmental design and qualification process for each component. For mechanical equipment, the staff concentrates its review on materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms). The reviewer confirms that the applicant has (1) identified safety-related mechanical equipment located in harsh environment areas, including its required operating time; (2) identified nonmetallic subcomponents of such equipment; (3) identified the environmental conditions and process parameters for which this equipment must be qualified; (4) identified nonmetallic material capabilities; and (5) evaluated environmental effects. 15. For electrical and mechanical equipment located in a mild environment, acceptable environmental design can be demonstrated by the "design/purchase" specifications for the equipment. The specifications must contain a description of the functional requirements for a specific environmental zone during normal environmental conditions and anticipated operational occurrences. A well-supported maintenance/surveillance program, in conjunction with a good preventive maintenance program, is sufficient to ensure that equipment that meets the design/purchase specifications is qualified for the designed life. Compliance with 10CFR50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," and associated guidance in RG 1.160 are sufficient to provide reasonable assurance that environmental considerations established during design are reviewed every refueling outage and maintained on a continuing basis to ensure that the qualified design life has not been reduced by thermal, radiation, and/or cyclic degradation resulting from unanticipated operational occurrences or service conditions. Modification to the replacement program and/or replacement of equipment should be based on the review of maintenance/surveillance data.		

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 30 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
3.11 Environmental Qualification of Mechanical and Electrical Equipment (continued)	16. For COL reviews the description of the operational program and proposed implementation milestone(s) for the environmental qualification program are reviewed in accordance with 10CFR50.49. The implementation milestone for the environmental qualification program is to have all qualification requirements met prior to the loading of fuel. Implementation is required by a license condition.		
3.12 ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports	A. Piping Analysis Methods B. Piping Modeling Techniques C. Piping Stress Analysis Criteria D. Piping Support Design	Conformance with no exceptions identified.	3.12,5.4
3.13 Threaded Fasteners - ASME Code Class 1, 2, and 3	1. Design Aspects 2. Preservice and Inservice Inspection Requirements	Conformance with no exceptions identified.	3.13
Branch Technical Position 3-1: Classification of Main Steam Components Other Than the Reactor Coolant Pressure Boundary for BWR Plants	The main steam line components of BWR plants should conform to the criteria listed in items 1 through 5 of the attached Table A-1. BWRs that do not include a main steam isolation valve leakage control system or main steam line shutoff valves and that credit fission product hold-up and retention in main steam piping and/or the condenser to address main steam isolation valve leakage in analyses of accident radiological consequences, should also conform to the criteria specified in item 6 of Table A-1. Figure A-1 illustrates acceptable quality group and seismic classifications for BWR main steam piping and components.	Not applicable. Applies to BWRs only.	N/A
Branch Technical Position 3-2: Classification of Main Steam Components Other Than the Reactor Coolant Pressure Boundary	The main steam and feedwater system components of BWR/6 plants should be classified in accordance with Branch Technical Position (BTP) 3-1, or alternately, in accordance with the attached Table B-1. The classifications indicated are consistent with the guidelines currently specified in RG 1.26 and RG 1.29. As an additional criterion, a suitable interface restraint should be provided at the point of departure from the Class I structure where the interface exists between the safety and nonsafety-related portions of the MSL and MFL.	Not applicable. Applies to BWRs only.	N/A

Table 1.9.2-3 US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment (sheet 31 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 3-3: Protection against Postulated Piping Failures in Fluid Systems outside Containment	Past applications for CP & Operating Licenses (OL) contained plant layouts where safety-related equipment or structures were located near the main steam and feedwater high energy lines on the basis of utilization of the "break exclusion" design basis in these lines. In consideration of the large magnitude of potential energy stored in these (main steam and feed) systems during normal plant operation, BTP 3-3 is intended to give clear guidance on acceptable methods for protecting essential equipment from the effects of postulated failures in these systems. 1. Plant Arrangement 2. Design Features 3. Analyses and Effects of Postulated Piping Failures	Conformance with no exceptions identified.	3.6.1, 3.6.2 Appendix 3E
Branch Technical Position 3-4: Postulated Rupture Locations in Fluid System Piping inside and outside Containment	This position on pipe rupture postulation is intended to comply with the requirements of 10CFR50, Appendix A, General Design Criteria (GDC) 4 for the design of nuclear power plant SSCs (SSCs). It is recognized that pipe rupture is a rare event that may only occur under unanticipated conditions, such as those which might be caused by possible design, construction, or operation errors; unanticipated loads; or unanticipated corrosive environments. The staff's observation of actual piping failures has indicated that they generally occur at high stress and fatigue locations, such as at the terminal ends of a piping system at its connection to the nozzles of a component. The criteria of this position are intended to utilize the available piping design information by postulating pipe ruptures at locations having relatively higher potential for failure, such that an adequate and practical level of protection may be achieved. A. High-Energy Fluid Systems Piping B. Moderate-Energy Fluid System Piping C. Type of Breaks and Leakage Cracks in Fluid System Piping	Conformance with no exceptions identified.	3.6.1, 3.6.2, 3.6.3, 3.12

Table 1.9.2-4 US-APWR Conformance with Standard Review Plan Chapter 4 Reactor (sheet 1 of 9)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
4.2 Fuel System Design	1. Design Bases. The fuel system design bases must reflect the four objectives described in Subsection I, Areas of Review. To satisfy these objectives, acceptance criteria are needed for fuel system damage, fuel rod failure, and fuel coolability. These criteria are discussed in the following paragraphs: (Additional text follows on requirements) 2. Description and Design Drawings. 3. Design Evaluation. 4. Testing, Inspection, and Surveillance Plans.	Conformance with no exceptions identified. It is noted that evaluations for criteria 1.B.iii, and 1.B.iv are discussed in Sec. 4.4, that evaluations for criteria 1.B.v, 1.C.ii and 1.C.iii are discussed in Sec. 15.4 and that evaluations for criteria 1.B.vii, 1. 1.C.i and 1.C.iv are discussed in Sec. 16.5, All criteria 1.A.i-1.A.vii and 1.B.i.	4.4, 15.4, 15.6.5
4.3 Nuclear Design	1. There are no direct or explicit criteria for the power densities and power distributions allowed during (and at the limits of) normal operation, either steady-state or load following. These limits are determined from an integrated consideration of fuel limits (SAR Section 4.2), thermal limits (SAR Section 4.4), scram limits (SAR Chapter 7), and transient and accident analyses (SAR Chapter 15). The design limits for power densities (and thus for peaking factors) during normal operation should be such that acceptable fuel design limits are not exceeded during anticipated transients and that other limits, such as the 1204EC (2200EF) peak cladding temperature allowed for loss-of-coolant accidents (LOCAs), are not exceeded during design-basis accidents. Consideration must also be made to the effect of coolant temperatures and enthalpy on the fuel and cladding temperatures. The limiting power distributions are then determined such that the limits on power densities and peaking factors can be maintained in operation. These limiting power distributions may be maintained (i.e., not exceeded) administratively (i.e., not by automatic scrams), provided a suitable demonstration is made that sufficient, properly translated information and alarms are available from the reactor instrumentation to keep the operator informed. The acceptance criteria in the area of power distribution are that the information presented should satisfactorily demonstrate that: (Additional text follows on requirements)	Conformance with no exceptions identified.	4.3

Table 1.9.2-4 US-APWR Conformance with Standard Review Plan Chapter 4 Reactor (sheet 2 of 9)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
4.3 Nuclear Design (continued)	2. The only directly applicable GDC in the area of reactivity coefficients is GDC 11, which states "...the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity," and is considered to be satisfied in light water reactors (LWRs) by the existence of the Doppler and negative power coefficients. There are no criteria that explicitly establish acceptable ranges of coefficient values or preclude the acceptability of a positive moderator temperature coefficient (MTC) such as may exist in PWRs at beginning of core life. The acceptability of the coefficients in a particular case is determined in the reviews of the analyses in which they are used, e.g., control requirement analyses, stability analyses, and transient and accident analyses. The use of spatial effects such as weighting approximations as appropriate for individual transients are included in the analysis reviews. The judgment to be made under this SRP section is whether the reactivity coefficients have been assigned suitably conservative values by the applicant. The basis for that judgment includes the use to be made of a coefficient, i.e., the analyses in which it is important; the state of the art for calculation of the coefficient; the uncertainty associated with such calculations, experimental checks of the coefficient in operating reactors; and any required checks of the coefficient in the startup program of the reactor under review. 3. Acceptance criteria relative to control rod patterns and reactivity worths include: A. The predicted control rod worths and reactivity insertion rates must be reasonable bounds to values that may occur in the reactor. These values are used in the transient and accident analyses and judgment as to the adequacy of the uncertainty allowances are made in the review of the transient and accident analyses. B. Equipment, operating limits, and procedures necessary to restrict potential rod worths or reactivity insertion rates should be shown to be capable of performing these functions. It is a position of the organization responsible for the review/assessment of nuclear design to require, where feasible, an alarm when any limit or restriction is violated or is about to be violated.		

Table 1.9.2-4 US-APWR Conformance with Standard Review Plan Chapter 4 Reactor (sheet 3 of 9)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
4.3 Nuclear Design (continued)	4. There are no specific criteria that must be met by the analytical methods or data that are used by an applicant or reactor vendor. In general, the analytical methods and database should be representative of the state of the art, and the experiments used to validate the analytical methods should be adequate representations of fuel designs in the reactor and encompass a sufficient range of variables and operating conditions.		
4.4 Thermal and Hydraulic Design	1. SRP Section 4.2 specifies the acceptance criteria for the evaluation of fuel design limits. One criterion provides assurance that there be at least a 95-percent probability at the 95-percent confidence level that the hot fuel rod in the core does not experience a DNB or transition condition during normal operation or AOOs. Uncertainties in the values of process parameters (e.g., reactor power, coolant flow rate, core bypass flow, inlet temperature and pressure, nuclear and engineering hot channel factors), core design parameters, and calculational methods used in the assessment of thermal margin should be treated with at least a 95-percent probability at the 95-percent confidence level. The assessment of thermal margin should also consider the uncertainties in instrumentation. The origin of each uncertainty parameter, such as fabrication uncertainty, computational uncertainty, or measurement uncertainty e.g., reactor power, coolant temperature, flow), should be identified. Each uncertainty parameter should be identified as statistical or deterministic and should clearly describe the methodologies used to combine uncertainties. Core design and operating changes for extended power uprates (EPUs) should be performed in a manner that ensures adequate safety margin. At a minimum, there should be a 95-percent probability at the 95-percent confidence level that a hot fuel rod in the reactor core will not experience a DNB or a transition condition during normal operation or AOOs. Specifically, this safety criterion should be satisfied while accounting for changes in radial and bundle power distribution, including any changes in critical heat flux ratio (CHFR) and CPR. The reviewer should confirm the adequacy of the flow-based average power range monitor flux trip and safety limit minimum critical power ratio at the uprated conditions (Review Standard RS-001).	Conformance with no exceptions identified.	4.4

Table 1.9.2-4 US-APWR Conformance with Standard Review Plan Chapter 4 Reactor (sheet 4 of 9)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
4.4 Thermal and Hydraulic Design (continued)	The reviewer should also ensure that the correlations used in the EPU analysis do not exceed their validation range under uprated normal operation and AOO conditions. The following are two examples of acceptable approaches to meeting this criterion: A. For departure from nucleate boiling ratio (DNBR), CHFR or CPR correlations, there should be a 95-percent probability at the 95-percent confidence level that the hot rod in the core does not experience a DNB or boiling transition condition during normal operation or AOOs. B. The limiting (minimum) value of DNBR, CHFR, or CPR correlations is to be established such that at least 99.9 percent of the fuel rods in the core will not experience a DNB or boiling transition during normal operation or AOOs. Correlations of critical heat flux are continually being revised as a result of additional experimental data, changes in fuel assembly design, and improved calculational techniques involving coolant mixing and the effect of axial power distributions. 2. Problems affecting DNBR or CPR limits, such as fuel densification or rod bowing, are accounted for by an appropriate design penalty which is determined experimentally or analytically. Subchannel hydraulic analysis codes, such as those described in “TEMPThermal Enthalpy Mixing Program,” BAW-10021, Babcock and Wilcox Company, April 1970 and “THINC-IC-An Improved Program for Thermal-Hydraulic Analysis Of Rod Bundle Cores,” WCAP-7956, Westinghouse Electric Corporation, June 1973, should be used to calculate local fluid conditions within fuel assemblies for use in PWR DNB correlations. The acceptability of such codes must be demonstrated by measurements made in large lattice experiments or power reactor cores. The review should include the effects of radial pressure gradients in the core flow distribution. The reviewer should also confirm that calculations of BWR fluid conditions for use in CHF correlations have been made in accordance with the models specified in “Loss of Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors,” NEDO-10329, Appendix C, General Electric Company, April 1971 and “General Electric Company Analytical Model for Loss of Coolant Accident Analysis in Accordance with 10CFRPart 50, Appendix K, “NEDO-20566, General Electric Company, November 1975.		

Table 1.9.2-4 US-APWR Conformance with Standard Review Plan Chapter 4 Reactor (sheet 5 of 9)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
4.4 Thermal and Hydraulic Design (continued)	3. The design should address core oscillations and thermal-hydraulic instabilities as described in SRP Section 15.9. 4. Methods for calculating single-phase and two-phase fluid flow in the reactor vessel and other components should include classical fluid mechanics relationships and appropriate empirical correlations. For components of unusual geometry, such as those listed below, these relationships should be confirmed empirically using representative databases from approved reports: A. Reactor vessel ("Reactor Vessel Model Flow Tests," BAW-10037 (nonproprietary version of BAW-10012), Rev. 2, Babcock and Wilcox Company, September 1968). B. Jet pump ("Design and Performance of General Electric Boiling Water Reactor Jet Pumps," APED-5460, General Electric Company, September 1968). C. Core flow distribution (BAW-10037 and "Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello," NEDO-10299, General Electric Company, January 1971, DRAFT Rev. 2, April 1996). D. Void fraction distribution for BWRs. 5. The proposed technical specifications should ensure that the plant can be safely operated at steady-state conditions under all expected combinations of system parameters. The safety limits and limiting safety settings must be established for each parameter, or combinations of parameters, to satisfy specific acceptance criterion 1, above. 6. Preoperational and initial startup test programs should follow the recommendations of Regulatory Guide 1.68, as it relates to measurements and the confirmation of thermal hydraulic design aspects. 7. The design description and proposed procedures for use of the loose parts monitoring system should be consistent with the requirements of Regulatory Guide 1.133. 8. The thermal-hydraulic design should account for the effects of crud in the CHF calculations in the core or in the pressure drop throughout the RCS. Process monitoring provisions should assure the capability to detect a 3-percent drop in the reactor coolant flow. The flow should be monitored every 24 hours.		

Table 1.9.2-4 US-APWR Conformance with Standard Review Plan Chapter 4 Reactor (sheet 6 of 9)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
4.4 Thermal and Hydraulic Design (continued)	9. Instrumentation provided for an unambiguous indication of ICC, such as primary coolant saturation meters in PWRs, reactor vessel measurement systems, and core exit thermocouples, should meet the design requirements of TMI Action Plan Item II.F.2 of NUREG-0737. Applicants subject to 10CFR50.34(f) should meet the requirements of 10CFR50.34(f)(2)(xviii). Procedures for detection and recovery from conditions of ICC must be consistent with technical guidelines, including applicable EPGs developed pursuant to the TMI action plan, that incorporate response predictions based on appropriate analyses. 10. Thermal-hydraulic stability performance of the core during an ATWS event should not exceed acceptable fuel design limits. SRP Sections 15.8 and 15.9 describe an acceptable method for performing such an analysis for BWR and PWR cores.		
4.5.1 Control Rod Drive Structural Materials	1. Materials Specifications. The properties of the materials selected for the CRDM should be equivalent to those of Section III, Appendix I, Division 1 of the ASME Code or Section II, Parts A, B, C, and D of the ASME Code. Cold-worked austenitic stainless steels should have a 0.2 percent offset yield strength no greater than 620 MPa (90,000 psi), to reduce the probability of stress corrosion cracking in these systems. Regulatory Guide (RG) 1.85 describes the acceptable code cases that may be used with these specifications. 2. Austenitic Stainless Steel Components. Acceptance criteria include criteria described in SRP Section 5.2.3, Subsections II.4.D and E, and the criteria described below. RG 1.44 describes accepted methods for preventing intergranular corrosion of stainless steel components. Furnace-sensitized material should not be allowed, and methods described in this guide should be followed for cleaning and protecting austenitic stainless steels from contamination during handling, storage, testing, and fabrication and for determining the degree of sensitization during welding. The controls for abrasive work on austenitic stainless steel surfaces should be adequate for preventing contamination that promotes stress corrosion cracking. The final surfaces should meet the acceptance standards specified in ASME NQA-1-1994 Edition, "Quality Assurance Requirements for Nuclear Facilities." Tools that contain materials that could contribute to stress-corrosion cracking or that, from previous usage, may be contaminated with such materials should not be used on austenitic stainless steel surfaces.	Conformance with no exceptions identified. Note: Description of the control rod drive mechanism is identified in Subsection 3.9.4. Additional description of fabrication control and welding control of the austenitic stainless steel is shown in Subsection 5.2.3.	4.5.1

Table 1.9.2-4 US-APWR Conformance with Standard Review Plan Chapter 4 Reactor (sheet 7 of 9)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
4.5.1 Control Rod Drive Structural Materials (continued)	3. Other Materials. All materials for use in this system should be selected for their compatibility with the reactor coolant as described in Articles NB-2160 and NB-3120 of the ASME Code. The tempering temperature of martensitic stainless steels and the aging temperature of precipitation-hardening stainless steels should be specified for assurance that these materials will not deteriorate from stress corrosion cracking in service. Acceptable heat treatmenttemperatures include aging at 565E - 595EC (1050E - 1100EF) for Type 17-4 PH and 565EC (1050EF) for Type 410 stainless steel. 4. Cleaning and Cleanliness Control. Onsite cleaning and cleanliness control should be in accordance with ASME NQA-1-1994 edition. The oxygen content of the water in vented tanks need not be controlled. Vented tanks with deionized or demineralized water are normal sources of water for final cleaning or flushing of finished surfaces. Halogenated hydrocarbon cleaning agents should not be used.		
4.5.2 Reactor Internal and Core Support Structure Materials	1. Materials. For core support structures and reactor internals, the permitted material specifications are those given in the ASME Code, Section III, Division 1, Sub-subarticle NG-2120. The properties of these materials are specified in Tables 2A, 2B and 4 of Section II of the Code. Additional permitted materials and their applications are identified in ASME Code Cases approved for use as described in Regulatory Guide 1.84, “Design, Fabrication, and Material Code Case Acceptability, ASME, Section III.” 2. Controls on Welding. Methods and controls for core support structures and reactor internals welds shall be in accordance with ASME Code, Section III, Division 1, Article NG-4000. The examination requirements and acceptance criteria for these welds are specified in Article NG-5000. 3. Nondestructive Examination. Nondestructive examinations shall be in accordance with the requirements of ASME Code, Section III, Division 1, Subarticle NG-2500. The nondestructive examination acceptance criteria shall be in accordance with the requirements of ASME Code, Section III, Division 1, Subarticle NG-5300.	Conformance with no exceptions identified. Note: Description of the reactor internals and the core support structures are identified in Subsection 3.9.5. Additional description of fabrication control and welding control of the austenitic stainless steel is shown in Subsection 5.2.3.	3.9.5, 5.2.3

Table 1.9.2-4 US-APWR Conformance with Standard Review Plan Chapter 4 Reactor (sheet 8 of 9)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
4.5.2 Reactor Internal and Core Support Structure Materials (continued)	4. Austenitic Stainless Steels. The acceptance criteria for this area of review are given in SRP Section 5.2.3, subsections II.2 and II.4.a, b, d, and e. Regulatory Guide 1.44 provides acceptance criteria for preventing inter-granular corrosion of stainless steel components. In conformance with this guide, furnace sensitized material should not be allowed. Methods described in this guide should be followed for cleaning and protecting austenitic stainless steel from contamination during handling, storage, testing, and fabrication, and for determining the degree of sensitization that occurs during welding. 5. Other Materials. All materials used for reactor internals and core support structures must be selected for compatibility with the reactor coolant, as specified in Subsubarticles NG-2160 and NG-3120 of Section III, Division 1 of the ASME Code. The tempering temperature of martensitic stainless steels and the aging temperature of precipitation-hardened stainless steels should be specified to provide assurance that these materials will not deteriorate in service. Acceptable heat treatment temperatures are 565EC -595EC (1050EF - 1100EF) for aging of Type 17-4 PH and 565EC (1050EF) for tempering of Type 410 stainless steel. Other materials shall have similar appropriate heat treat and fabrication controls in accordance with strength and compatibility requirements.		
4.6 Functional Design of Control Rod Drive System	1. To meet the requirements of GDC 4, the CRDS should remain functional and provide reactor shutdown capabilities under adverse environmental conditions and after postulated accidents. 2. To meet the requirements of GDC 23, the CRDS should fail in an acceptable condition, even under adverse conditions, that prevents damage to the fuel cladding and excessive reactivity changes during failure. 3. To meet the requirements of GDC 25, the design of the reactivity control systems should assure that a single malfunction of the CRDS will not result in exceeding acceptable fuel design limits. 4. To meet the requirements of GDC 26, the CRDS should be capable of providing sufficient operational control and reliability during reactivity changes during normal operation and anticipated operational occurrences.	Conformance with no exceptions identified. Note: Description of the control rod drive mechanism is identified in Subsection 3.9.4. Description of the reactor trip and reactor control is identified in Subsection 7.2 and 7.7. CRDM cooling system is described in Subsection 9.4. ECCS is described in Subsection 6.3. CVCS is described in Subsection 9.3.4. Investigation of the capability of CVCS and ECCS is described in Chapter 15.	3.9.4, 7.2, 7.7, 6.3, 9.4, 9.3.4, Chapter 15

Table 1.9.2-4 US-APWR Conformance with Standard Review Plan Chapter 4 Reactor (sheet 9 of 9)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
4.6 Functional Design of Control Rod Drive System (continued)	5. To meet the requirements of GDC 27, the combined capability of CRDS and emergency core cooling system should reliably control the reactivity changes to assure the capability to cool the core under accident conditions. 6. To meet the requirements of GDC 28, the CRDS should be designed to assure that reactivity accidents do not result in damage to the reactor coolant pressure boundary, or result in sufficient damage to the core or support structures so as to significantly impair coolability. 7. CRDS should be designed to ensure an extremely high probability of functioning during anticipated operational occurrences to in conformance is GDC, 29. 8. To meet the requirements of 10CFR50.62(c)(3), BWR plants should have an alternate rod injection system that is diverse and independent from the reactor trip system and should have redundant scram air header exhaust valves.		
Branch Technical Position 4-1, Westinghouse Constant Axial Offset Control (CAOC)	In connection with the staff review of WCAP-8185 (17 × 17), the staff reviewed and accepted a scheme developed by Westinghouse for operating reactors that assures that throughout the core cycle, including during the most limiting power maneuvers the total peaking factor, FQ, will not exceed the value consistent with the LOCA or other limiting accident analysis. This operating scheme, called constant axial offset control (CAOC), involves maintaining the axial flux difference within a narrow tolerance band around a burnup-dependent target in an attempt to minimize the variation of the axial distribution of xenon during plant maneuvers. The CAOC methodology, which is described in WCAP-8385 and WCAP-8403, entails (1) establishing an envelope of allowed power shapes and power densities, (2) devising an operating strategy for c the ycle which maximizes plant flexibility (maneuvering) and minimizes axial power shape changes, (3) demonstrating that this strategy will not result in core conditions that violate the envelope of permissible core power characteristics, and (4) demonstrating that this power distribution control scheme can be effectively supervised with ex-core detectors.	Conformance with exceptions. Not directly applicable. However US-APWR will operate using a CAOC methodology, within the range of peaking factors and target axial offset bands currently accepted by NRC for similar PWRs.	4.3

Table 1.9.2-5 US-APWR Conformance with Standard Review Plan Chapter 5 Reactor Coolant and Connecting Systems (sheet 1 of 14)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
5.2.1.1 Compliance with the Codes and Standards Rule, 10CFR50.55a	To meet the requirements of GDC 1 and 10CFR50.55a, RG 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," which describes an acceptable method for determining quality standards for Quality Group B, C, and D water- and steam-containing components important to safety of water-cooled nuclear power plants, is used.	Conformance with no exceptions identified.	5.2.1.1
5.2.1.2 Applicable Code Cases	To meet the requirements of General Design Criterion 1 and 10CFR50.55a, the following regulatory guides are used: 1. Regulatory Guide 1.84, "Design and Fabrication Code Case Acceptability, ASME Section III, Division 1." This guide lists those Section III, Division 1, ASME Code Cases oriented to design, fabrication, materials, and testing, which are acceptable to the staff for implementation in the licensing of nuclear power plants. 2. Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1." This guide lists those Section XI ASME Code Cases which are acceptable to the staff for use in the inservice inspection of components and their supports, as described in the first paragraph of subsection I, of this SRP. 3. Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code." This guide lists ASME OM Code Cases oriented to operation and maintenance for nuclear power plant components which are acceptable to the staff for implementation in the licensing of nuclear power plants. Code Cases pertaining to ASME Code Section III, Division 2, as well as Code Cases alternatives to Regulatory Guides 1.84, 1.147, or 1.192, or for those not covered in Regulatory Guides 1.84, 1.147, or 1.192 may be acceptable in either of the following cases: 1. If the proposed Code Cases provide an acceptable level of quality and safety; or 2. If compliance with the specified requirements of 10CFR50.55a would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.	Conformance with no exceptions identified.	5.2.1.2

Table 1.9.2-5 US-APWR Conformance with Standard Review Plan Chapter 5 Reactor Coolant and Connecting Systems (sheet 2 of 14)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
5.2.2 Overpressure Protection	1. Material Specifications. 2. Design Requirements for BWRs Operating at Power 3. Design Requirements for PWRs Operating at Power 4. Design Requirements for PWRs Operating at Low Temperature (Startup, Shutdown). 5. Testing and Inspections 6. Technical Specifications 7. TMI Action Plan Requirements. Section II.D.1 of the TMI Action Plan requires an applicant submit a plant specific report regarding relief valve (RV) and safety valve (SV) testing. Section II.D.3 of the TMI Action Plan requires that RVs and SVs be provide with direct valve position indication. Generic Letters No. 82-16 and 83-02 requires sections II.D.1 and II.D.3 be covered by technical specifications while NUREG -0737 section II.K.3.3 specifies reporting for section II.D.1 and II.D.3.	Conformance with no exceptions identified. The overpressure protection system design of US-APWR is spring-loaded safety or relief valves. Some requirements are for active relief devices, so some items are not applicable.	5.2.2
5.2.3 Reactor Coolant Pressure Boundary Materials	1. Material Specifications 2. Compatibility of Materials with the Reactor Coolant 3. Fabrication and Processing of Ferritic Materials 4. Fabrication and Processing of Austenitic Stainless Steel	Conformance with no exceptions identified.	5.2.3
5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing	1. System Boundary Subject to Inspection 2. Accessibility 3. Examination Categories and Methods 4. Inspection Intervals 5. Evaluation of Examination Results 6. System Pressure Tests 7. Code Exemptions 8. Code Cases 9. Augmented ISI to Protect Against Postulated Piping Failures 10. Other Inspection Programs 11. Operational Programs	Conformance with exceptions. Criteria #7, #8 and #11 are identified in the SRP as a COL Applicant responsibility. Criterion #9: Not applicable. Subsection II in SRP 5.2.4 describes the high-energy system piping between containment isolation valves should be received an augmented ISI. Criterion #10: Not applicable. Subsection II in SRP 5.2.4 describes that for PWR plants, the applicant has established a program to detect and correct potential RCPB corrosion caused by boric acid leaks as described in generic Letter 88-05.	5.2.4

Table 1.9.2-5 US-APWR Conformance with Standard Review Plan Chapter 5 Reactor Coolant and Connecting Systems (sheet 3 of 14)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
5.2.5 Reactor Coolant Pressure Boundary Leakage Detection	1. For GDC 2, acceptance is based on the guidelines of RG 1.29, Positions C.1 and C.2. 2. For GDC 30, acceptance is based on meeting the guidelines of RG 1.45.	Conformance with no exceptions identified. System design meets R.G1.29 and R.G1.45 described in SRP Acceptance criteria.	5.2.5
5.3.1 Reactor Vessel Materials	1. Materials 2. Special Processes Used for Manufacture and Fabrication of Components 3. Special Methods for Nondestructive Examination 4. Special Controls and Special Processes Used for Ferritic Steels and Austenitic Stainless Steels 5. Fracture Toughness 6. Material Surveillance 7. Reactor Vessel Fasteners	Conformance with no exceptions identified.	5.3.1
5.3.2 Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	1. Pressure-Temperature Limits 2. Upper-Shelf Energy 3. Pressurized Thermal Shock	Conformance with no exceptions identified.	5.3.2
5.3.3 Reactor Vessel Integrity	1. Design 2. Materials of Construction 3. Fabrication Methods 4. Inspection Requirements 5. Shipment and Installation 6. Operating Conditions 7. Inservice Surveillance 8. Operational Programs	Conformance with no exceptions identified.	5.3.3

Table 1.9.2-5 US-APWR Conformance with Standard Review Plan Chapter 5 Reactor Coolant and Connecting Systems (sheet 4 of 14)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
5.4 Reactor Coolant System Component and Subsystem Design	1. Reactor Coolant Pumps or Circulation Pumps [BWR] 2. Steam Generators 3. Reactor Coolant System Piping and Valves 4. Main Steam Line Flow Restrictions 5. Pressurizer 6. Reactor Core Isolation Cooling System (BWR) / Isolation Condenser System [ESBWR]) 7. Residual Heat Removal System / Passive Residual Heat Removal System [ALWR] /Shutdown Cooling Mode of the Reactor Water Cleanup System [ESBWR] 8. Reactor Water Cleanup System [BWR] / Reactor Water Cleanup/Shutdown Cooling System [ESBWR] 9. Reactor Coolant System Pressure Relief Devices / Reactor Coolant Depressurization Systems 10. Reactor Coolant System Component Supports 11. Pressurizer Relief Discharge System 12. RCS High-Point Vents 13. Main Steam Line, Feedwater, and Auxiliary Feedwater Piping Specific SRP acceptance criteria, acceptable to meet the relevant requirements of the NRC's regulations identified in the SRP sections specified above, are provided in the specific SRP sections for the reviews described in Subsection I of this SRP section.	Conformance with exceptions. criteria #6 and #8 apply only to BWRs.	5.4
5.4.1.1 Pump Flywheel Integrity (PWR)	Regulatory Guide (RG) 1.14 provides positions acceptable to the staff in meeting these requirements to ensure the potential for failures of the flywheels of reactor coolant pump motors in light-water-cooled power reactors is minimized. 1. Materials Selection and Fabrication 2. Fracture Toughness 3. Preservice Inspection 4. Flywheel Design 5. Overspeed Test 6. Inservice Inspection (ISI) 7. Operational Programs	Conformance with no exceptions identified.	5.4.1.1

Table 1.9.2-5 US-APWR Conformance with Standard Review Plan Chapter 5 Reactor Coolant and Connecting Systems (sheet 5 of 14)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
5.4.2.1 Steam Generator Materials	1. Selection, Processing, Testing, and Inspection of Materials 2. Steam Generator Design 3. Fabrication and Processing of Ferritic Materials 4. Fabrication and Processing of Austenitic Stainless Steel (if austenitic stainless steel is used for pressure boundary applications) 5. Compatibility of Materials with the Primary (Reactor) and Secondary Coolant and Cleanliness Control 6. Provisions for Accessing the Secondary Side of the Steam Generator	Conformance with no exceptions identified.	5.4.2.1
5.4.2.2 Steam Generator Program	1. Steam generator tubes are susceptible to degradation. This degradation can occur anywhere along the length of the tube. As a result, each tube is required to be accessible for inspection along its entire length and removable from service if unacceptable flaws are observed. The entire length of each tube must be inspectable using currently available nondestructive examination methods and techniques capable of finding the forms of degradation that may occur during the service life of the steam generators. The design of the steam generators should permit tubes with unacceptable flaws to be removed from service to ensure that tube integrity will be maintained. Tubes with unacceptable flaws should also be capable of being stabilized if it is determined that a plugged tube potentially may sever (as a result of continued degradation) and subsequently affect the integrity of an active tube. (Additional text follows on requirements) 2. A steam generator program is needed to ensure the effective monitoring and management of tube degradation and degradation precursors (so as to ensure steam generator tube integrity). (Additional text follows on requirements)	Conformance with no exceptions identified.	5.4.2.2

Table 1.9.2-5 US-APWR Conformance with Standard Review Plan Chapter 5 Reactor Coolant and Connecting Systems (sheet 6 of 14)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
5.4.2.2 Steam Generator Program (continued)	3. The latest revisions of NUREG-1430, NUREG-1431, and NUREG-1432 provide for the establishment and implementation of a steam generator program to ensure that tube integrity is maintained for the operating interval between tube inspections, consistent with the requirements of GDC 32. The Technical Specifications provide the objectives of the steam generator program, maximum limits on the quantity of primary-to secondary leakage permitted during operation, maximum time interval between inspections, objectives of the techniques used to inspect the tubes, tube repair criteria, and tube repair methods. (Additional text follows on requirements) 4. With respect to the steam generator tube repair criteria, Regulatory Guide (RG) 1.121 describes a methodology acceptable to the NRC staff for determining the repair criteria specified in the Technical Specifications. Specifically, RG 1.121 describes a methodology for determining the minimum acceptable steam generator tube wall thickness. (Additional text follows on requirements) 5. With respect to tube repair methods, the review of these methods ensures that the repair is accessible for inspection and that techniques are available to find the forms of degradation to which the repair may be susceptible. The acceptability of any materials used in the repair is evaluated under SRP Section 5.4.2.1. The review of the acceptability of the mechanical design of the repair is consistent with the design requirements of the ASME Code and the steam generator performance criteria in the Standard Technical Specifications. The repair criteria for the repair method are reviewed under the guidance in RG 1.121.		

Table 1.9.2-5 US-APWR Conformance with Standard Review Plan Chapter 5 Reactor Coolant and Connecting Systems (sheet 7 of 14)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
5.4.2.2 Steam Generator Program (continued)	6. The latest revisions of NUREG-1430, NUREG-1431, and NUREG-1432 address ISI; however, preservice inspections are essential in assessing the nature and significance of indications detected during ISI. As a result, it is important to inspect all tubes before placing the steam generators in service, using techniques that should be used during subsequent inspections (i.e., ISI). Although preservice inspections should use techniques that are expected to be employed during ISI, this expectation should not be construed to inhibit the use of new technology or to imply that the techniques used during the preservice inspection will always remain acceptable (i.e., different techniques may be appropriate based on operating experience). 7. 10CFR50.55a(b)(2)(iii) specifically addresses the inspection of steam generator tubes and states that if the plant Technical Specifications include inspection requirements that differ from those in Article IWB-2000 of Section XI of the ASME Code, the Technical Specifications govern. This requirement is intended to resolve any conflict between the requirements in the ASME Code and the Technical Specifications. If a conflict (i.e., difference) does not exist pertaining to a specific requirement, both the requirements of the ASME Code and the Technical Specifications must be met. In general, the requirements in the ASME Code and the Technical Specifications are complementary. 8. For applicants referencing a certified design, the Standard Technical Specifications associated with the referenced design will specify the guidelines for periodic inspection and testing of the steam generator tube portion of the RCPB. 9. Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestones for the Steam Generator Program are reviewed in accordance with 10CFR55a (g) as it relates to periodic inspection and testing of the steam generator tubes as detailed in Section XI of the ASME Code. The implementation milestone is the establishment and completion of an acceptable steam generator program per Article IWA-2430(b) of Section XI of the ASME Code before placing the plant into commercial service. (Additional text follows on requirements)		

Table 1.9.2-5 US-APWR Conformance with Standard Review Plan Chapter 5 Reactor Coolant and Connecting Systems (sheet 8 of 14)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
5.4.6 Reactor Core Isolation Cooling System (BWR)	This SRP is for boiling water reactors and does not apply to the US-APWR	Not applicable. Applies to BWRs only.	N/A
5.4.7 Residual Heat Removal (RHR) System	1. The system or systems must satisfy the functional, isolation, pressure relief, pump protection, and test requirements specified in Branch Technical Position BTP 5-4. 2. To meet the requirements of GDC 4, design features and operating procedures should be provided to prevent damaging water hammer caused by such mechanisms as voided lines. 3. Interfaces between the RHR system and the RCIC and component or service water systems should be designed so that operation of one does not interfere with, and provides proper support (where required) for, the other. In relation to these and other shared systems (e.g., emergency core cooling and containment heat removal systems), the RHR system must conform to GDC 5. 4. When the RHR system is used to control or mitigate the consequences of an accident, it must meet the design requirements of an engineered safety feature system. This includes meeting the guidelines of Regulatory Guide 1.82 regarding water sources for long term recirculation cooling following a loss-of-coolant accident.	Conformance with no exceptions identified.	5.4.7
5.4.8 Reactor Water Cleanup System (BWR)	This SRP is for boiling water reactors and does not apply to the US-APWR …	Not applicable. Applies to BWRs only.	N/A
5.4.11 Pressurizer Relief Tank	1. Acceptance as it relates to the protection of essential systems from the effects of earthquakes is based on meeting the guidelines in Position C.2 of Regulatory Guide 1.29 regarding the location of the tank in relation to other plant systems (the design of the tank system should be such that the plant safety-related systems would continue to perform their safety functions in the event of a tank failure) and in Position C.3 regarding the extension of seismic Category I boundaries.	Conformance with no exceptions identified. System design meets R.G. 1.29 and GDC 4 described in SRP Acceptance criteria.	5.4.11

Table 1.9.2-5 US-APWR Conformance with Standard Review Plan Chapter 5 Reactor Coolant and Connecting Systems (sheet 9 of 14)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
5.4.11 Pressurizer Relief Tank (continued)	2. The staff uses the following specific criteria to determine whether the requirements of GDC 4 are met: A. The rupture disks have a relief capacity that at least equals the combined capacity of the pressurizer relief and safety valves, with sufficient allowance for rupture disk tolerance. B. The pressurizer relief tank volume and the quantity of water initially stored in the tank should be such that no steam or water will be released to containment under any normal operating conditions or AOOs. It should be assumed that the initial temperature of water inside the tank will be no lower than 49 EC (120 EF). Systems performing similar functions should also be shown to have no release to containment during normal operations and AOOs. C. The design of the pressurizer relief tank and rupture disk should accommodate full vacuum so that the tank will not collapse if the contents are cooled after a discharge of steam without the addition of nitrogen. D. Alarms for high temperature, high pressure, and high and low liquid levels for the pressurizer relief tank have been provided. Systems performing similar functions should also have appropriate instrumentation to inform the operator about the condition of the systems. E. The location of the tank should be such that the rupture discs do not pose a missile threat to safety-related equipment.		
5.4.12 Reactor Coolant System High Point Vents	1. The reactor coolant vent design must ensure that use of these vents during and following an accident does not aggravate the challenge to containment or the course of the accident. 2. Vent capability should be provided on high points of the RCS (including the pressurizer on PWRs and the hot legs on Babcock and Wilcox designs) to vent gases which may inhibit core cooling. For reactors with U-tube steam generators, procedures should be developed to remove sufficient gas from the U-tubes to ensure continued core cooling, since it is impractical to individually vent the thousands of U-tubes. In general, vent paths are not required for local high points at locations where gas accumulation would not be expected to jeopardize core cooling such as a reactor coolant pump valve body.	Conformance with no exceptions identified. Regarding to the criteria 9, Chapter 19 discuses a human-factor analysis.	5.4.12

Table 1.9.2-5 US-APWR Conformance with Standard Review Plan Chapter 5 Reactor Coolant and Connecting Systems (sheet 10 of 14)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
5.4.12 Reactor Coolant System High Point Vents (continued)	3. A single failure of a vent valve, power supply, or control system should not prevent isolation of the vent path. On boiling water reactors, block valves are not required in lines with safety valves used for venting. 4. The design should incorporate sufficient redundancy to minimize the probability of inadvertent actuation. Other methods to reduce the chances of inadvertent actuation, such as removing power or administrative controls, may be considered. 5. Since the RCS vent will be part of the RCPB, all requirements for the RCPB must be met. 6. The size of the vent should be smaller than the size corresponding to the definition of a LOCA (Appendix A to 10CFR50, 10CFR52.47(a) (1) (ii), and 10CFR52.79(b)) to avoid unnecessary challenges to the ECCS, unless the applicant provides justification for a larger size. 7. Vent paths to the containment should discharge into areas that provide good mixing with containment air and are able to withstand steam, water, non-condensable, and mixtures of the above. 8. The vent system should be operable from the control room and provide positive valve position indication. Power should be supplied from emergency buses. 9. It is important that the control room displays and controls for the RCS vents do not increase the potential for operator error. A human-factor analysis should be performed that considers the following: A. The use of this information by an operator during both normal and abnormal plant conditions B. Integration into emergency procedures C. Integration into operator training D. Other alarms during an emergency and need for prioritization of alarms 10. The design should have provisions for testing the operability of the reactor coolant vent system. Testing should be performed in accordance with Subsection IWV of Section XI of the ASME Code for Category B valves.		

Table 1.9.2-5 US-APWR Conformance with Standard Review Plan Chapter 5 Reactor Coolant and Connecting Systems (sheet 11 of 14)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
5.4.12 Reactor Coolant System High Point Vents (continued)	11. The reactor coolant vent system (i.e., vent valves, block valves, position indication devices, cable terminations, and piping) should be seismically and environmentally qualified in accordance with IEEE 344, as supplemented by Regulatory Guide 1.100 and Regulatory Guide 1.92. Environmental qualifications must be in accordance with 10CFR50.49. 12. The reactor coolant vent system should be designed to withstand the dynamic loads that will be encountered during operation from high RCS pressure to the approximate atmospheric pressure at the vent system exhaust. 13. Procedures to effectively operate the vent system must consider when venting is needed and when it is not needed. A variety of initial conditions for which venting may be required should be considered. Operator actions and the necessary instrumentation should be identified. 14. The reactor coolant vent system should meet the quality assurance acceptance criteria provided in SRP Chapter 17.		
5.4.13 Isolation Condenser System (BWR)	This SRP is for boiling water reactors and does not apply to the US-APWR	Not applicable. Applies to BWRs only.	N/A
Branch Technical Position 5-1: Monitoring of Secondary Side Water Chemistry in PWR Steam Generators	1. The applicant’s final safety analysis report (FSAR) should describe the implementation of a secondary water chemistry monitoring and control program in accordance with the supplier’s recommended procedure to inhibit steam generator corrosion and tube degradation. The applicant should address how its program meets industry guidelines (e.g., EPRI’s secondary water chemistry guidelines and Nuclear Energy Institute (NEI) 97-06). In addition, this program should cover all operational modes. Each of the modes should be defined with regard to percent rated thermal power and approximate temperature range.	Conformance with no exceptions identified. Specifications in DCD are almost consistent with EPRI Guideline.	10.3.5

Table 1.9.2-5 US-APWR Conformance with Standard Review Plan Chapter 5 Reactor Coolant and Connecting Systems (sheet 12 of 14)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 5-1: Monitoring of Secondary Side Water Chemistry in PWR Steam Generators (continued)	2. The secondary water chemistry monitoring and control program should identify a sampling schedule for critical parameters during each mode of operation, as well as the acceptance control criteria for these parameters. At a minimum, the program should control pH, cation conductivity, sodium, and dissolved oxygen. However, other parameters merit consideration, such as specific conductivity, chloride, fluoride, suspended solids, silica, total iron, copper, sulfate, lead, ammonia, and residual hydrazine. Additives to each steam generator should be controlled separately. 3. The reviewer will evaluate the secondary water chemistry control and monitoring program of each individual plant. Significant deviations from the industry guidelines should be noted and justified technically. Records should be made of the monitored item values and should be made available for audit and inspection when deemed necessary. 4. Routine changes to the secondary water chemistry control and monitoring program should be reported as part of the biannual FSAR update, as required by 10CFR50.71. Changes shall be evaluated in accordance with the requirements of 10CFR50		
Branch Technical Position 5-2: Overpressure Protection of Pressurized-Water Reactors While Operating at Low Temperatures	1. A system should be designed and installed that will prevent exceeding the applicable technical specifications and Appendix G limits for the RCS while operating at low temperatures. The system should be capable of relieving pressure during all anticipated overpressurization events at a rate sufficient to satisfy the technical specification limits, particularly while the RCS is in a water-solid condition. 2. The low-temperature overpressure protection system should be operable during startup and shutdown conditions below the enable temperature, defined as the water temperature corresponding to a metal temperature of at least RT(NDT) + 50EC (90EF) at the beltline location (1/4t or 3/4t) that is controlling in the Appendix G limit calculations.	Conformance with exceptions. Criteria #3, #9 and #10 are not applicable to the US-APWR because the RHR relief valve is passive, has no interlock, and therefore is not considered for single active component failure.	5.2, 5.3

Table 1.9.2-5 US-APWR Conformance with Standard Review Plan Chapter 5 Reactor Coolant and Connecting Systems (sheet 13 of 14)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 5-2: Overpressure Protection of Pressurized-Water Reactors While Operating at Low Temperatures (continued)	3. The system should be able to perform its function assuming any single active component failure. Analyses using appropriate calculational techniques must demonstrate that the system will provide the required pressure relief capacity assuming the most limiting single active failure. The cause for initiation of the event (e.g., operator error, component malfunction) should not be considered as the single active failure. The analyses should assume the most limiting allowable operating conditions and systems configuration at the time of the postulated cause of the overpressure event. All potential overpressurization events should be considered when establishing the worst-case event. Some events may be prevented by using protective interlocks or by locking out power. These events should be identified individually. If the analysis excludes the events, the controls to prevent these events should be in the plant technical specifications. 4. The design of the system should use Institute of Electrical and Electronics Engineers (IEEE) Standard 603 as guidance. The system may be manually enabled; however, an alarm should be provided to alert the operator to enable the system at the correct plant condition during cooldown. Positive indication should be provided to indicate when the system is enabled. An alarm should activate when the protective action is initiated. The reviewer responsible for instrumentation and controls will assist in reviews of the design criteria and the design for the low-temperature overpressure protection system controls and instrumentation, as described in Subsection I of SRP Section 5.2.2. 5. To ensure operational readiness, the overpressure protection system should be testable. Technical specification surveillance requirements should include the following: A. A test performed to ensure operability of the system (exclusive of relief valves) before each shutdown. B. A test for valve operability, as a minimum, to be conducted as specified in the ASME Code Section XI.		

Table 1.9.2-5 US-APWR Conformance with Standard Review Plan Chapter 5 Reactor Coolant and Connecting Systems (sheet 14 of 14)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 5-2: Overpressure Protection of Pressurized-Water Reactors While Operating at Low Temperatures (continued)	6. The system must meet the requirements of Regulatory Guide 1.26 and Section III of the ASME Code. 7. The design of the overpressure protection system should function during an operating basis earthquake. It should not compromise the design criteria of any other safety grade system with which it would interface, such that the requirements of Regulatory Guide 1.29 are met. 8. The overpressure protection system should not depend on the availability of offsite power to perform its function. The system should be operable from battery-backed power sources, not necessarily Class 1E buses. 9. Overpressure protection systems that take credit for active component(s) to mitigate the consequences of an overpressurization event should include additional analyses considering inadvertent system initiation/actuation or should provide justification that existing analyses bound such an event. 10. If pressure relief is from a low-pressure system not normally connected to the primary system, interlocks that would isolate the low-pressure system from the primary coolant system should not defeat the overpressure protection function (see Branch Technical Position 7-1).		
Branch Technical Position 5-3: Fracture Toughness Requirements	1. Preservice Fracture Toughness Test Requirements. 2. Operating Limitations for Fracture Toughness 3. Inservice Surveillance of Fracture Toughness 4. Pressurized Thermal Shock (PWR only)	Conformance with no exceptions identified.	5.2, 5.3.
Branch Technical Position 5-4: Design Requirements of the Residual Heat Removal System	1. Functional Requirements 2. RHR System Isolation Requirements 3. Pressure Relief Requirements 4. Pump Protection Requirements 5. Test Requirements 6. Operational Procedures 7. Auxiliary Feedwater Supply	Conformance with no exceptions identified.	5.4.7

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 1 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.1.1 Engineered Safety Features Materials	1. Materials and Fabrication 2. Composition and Compatibility of ESF Fluids 3. Component and Systems Cleaning 4. Thermal Insulation	Conformance with no exceptions identified.	6.1
6.1.2 Protective Coating Systems (Paints) - Organic Materials	1. A coating system to be applied inside a containment is acceptable if it meets the regulatory positions of Regulatory Guide 1.54 and the standards of ASTM D5144-00 and ASTM D3911-03.	Conformance with no exceptions identified.	6.1.2
6.2.1 Containment Functional Design	A separate SRP section has been prepared for each of these areas. 1 Pressurized water reactor (PWR) dry containments, including sub-atmospheric containments (SRP Section 6.2.1.1.A). 2. Ice condenser containments (SRP Section 6.2.1.1.B). 3. Mark I, II, III, Boiling Water Reactor (BWR), Advanced Boiling Water Rector (ABWR) and Economic Simplified Boiling Water Reactor (ESBWR) pressure-suppression type containments (SRP Section 6.2.1.1.C). 4. Subcompartment analysis (SRP Section 6.2.1.2). 5. Mass and energy release analysis for postulated loss-of-coolant accidents (SRP Section 6.2.1.3). 6. Mass and energy release analysis for postulated secondary system pipe ruptures (SRP Section 6.2.1.4). 7. Minimum containment pressure analysis for emergency core cooling system (ECCS) performance capability studies (SRP Section 6.2.1.5). Specific SRP acceptance criteria are provided in the referenced SRP Sections. Areas related to the evaluation of the containment functional capability are treated in other SRP sections; e.g., Containment Heat Removal (SRP Section 6.2.2), Containment Isolation System (SRP Section 6.2.4), Combustible Gas Control (SRP Section 6.2.5), and Containment Leakage Testing (SRP Section 6.2.6). In addition, the evaluation of the secondary containment functional design capability is reviewed in SRP Section 6.2.3.	Conformance with no exceptions identified.	6.2.1

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 2 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.2.1.1.A PWR Dry Containments, Including Sub-atmospheric Containments	1. To satisfy the requirements of GDC 16 and 50 regarding sufficient design margin, for plants at the construction permit (CP) stage of review, the containment design pressure should provide at least a 10% margin above the accepted peak calculated containment pressure following a loss-of-coolant accident, or a steam or feedwater line break. For plants at the operating license (OL) stage of review, the peak calculated containment pressure following a loss-of-coolant accident, or a steam or feedwater line break, should be less than the containment design pressure. In general, the peak calculated containment pressure should be approximately the same as at the construction permit or design certification stage of review. However, revised or upgraded analytical models or minor changes in the as-built design of the plant may result in a decrease in the margin. 2. To satisfy the requirements of GDC 38 to rapidly reduce the containment pressure, the containment pressure should be reduced to less than 50% of the peak calculated pressure for the design basis loss-of-coolant accident within 24 hours after the postulated accident. If analysis shows that the calculated containment pressure may not be reduced to 50% of the peak calculated pressure within 24 hours, the organization responsible for SRP Section 15.0.3 should be notified. 3. To satisfy the requirement of GDC 38 to rapidly reduce the containment pressure, the containment pressure for subatmospheric containments should be reduced to below atmospheric pressure within one hour after the postulated accident, and the subatmospheric condition maintained for at least 30 days. 4. To satisfy the requirements of GDC 38 and 50 with respect to the containment heat removal capability and design margin, the loss-of-coolant accident analysis should be based on the assumption of loss of offsite power and the most severe single failure in the emergency power system (e.g., a diesel generator failure), the containment heat removal systems (e.g., a fan, pump, or valve failure), or the core cooling systems (e.g., a pump or valve failure). The selection made should result in the highest calculated containment pressure.	Conformance with no exceptions identified. As for Criterion 9, the structural design pressure of each subcompartment is determined based on the design experience. The design differential pressure is to be confirmed by comparing with the calculated peak pressure in the technical report being prepared.	6.2.1, 19.2.3.3.2

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 3 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.2.1.1.A PWR Dry Containments, Including Sub-atmospheric Containments (continued)	5. To satisfy the requirements of GDC 38 and 50 with respect to the containment heat removal capability and design margin, the containment response analysis for postulated secondary system pipe ruptures should be based on the most severe single active failure in the containment heat removal systems (e.g., a fan, pump, or valve failure) or the secondary system isolation provisions (e.g., main steam isolation valve failure or feedwater line isolation valve failure). The analysis should also be based on a spectrum of pipe break sizes and reactor power levels. The accident conditions selected should result in the highest calculated containment pressure or temperature depending on the environmental response to main steam line break accidents are found in NUREG-0588. 6. To satisfy the requirements of GDC 38 and 50 with respect to the functional capability of the containment heat removal systems and containment structure under loss-of-coolant accident conditions, provisions should be made to protect the containment structure against possible damage from external pressure conditions that may result, for example, from inadvertent operation of containment heat removal systems. The provisions made should include conservative structural design to assure that the containment structure is capable of withstanding the maximum expected external pressure; or interlocks in the plant protection system and administrative controls to preclude inadvertent operation of the systems. If the containment is designed to withstand the maximum expected external pressure, the external design pressure of the containment should provide an adequate margin above the maximum expected external pressure to account for uncertainties in the analysis of the postulated event.		

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 4 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.2.1.1.A PWR Dry Containments, Including Sub-atmospheric Containments (continued)	7. In accordance with the requirements of GDC 13 and 64, and 10CFR50.34(f) (2) (xvii) (for those applicants subject to 10CFR50.34(f)), instrumentation capable of operating in the post-accident environment should be provided to monitor the containment atmosphere pressure and temperature and the sump water level and temperature following an accident. The instrumentation should have adequate range, accuracy, and response to assure that the above parameters can be tracked and recorded throughout the course of an accident. See Item II.F.1 of NUREG-0737 and NUREG-0718, and Branch Technical Position 7-10, “Guidance on Application of Regulatory Guide 1.97.” 8. In accordance with 10CFR50.46 Appendix K, I.D.2, the minimum calculated containment pressure should not be less than that used in the analysis of the emergency core cooling system capability (See SRP Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies"). 9. In accordance with GDC 4, containment internal structures and system components (e.g., reactor vessel, pressurizer, steam generators) and supports should be designed to withstand the differential pressure loadings that may be imposed as a result of pipe breaks within the containment subcompartments (See SRP Section 6.2.1.2, "Subcompartment Analysis"). 10. In meeting the requirements of 10CFR50.34(f)(3)(v)(A)(1), applicants subject to this section should evaluate an accident that releases hydrogen generated from a 100% fuel clad metal-water reaction. The evaluation should demonstrate that the appropriate article for service level C limits (considering pressure and dead load only), for either concrete or steel containments, from ASME Boiler Pressure Vessel Code, Section III, are met. In addition to the containment pressurization caused directly by this accident, the increase in pressure from hydrogen burning in containment should be analyzed.		

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 5 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.2.1.1.B Ice Condenser Containments	This SRP is for Ice Condenser Containments and does not apply to the US-APWR.	Not applicable.	N/A
6.2.1.1.C Pressure-Suppression Type BWR Containments	This SRP is for Pressure-Suppression Boiling Water Reactor Containments and does not apply to the US-APWR	Not applicable. This section is applied for BWR only.	N/A
6.2.1.2 Sub-compartment Analysis	1. Nodalization Schemes 2. Initial Thermodynamic Conditions 3. Vent Flow Path and Distribution of Mass and Energy Released 4. Design Pressure	Conformance with no exceptions identified. A technical report with regard to the containment subcompartment analysis conforming to this SRP is being prepared.	6.2.1.2.3
6.2.1.3 Mass and Energy Release Analysis for Postulated LOCAs	1. General Design Criterion 50 and Appendix K to 10CFRPart 50 (Additional text follows on requirements) 2. 10CFR52.47(b) (1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations. 3. 10CFR52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations. 10CFR52.47(a)(1)(vi) provides the requirement for ITAAC for design certification reviews.	Conformance with no exceptions identified.	6.2.1.3,14.3.4.4

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 6 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	1. Sources of Energy 2. Mass and Energy Release Rate 3. Single-Failure Analyses	Conformance with no exceptions identified.	6.2.1.4
6.2.1.5 Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	1. To meet the requirements of 10CFR50.46(a)(1)(i), the model to determine minimum containment pressure for ECCS studies should comply with Regulatory Guide (RG) 1.157, Position C.3.12.1, which describes acceptable containment pressure models for ECCS performance analysis. 2. To meet the requirements of 10CFRPart 50.46(a)(1)(ii), the following specific criteria indicate the conservatism that analyses of the containment response to LOCAs should have for determining the minimum containment pressure for ECCS performance capability studies: A. Calculations of the mass and energy released during postulated LOCAs should be based on the requirements of 10CFRPart 50, Appendix K. B. Branch Technical Position 6-2, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation," delineates the calculation approach that should be followed for a conservative prediction of the minimum containment pressure.	Conformance with no exceptions identified.	6.2.1.5
6.2.2 Containment Heat Removal Systems	1. The containment heat removal systems should meet the redundancy and power source requirements for an engineered safety feature (i.e., the results of failure modes and effects analyses of each system should ensure that the system is capable of withstanding a single failure without loss of function). This conforms to the requirements of GDC 38.	Conformance with exceptions. Criteria 4 is not applied to US-APWR, because the US-APWR does not have the fan cooler system for containment heat removal following the design base accident.	6.2.2

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 7 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.2.2 Containment Heat Removal Systems (continued)	2. With regard to GDC 38 as it relates to the capability of the containment system to accomplish its safety function, the spray system should be designed to accomplish this without pump damage caused by cavitation. A supporting analysis should be presented in sufficient detail to permit the staff to determine the adequacy of the analysis. This analysis should also demonstrate that the available NPSH is greater than or equal to the required NPSH. Regulatory Guide 1.82, Revision 3 (Ref. 1), describes methods acceptable to the staff for evaluating the NPSH margin. If containment accident pressure is credited in determining available NPSH, an evaluation of the contribution to plant risk from inadequate containment pressure should be made. One acceptable way of making this evaluation is to address the five key principles of risk-informed decision making stated in Section 2 of Regulatory Guide 1.174 (Ref. 2). 3. In evaluating the performance capability of the CSS to satisfy GDC 38, the analyses of its heat removal capability should be based on the following considerations: A. The locations of the spray headers relative to the internal structures. B. The arrangement of the spray nozzles on the spray headers and the expected spray pattern. The spray systems should be designed to ensure that the spray header and nozzle arrangements produce spray patterns which maximize the containment volume covered and minimize the overlapping of the sprays. C. The spray drop size spectrum and mean drop size emitted from each type of nozzle as a function of differential pressure across the nozzle. D. The effect of drop residence time and drop size on the heat removal effectiveness of the spray droplets.		

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 8 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.2.2 Containment Heat Removal Systems (continued)	4. In evaluating the performance capability of the fan cooler system to satisfy GDC 38, the design heat removal capability (i.e., heat removal rate versus containment temperature) of the fan coolers should be established on the basis of qualification tests on production units or acceptable analyses that take into account the expected post accident environmental conditions and variations in major operating parameters, such as the containment atmosphere steam-air ratio, condensation on finned surfaces, and cooling water temperature and flow rate. The equipment housing and ducting associated with the fan cooler system should be analyzed to determine that the design is adequate to withstand the effects of containment pressure following a LOCA. Fan cooler system designs that contain components that do not have a post accident safety function should be designed so that failure of nonsafety-related equipment will not prevent the fan cooler system from accomplishing its safety function. 5. In evaluating the heat removal capability of the containment heat removal system to satisfy GDC 38, the potential for surface fouling of the secondary sides of fan cooler, recirculation, and RHR heat exchangers by the cooling water over the life of the plant and the effect of surface fouling on the heat removal capacity of the heat exchangers. The application should discuss the results of the analysis. The results will be acceptable if they demonstrate that provisions such as closed cooling water systems are provided 6.2.2-5 Revision 5 - March 2007 to prevent surface fouling or that surface fouling has been taken into account in the establishment of the heat removal capability of the heat exchangers.		

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 9 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.2.2 Containment Heat Removal Systems (continued)	6. To satisfy the requirements of GDC 38 and 10CFR50.46(b)(5) regarding the long-term spray system(s) and ECCS(s), the containment emergency sump(s) in PWRs and suppression pools in BWRs should be designed to provide a reliable, long-term water source for ECCS and CSS pumps. The containment design should allow for the drainage of spray and emergency core cooling water to the emergency sump(s) or suppression pool and for recirculation of this water through the containment sprays and ECCSs. The design of the sumps or suppression pools and the protective strainer assemblies is a critical element in ensuring long-term recirculation cooling capability. Therefore, adequate design consideration of (1) sump and suppression pool hydraulic performance, (2) evaluation of potential debris generation and associated effects including debris screen blockage, (3) RHR and CSS pump performance under postulated post-LOCA conditions, and (4) impacts of debris penetrating strainers on long-term coolability of the core is necessary. Regulatory Guide 1.82, Revision 3, as modified and supplemented for PWRs by the Nuclear Energy Institute (NEI) Guidance Report (GR) (Ref. 3) and the NRC safety evaluation (SE) (Ref. 4), provide guidance for PWR debris evaluations. Regulatory Guide 1.82, Revision 3, as supplemented by the NRC-approved Boiling Water Reactor Owners' Group (BWROG) Utility Resolution Guidance (URG) (Ref. 5), provide guidance for BWR debris evaluations. 7. In meeting the requirements of GDC 39 and 40 regarding inspection and testing, the design of the containment heat removal systems should provide for periodic inspection and operability testing of the systems and system components such as pumps, valves, duct pressure-relieving devices, and spray nozzles. 8. To satisfy the system design requirements of GDC 38, instrumentation should be provided to monitor the performance of the containment heat removal system and its components under normal and accident conditions. The instrumentation should determine whether a system is performing its intended function or whether a system train or component is malfunctioning and should be isolated.		

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 10 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.2.3 Secondary Containment Functional Design	This SRP is for boiling water reactor designs that feature secondary containments and does not apply to the US-APWR.	Not applicable. This is a BWR requirement; US-APWR does not have a secondary containment.	N/A
6.2.4 Containment Isolation System	1. Regulatory Guide (RG) 1.11 describes acceptable containment isolation provisions for instrument lines. In addition, instrument lines closed both inside and outside containment are designed to withstand pressure and temperature conditions following a loss-of-coolant accident (LOCA) and dynamic effects are acceptable without isolation valves. 2. Containment isolation provisions for lines in engineered safety feature or engineered safety feature-related systems may include remote-manual valves, but should detect possible leakage from these lines outside containment. 3. Containment isolation provisions for lines in systems needed for safe shutdown of the plant (e.g., liquid poison system, reactor core isolation cooling system, and isolation condenser system) may include remote-manual valves, but there should be provisions for detecting leakage from such lines outside containment. 4. Containment isolation provisions for lines in the systems of items 2 and 3 normally consist of one isolation valve inside and one outside containment. If it is not practical to locate a valve inside containment (for example, the valve may be under water as a result of an accident), both valves may be located outside containment. For this type of isolation valve arrangement, the valve nearer the containment and the piping between the containment and the valve should be enclosed in a leak-tight or controlled-leakage housing. If, in lieu of housing, the piping and valve are designed to preclude a breach of piping integrity, the design should comply with SRP Section 3.6.2 requirements. Design of the valve or the piping compartment should provide the capability to detect and terminate leakage from the valve shaft or bonnet seals.	Conformance with exceptions. Criterion 4 is not applied to US-APWR ,because there is no configuration that both isolation valves are outside containment.	6.2.4

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 11 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.2.4 Containment Isolation System (continued)	5. Containment isolation provisions for lines in engineered safety feature or engineered safety feature-related systems normally consist of two isolation valves in series. A single isolation valve is acceptable if system reliability can be shown to be greater, the system is closed outside containment, and a single active failure can be accommodated with only one isolation valve in the line. The closed system outside containment should be protected from missiles, designed to seismic Category I and Group B quality standards, and have a design temperature and pressure rating at least equal to that for the containment. The closed system outside containment should be leak-tested unless system integrity can be shown to be maintained during normal plant operations. For this type of isolation valve arrangement the valve is located outside containment, and the piping between the containment and the valve should be enclosed in leak-tight or controlled-leakage housing. If, in lieu of housing, piping and valve are designed conservatively to preclude a breach of piping integrity, the design should comply with SRP Section 3.6.2 requirements. Design of the valve or the piping compartment should provide the capability to detect and terminate leakage from the valve shaft or bonnet seals. 6. Sealed-closed barriers may be used in place of automatic isolation valves. Sealed-closed barriers include blind flanges and sealed-closed isolation valves which may be closed manual valves, closed remote-manual valves, or closed automatic valves which remain closed after a LOCA. Sealed-closed isolation valves should be under administrative control so they cannot be opened inadvertently. Administrative control includes mechanical devices to seal or lock the valve closed or to prevent power supply to the valve operator. 7. Relief valves may be used as isolation valves provided the relief setpoint is greater than 1.5 times the containment design pressure.		

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 12 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.2.4 Containment Isolation System (continued)	8. 10CFR50.34(f)(2)(xiv) requires that systems penetrating the containment be classified as either essential or nonessential. Reference 26 presents guidance on the classification of systems as essential and nonessential. Essential systems, like those described in items 2 and 3, may include remote-manual containment isolation valves, but there should be provisions for detecting leakage from the lines outside containment. 10CFR50.34(f)(2)(xiv) also requires that nonessential systems be isolated automatically by the containment isolation signal. 9. Isolation valves outside containment should be located as close to it as practical, as required by GDCs 55, 56, and 57. 10. To meet the requirements of GDCs 55 and 56, upon loss of actuating power, automatic isolation valves should take the position of greatest safety. The position of an isolation valve for normal and shutdown plant operating and post-accident conditions depends on the fluid system function. If a fluid system has no post-accident function, the isolation valves in the lines should be closed automatically. For engineered safety feature or engineered safety feature-related systems, isolation valves in the lines may remain open or be opened. In a power failure to the valve operator isolation valves should be in the “safe” position, normally the post-accident valve position. For lines equipped with motor-operated valves, a loss of actuating power leaves the affected valve in the “as-is” position, which may be the open position; however, redundant isolation barriers ensure that the isolation function for the line is satisfied. All power-operated isolation valves should have position indications in the main control room. 11. To improve the reliability of the isolation function, addressed in GDC 54, 10CFR50.34(f)(2)(xiv) requires reduction of the containment setpoint pressure that initiates containment isolation for nonessential penetrations to the minimum value compatible with normal operating conditions. 12. There should be diversity in the parameters sensed for the initiation of containment isolation to satisfy the GDC 54 requirement for reliable isolation capability.		

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 13 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.2.4 Containment Isolation System (continued)	13. To improve the reliability of the isolation function, addressed in GDC 56, system lines which provide open paths from the containment to the environs (e.g., purge and vent lines addressed in 10CFR50.34(f)(2)(xiv)) should be equipped with radiation monitors capable of isolating these lines upon a high-radiation signal, which should not be considered a diverse containment isolation parameter. 14. In meeting GDC 54 requirements, the performance capability of the isolation function should reflect the safety importance of isolating system lines. Consequently, containment isolation valve closure times should be selected for rapid isolation of the containment following postulated accidents. Valve closure time for a power-operated valve to be in the fully-closed position after the actuator power has reached the operator assembly does not include the time to reach actuation signal setpoints or instrument delay times, which, with system design capabilities, should be considered for establishing valve closure times. For lines providing open paths from the containment to the environs (e.g., the containment purge and vent lines), isolation valve closure times of five seconds or less may be necessary. The closure times of these valves should be established to minimize the release of containment atmosphere to the environs, to mitigate the offsite radiological consequences, and to prevent degradation of emergency core cooling system effectiveness by reduced containment back-pressure. Analyses of the radiological consequences and the effect on the containment back-pressure of the release of containment atmosphere should justify the selected valve closure time. Branch Technical Position (BTP) 6-4 presents additional guidance on the design and use of containment purge systems which may be used during the normal plant operating modes (i.e., startup, power operation, hot standby, and hot shutdown). Containment purge valves that do not satisfy the operability criteria of Branch Technical Position 6-4 must be sealed closed as defined in subsection II.6 of this SRP section during operational conditions 1, 2, 3, and 4. Furthermore, closure of these valves must be verified at least every 31 days. These requirements should be incorporated into the technical specifications for plant operation.		

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 14 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.2.4 Containment Isolation System (continued)	15. The use of a closed system inside containment as one of the isolation barriers is acceptable if the closed system design satisfies the following requirements: A. The system does not connect with either the reactor coolant system or the containment atmosphere. B. The system is protected against missiles and pipe whip. C. The system is designated seismic Category I. D. The system is classified Quality Group B. E. The system is designed to withstand temperatures equal to at least that of the containment design. F. The system is designed to withstand the external pressure from the containment structure acceptance test. G. The system is designed to withstand the LOCA transient and environment. As to the structural design of containment internal structures and piping systems, the protection against loss of function from missiles, pipe whip, and earthquakes is acceptable if 1) isolation barriers are located behind missile barriers; 2) pipe whip was considered in the design of pipe restraints and the location of piping penetrating the containment; and 3) the isolation barriers, including the piping between isolation valves, are designated seismic Category I, i.e., designed to withstand the effects of the safe-shutdown earthquake, as recommended by Regulatory Guide 1.29. 16. To meet the requirements of GDCs 1, 2, 4, and 54, appropriate reliability and performance considerations should be included in the design of isolation barriers to reflect the safety importance of their integrity (i.e., containment capability) under accident conditions. The design criteria for components performing a containment isolation function, including the isolation barriers and the piping between them or the piping between the containment and the outermost isolation barrier, are acceptable if: A. Group B quality standards, as defined in RG 1.26, apply to the components, unless the service function dictates that Group A quality standards apply. B. The components are designated seismic Category I in accordance with RG 1.29.		

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 15 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.2.4 Containment Isolation System (continued)	17. GDC 54 requires reliable isolation capability; therefore, for remote-manual isolation valves, the design of the containment isolation system is acceptable if there are provisions to allow the operator in the main control room to know when to isolate fluid systems equipped with remote-manual isolation valves. Such provisions may include instruments to measure flow rate, sump water level, temperature, pressure, and radiation level. 18. GDC 54 specifies requirements for the containment isolation system; therefore, to satisfy GDC 54, the design of the containment isolation system should provide for operability testing of the containment isolation valves and leakage rate testing of the isolation barriers. The isolation valve testing program should be consistent with that proposed for other engineered safety features. SRP Section 6.2.6 presents acceptance criteria for the leakage rate testing program for containment isolation barriers. 19. GDC 54 requires reliable isolation capability. To satisfy this requirement, the design of the containment isolation system should reduce the possibility of unintended isolation valve reopening following isolation. 10CFR50.34(f)(2)(xiv) requires control systems for automatic containment isolation valves be designed for resetting the isolation signal without automatically reopening the valves. Reopening of containment isolation valves should require deliberate operator action and combined reopening of containment isolation valves is not acceptable. Reopening of isolation valves must be valve by valve or line by line, provided that electrical independence and other single-failure criteria remain satisfied. Administrative provisions to close all isolation valves manually before resetting the isolation signals is not an acceptable method for meeting this design requirement. 20. In meeting 10CFR50.34(f)(2)(xv) purging requirements, the regulatory guidance of BTP 6-4, “Containment Purging During Normal Plant Operations,” should be used to establish compliance with this regulation. 21. RG 1.155, “Station Blackout,” Regulatory Position C.3.2.7, provides guidance for meeting the requirements of the SBO rule, 10CFR50.63(a)(2), for containment isolation valves and valve position indication.		

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 16 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.2.4 Containment Isolation System (continued)	22. 10CFRPart 50, Appendix K, provides guidance for the determination of the extent of fuel failure (source term) in the radiological calculations.		
6.2.5 Combustible Gas Control in Containment	1. In meeting the requirements of 10CFRPart 50, § 50.44, and GDC 41 to provide systems to control the concentration of hydrogen in the containment atmosphere, materials within the containment that would yield hydrogen gas due to corrosion from the emergency cooling or containment spray solutions should be identified, and their use should be limited as much as practicable. 2. In meeting the requirements of 10CFRPart 50, § 50.44, and GDC 41 to provide systems to control the concentration of hydrogen or oxygen in the containment atmosphere, the applicant should demonstrate by analysis, for non-inerted containments, that the design can safely accommodate hydrogen generated by an equivalent of a 100 percent fuel clad-coolant reaction, while limiting containment hydrogen concentration, with the hydrogen uniformly distributed, to less than 10 percent (by volume), and while maintaining containment structural integrity. 3. In meeting the requirements of 10CFRPart 50, § 50.44I(3), regarding equipment survivability, equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment structural integrity should 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 percent fuel clad-coolant reaction including the environmental conditions created by activation of the combustible gas control system.	Conformance with no exceptions identified.	6.2.5

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 17 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.2.5 Combustible Gas Control in Containment (continued)	4. In meeting the requirements of 10CFRPart 50, § 50.44, to provide the capability for ensuring a mixed atmosphere in the containment during design bases and significant beyond-design-bases accidents, and of GDC 41 to provide systems as necessary to ensure that containment integrity is maintained, this capability may be provided by an active, passive, or combination system. Active systems may consist of a fan, a fan cooler, or containment spray. For passive or combination systems that use convective mixing to mix the combustible gases, the containment internal structures should have design features which promote the free circulation of the atmosphere. For all containment types, an analysis of the effectiveness of the method used for providing a mixed atmosphere should be provided. This analysis is acceptable if it shows that combustible gases will not accumulate within a compartment or cubicle to form a combustible or detonable mixture that could cause loss of containment integrity. Atmosphere mixing systems prevent local accumulation of combustible or detonable gases which could threaten containment integrity or equipment operating in a local compartment. Active systems installed to mitigate this threat should be reliable, redundant, single-failure proof, able to be tested and inspected, and remain operable with a loss of onsite or offsite power. 5. In meeting the requirements of 10CFRPart 50, § 50.44, and GDC 41 regarding the functional capability of the combustible gas control systems to ensure that containment integrity is maintained, the design should meet the provisions of RG 1.7, Revision 3, Section C.1.		

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 18 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.2.5 Combustible Gas Control in Containment (continued)	6. To satisfy the design requirements of GDC 41: A. Performance tests should be performed on system components, such as hydrogen igniters and combustible gas monitors. The tests should support the analyses of the functional capability of the equipment. B. Combustible gas control system designs should include instrumentation needed to monitor system or component performance under normal and accident conditions. The instrumentation should be capable of determining that a system is performing its intended function, or that a system train or component is malfunctioning and should be isolated. The instrumentation should have readout and alarm capability in the control room. The containment hydrogen and oxygen monitors should meet the provisions of RG 1.7, Revision 3, Section C.2. 7. To satisfy the inspection and test requirements of GDC 41, 42, and 43, combustible gas control systems should be designed with provisions for periodic inservice inspection, operability testing, and leak rate testing of the systems or components. 8. In meeting the requirements of 10CFRPart 50, § 50.44I(5), regarding containment structural integrity, an analysis must demonstrate containment structural integrity, using an analytical technique that is accepted by the NRC staff and including sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. The analysis must address an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by combustible gas burning. Systems necessary to ensure containment integrity must also demonstrate the capability to perform their functions under these conditions. One acceptable analytical technique is a demonstration that specific criteria of the ASME Boiler and Pressure Vessel Code, described in RG 1.7, Revision 3, Section C.5, are met. 9. In meeting the requirements of 10CFRPart 50, § 50.44I, and GDC 41 for the design and functional capability of the combustible gas control systems, preliminary system designs and statements of intent in the SAR are acceptable at the CP stage of review if the guidelines of RG 1.7, Revision 3, are endorsed.		

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 19 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.2.6 Containment Leakage Testing	The reactor containment leakage rate testing program, as described in the safety analysis report (SAR) or design certification document (DCD), will be acceptable if: A. Under Option A, it meets the requirements stated in Option A of Appendix J to 10CFRPart 50. Appendix J, Option A, provides the test requirements and acceptance criteria for preoperational and periodic leakage rate testing of the reactor containment and of systems and components which penetrate the containment. Exemption from Appendix J requirements will be reviewed on a case-by-case basis. (See additional text on this option in the SRP Acceptance Criteria section) B. Under Option B, it meets the requirements stated in Option B of Appendix J to 10CFRPart 50 and, under section V.B.2 and V.B.3 of Option B, either complies with methods approved by the Commission and endorsed in a regulatory guide (RG 1.163) and includes a requirement to do so in the Technical Specifications, or complies with the provisions of some other implementation document which has been adequately justified to the staff, with supporting analyses, and is cited as a requirement in the Technical Specifications. As of the publication date of this SRP revision, virtually all applicants and licensees using Option B have chosen compliance with RG 1.163, so this Standard Review Plan (SRP) is written assuming that future applicants will do the same. (See additional text on this option in the SRP Acceptance Criteria section)	Conformance with no exceptions identified.	6.2.6
6.2.7 Fracture Prevention of Containment Pressure Boundary	1. To meet the requirements of GDC 1, 16 and 51, ferritic containment pressure boundary materials should meet the fracture toughness criteria and requirements for testing identified in Article NE-2300 of Section III, Division 1 or Article CC-2520 of Section III, Division 2 of the ASME Code or, for materials that were not fracture toughness tested as discussed below, the fracture toughness criteria for Class 2 components identified in the Summer 1977 Addenda to Section III, Division 1, Subsection NC of the ASME Code.	Conformance with no exceptions identified.	6.2.7

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 20 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.2.7 Fracture Prevention of Containment Pressure Boundary (continued)	2. Mandatory fracture toughness testing of ASME Code Section III Class 2 materials was first identified in the Summer 1977 Addenda Code Class 2 rules. As a result, cases exist where Class 2 ferritic materials of the reactor containment pressure boundary were not fracture toughness tested, because the ASME Code Edition and Addenda in effect at the time the components were ordered, did not require that they be tested. The staff's assessment of the fracture toughness of materials that were not fracture toughness tested is based on the metallurgical characterization of these materials and fracture toughness data presented in NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports," and ASME Code Section III, Summer 1977 Addenda, Subsection NC. The metallurgical characterization of these materials, with respect to their fracture toughness, is developed from a review of how these materials were fabricated and what thermal history they experienced during fabrication. The metallurgical characterization of these materials, when correlated with the data presented in NUREG-0577 and the Summer 1977 Addenda of the ASME Code Section III, provides the technical basis for the staff's evaluation of the compliance with Code Class 2 requirements of the materials which were not fracture toughness tested.		

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 21 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.3 Emergency Core Cooling System (continued)	1. In regard to the ECCS acceptance criteria of 10CFR50.46, the five major performance criteria deal with: A. Peak cladding temperature. B. Maximum calculated cladding oxidation. C. Maximum hydrogen generation. D. Coolable core geometry E. Long-term cooling. Guidance, procedures and methods that are acceptable for meeting the requirements for a realistic or best-estimate evaluation model for ECCS performance can be found in Regulatory Guide 1.157. This method must identify and account for uncertainties in the analysis method and inputs such that there is a high level of probability that the acceptance criteria is not exceeded (addresses Generic Issue C-4). Alternatively, Appendix K to 10CFRPart 50 contains guidance for conservative ECCS evaluation models. These areas are reviewed as a part of the effort associated with the LOCA analysis (SRP Section 15.6.5). However, the impact of various postulated single failures on the operability of the ECCS, ECCS response times, break locations (including ECCS break locations), and break sizes impacting ECCS capabilities are evaluated under this SRP section.	Conformance with exceptions. Criterion 9 is applied to BWR. BTP 6-5 item E is applied to traditional PWR with a switchover from the injection mode to the recirculation cooling mode.	6.3

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 22 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.3 Emergency Core Cooling System (continued)	2. The ECCS must meet the requirements of GDC 35. The system must have alternate sources of electric power, as required by GDC 17, and must be able to withstand a single failure. The ECCS should retain its capability to cool the core in the event of a failure of any single active component during the short term immediately following an accident, or a single active or passive failure during the long-term recirculation cooling phase following an accident. A passive failure in a fluid system is a breach in the fluid pressure boundary or mechanical failure that adversely affects a flowpath. SECY-94-084 states the approved position that passive advanced light-water reactor designs need not assume passive component failures in addition to the initiating failure in the application of single-failure criterion to assure safety of the plant. In addition, the staff considers, on a long-term basis, passive component failures in fluid as potential accident initiators, in addition to initiating events. Check valves in the passive safety systems (except those for which proper function can be demonstrated and documented) are considered components subject to single-failure consideration. 3. The ECCS must be designed to permit periodic inservice inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, piping, pumps, and valves in accordance with the requirements of GDC 36. The ECCS must be designed to permit testing of the operability of the system throughout the life of the plant, including the full operational sequence that brings the system into operation, as required by GDC 37. 4. The combined reactivity control system capability associated with ECCS must meet the requirements of GDC 27 and should conform to the recommendation of Regulatory Guide 1.47. The primary mode of actuation for the ECCS must be automatic, and actuation must be initiated by signals of suitable diversity and redundance. Provisions should also be made for manual actuation, monitoring, and control of the ECCS from the reactor control room. 5. The design of the ECCS should conform to the recommendations of Regulatory Guide 1.1.		

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 23 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.3 Emergency Core Cooling System (continued)	6. Design features and operating procedures, designed to prevent damaging water hammer due to such mechanisms as voided discharge lines and water entrainment in steam lines shall be provided, in order to meet the requirements of GDC 4. 7. The design of those portions of the system which are not safety related, whose failures could have an adverse effect on the ECCS system, must be in accordance with GDC 2, and acceptance is based on meeting Position C2 of Regulatory Guide 1.29. Also see SECY-94-084 for policy and technical issues associated with the regulatory treatment of non-safety systems in passive plant designs. 8. Interfaces between the ECCS and component or service water systems must be such that operation of one does not interfere with, and provides proper support (where required) for, the other. In relation to these and other shared systems, e.g., residual heat removal (RHR) and containment heat removal systems, the ECCS must conform to GDC 5. 9. The requirements of Task Action Plan Item II.K.3(15) of NUREG-0737 and NUREG-0718, which involves isolation of HPCI and RCIC for BWR plants, should also be satisfied. 10. The requirements and guidance regarding ECCS outage times and reports on ECCS unavailability, contained in Task Action Plan Item II.K.3.17, and Generic issue B-61, must also be satisfied.		
6.4 Control Room Habitability System	1. Control Room Emergency Zone 2. Ventilation System Criteria 3. Pressurization Systems 4. Emergency Standby Atmosphere Filtration System 5. Relative Location of Source and Control Room 6. Radiation Hazards 7. Toxic Gas Hazards	Conformance with exceptions. The control room habitability during a postulated hazardous chemical release is addressed in COL. The control room will be maintained as the backup center from which technical support center personnel can safely operate in the case of an accident.	6.4

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 24 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.5.1 ESF Atmosphere Cleanup Systems (continued)	1. General Design Criterion (GDC) 19, as it relates to maintaining the control room in a safe condition under accident conditions (LOCAs). 2. GDC 41, as it relates to providing systems to control the release of fission products to the environment and to control the concentration of hydrogen, oxygen, and other substances in containment following postulated accidents. 3. GDC 42, as it relates to designing containment ESF atmosphere cleanup systems to permit inspection. 4. GDC 43, as it relates to designing containment ESF atmosphere cleanup systems to permit pressure and functional testing. 5. GDC 61 as it relates to the design of systems for radioactivity control under normal and postulated accident conditions 6. GDC 64 as it relates to monitoring releases of radioactivity from normal operations, including anticipated operational occurrences, and from postulated accidents. 7. 10CFR52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC)… 8. 10CFR52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses… Relevant aspects of the requirements listed above are met by use of the regulatory positions of Regulatory Guide 1.52 as to the design, testing, and maintenance of ESF atmosphere cleanup system air filtration and adsorption units.	Conformance with no exceptions identified.	6.5.1
6.5.2 Containment Spray as a Fission Product Cleanup System	1. Design Requirements for Fission Product Removal 2. Testing 3. Technical Specifications	Conformance with exceptions. Criterion 3B is not applied to US-APWR, because the US-APWR does not have the containment spray chemical additive tanks.	6.5.2
6.5.3 Fission Product Control Systems and Structures	1. Primary Containment 2. Secondary Containment. 3. (No criterion stated) 4. Other Fission Product Control Systems	Conformance with exceptions. Criterion 2 is not applied to US-APWR, because the US-APWR does not have the secondary containment.	6.5.3

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 25 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.5.4 Ice Condenser as a Fission Product Cleanup System	The acceptance criteria for the fission product cleanup function of ice condenser system are based on the relevant requirements of the following regulations: A. General Design Criterion 41 B. General Design Criterion 42 C. General Design Criterion 43	Not applicable. US-APWR does not have the ice condenser containments.	N/A
6.5.5 Pressure Suppression Pool as a Fission Product Cleanup System	This SRP applies to boiling water reactors and is not applicable to the US-APWR.	Not applicable. Requirements apply only to BWRs.	N/A
6.6 Inservice Inspection and Testing of Class 2 and 3 Components	1. Components Subject to Inspection 2. Accessibility 3. Examination Categories and Methods 4. Inspection Intervals 5. Evaluation of Examination Results 6. System Pressure Tests 7. Augmented ISI to Protect Against Postulated Piping Failures 8. Code Exemptions 9. Relief Requests 10. Code Cases 11. Operational Programs	Conformance with exceptions. Criteria 8,9,10 and 11:There are discussed in inservice inspection program prepared for COL application.	6.5.2
6.7 Main Steam Isolation Valve Leakage Control System (BWR)	This SRP applies to boiling water reactors and is not applicable to the US-APWR	Not applicable. Requirements apply only to BWRs.	N/A
Branch Technical Position 6-1: pH For Emergency Coolant Water for Pressurized Water Reactors	1. Minimum pH should be 7.0. 2. For the spray water recirculated from the containment sump, the higher the pH in the 7.0 to 9.5 range, the greater the assurance that no stress corrosion cracking will occur. See SRP Section 6.5.2 for additional water chemistry requirements related to fission product removal. 3. If a pH greater than 7.5 is used, consideration should be given to the hydrogen generation problem from corrosion of aluminum in the containment.	Conformance with no exceptions identified.	6.1.1.2, 6.3.1.3

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 26 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 6-2: Minimum Containment Pressure Model for PWR ECCS Performance Evaluation	1. Input Information for Model A. Initial Containment Internal Conditions B. Initial Outside Containment Ambient Conditions C. Containment Volume D. Purge Supply and Exhaust Systems 2. Active Heat Sinks A. Spray and Fan Cooling Systems B. Containment Steam Mixing With Spilled ECCS Water C. Containment Steam Mixing With Water from Ice Melt 3. Passive Heat Sinks A. Identification B. Heat Transfer Coefficients	Conformance with no exceptions identified.	6.2.1.5
Branch Technical Position 6-3: Determination of Bypass Leakage Paths in Dual Containment Plants	1. A secondary containment structure should enclose the primary containment structure … 2. Direct leakage from the primary containment to the environment, equivalent to the design leak rate of the primary containment 3. The secondary containment depressurization and filtration systems… 4. For secondary containments… 5. The following leakage barriers in paths which do not terminate within the secondary containment… 6. The total leakage rate for all potential bypass leakage paths… 7. There should be provisions for preoperational and periodic leakage rate testing… 8. If air or water sealing systems or leakage control systems are proposed… 9. If a closed system is proposed as a leakage boundary to preclude bypass leakage…	Not applicable. US-APWR is not a dual containment design.	N/A

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 27 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 6-4: Containment Purging During Normal Plant Operations	1. The on-line purge system should be designed in accordance with the following criteria: A. GDC 54 requires that the reliability and performance capabilities of containment isolation valves reflect the safety importance of isolating the systems penetrating the containment boundary; therefore, the performance and reliability of the purge system isolation valves should be consistent with the operability assurance program of SRP Section 3.10. The design basis for the valves and actuators should include the buildup of containment pressure for the LOCA break spectrum and the supply line and exhaust line flows as a function of time up to and during valve closure. B. The number of supply and exhaust lines should be limited to one supply line and one exhaust line to improve the reliability of the isolation function as required by GDC 54 and to facilitate compliance with the requirements of 10CFRPart 50, Appendix K, for the containment pressure used in the evaluation of ECCS effectiveness and 10CFRPart 100 for offsite radiological consequences. C. The size of the lines should not exceed about eight inches in diameter without detailed justification for larger line sizes to improve the reliability and performance capability of the isolation and containment functions as required by GDC 54 and to facilitate compliance with the requirements of 10CFRPart 50, Appendix K, for the containment pressure used in evaluating ECCS effectiveness and 10CFRPart 100 for the offsite radiological consequences. D. As required by GDC 54, the containment isolation provisions for the purge system lines should meet the standards appropriate to engineered safety features (i.e., quality, redundancy, testability and other appropriate criteria) to reflect the importance to safety of isolating these lines. GDC 56 establishes explicit requirements for isolation barriers in purge system lines.	Conformance with no exceptions identified.	6.2.4, 6.2.6, 9.4.6,15.6.5

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 28 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 6-4: Containment Purging During Normal Plant Operations (continued)	E. To improve the reliability of the isolation function addressed in GDC 54, instrumentation and control systems isolating the purge system lines should be independent and actuated by diverse parameters (e.g., containment pressure, safety injection actuation, and containment radiation level). Furthermore, if energy is required to close the valves, at least two sources of energy must be provided, either of which can effect the isolation function. F. Purge system isolation valve closure times, including instrumentation delays, should not exceed five seconds to facilitate compliance with 10CFRPart 100 for offsite radiological consequences. G. Isolation valve closure must not be prevented by debris which could become entrained in the escaping air and steam. 2. The purge system should not be relied on for temperature and humidity control within the containment. 3. The need for purging of the containment should be minimized by containment atmosphere cleanup systems within the containment. 4. The availability of the isolation function and the leakage rate of the isolation valves during reactor operation should be tested. 5. The following analyses should justify the containment purge system design:		

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 29 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 6-4: Containment Purging During Normal Plant Operations (continued)	A. An analysis of the radiological consequences of a LOCA should be done for a spectrum of break sizes, and the instrumentation and setpoints that will actuate the purge valve closures should be identified. The source term in the radiological calculations should be based on a calculation under the terms of 10CFRPart 50, Appendix K, to the extent of fuel failure and the concomitant release of fission products and the fission product activity in the primary coolant. A pre-existing iodine spike should be considered in determining primary coolant activity. The volume of containment in which fission products are mixed should be justified, and the fission products from the above sources should be assumed to be released through the open purge valves during the maximum interval required for valve closure. The radiological consequences should be within 10CFRPart 100 guideline values. B. An analysis which demonstrates the acceptability of the provisions made to protect structures and safety-related equipment (e.g., fans, filters, and ductwork) located beyond the purge system isolation valves against loss of function in the environment created by the escaping air and steam. C. An analysis of the reduction in the containment pressure caused by the partial loss of containment atmosphere during the accident for ECCS back pressure determination. D. The maximum allowable leak rate of the purge isolation valves should be specified case by case with appropriate consideration for valve size, maximum allowable leakage rate for the containment (as defined in 10CFRPart 50, Appendix J), and, where appropriate, the maximum allowable bypass leakage fraction for dual containments.		

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 30 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 6-5: Currently the Responsibility of Reactor Systems Piping from the RWST (or BWST) and Containment Sump(s)to the Safety Injection Pumps	A. The single active failure criterion defined in (a) and (b) above will be applied in evaluating the design of the piping systems that connect the safety injection pumps to the refueling water storage tank (RWST) or (BWST) and the containment sumps. B. The piping systems, including valves, shall be designed to satisfy the requirements listed below without the need to disconnect the power to any valve. C. The valves and piping between the RWST (or BWST) and the safety injection pumps must be arranged so that no single failure will prevent the minimum flow to the core required to satisfy 10CFR50.46. D. The valves and piping between the RWST (or BWST) and safety injection pumps must be arranged so that no single active failure will result in damage to pumps such that the minimum flow requirements for long-term core and containment cooling after a LOCA are not satisfied. E. The valves and piping that connect the RWST (or BWST) and the containment sump(s) to the safety injection pumps must be arranged so as not to preclude automatic switchover from the injection mode of ECCS operation to recirculation cooling from the sump. These piping systems must be arranged so that the differential pressure between the sump and the RWST (or BWST), even if there is a single active failure, will not result in a loss of core cooling or a path that permits release of radioactive material from the containment to the environment.	Conformance with exceptions. Regarding criterion E, the US-APWR RWSP is inside containment, so no switchover to an external water source is required.	6.3

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 1 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
7.0 Instrumentation and Controls – Overview of Review Process	This SRP does not establish any unique acceptance criteria, but makes reference to other SRP sections (7.1 through 7.9) in which criteria are specified.	Not applicable. SRP establishes no specific acceptance criteria.	N/A
Appendix 7.0-A Review Process for Digital Instrumentation and Control Systems	This SRP Appendix does not establish any unique acceptance criteria, but makes reference to other SRP sections (7.1 through 7.9) in which criteria are specified.	Not applicable. SRP establishes no specific acceptance criteria.	N/A
7.1 Instrumentation and Controls – Introduction	1. SRP Table 7-1, Section 3 (Staff Requirements Memoranda), Section 4 (Regulatory Guides), and Section 5 (Branch Technical Positions), list the SRP acceptance criteria applicable to I&C systems important to safety. Sources of the acceptance criteria are as follows: (Additional text follows on requirements) 2. Use of IEEE Std. 603-1991 and IEEE Std. 279-1971 for Non-Safety Systems (Additional text follows on requirements) 3. Location of Detailed Acceptance Criteria and Review Methods • SRP Appendix 7.1-A provides guidance on the applicability and review methods to be used in evaluating conformance to the regulatory requirements and SRP acceptance criteria for I&C systems important to safety. In three cases the discussion of review methods are extensive and is located in separate appendices that are referenced by SRP Appendix 7.1-A. These appendices are: • SRP Appendix 7.1-B provides guidance for evaluating conformance to the requirements of IEEE Std. 279-1971. • SRP Appendix 7.1-C provides guidance for evaluating conformance to IEEE Std. 603-1991.	Conformance with no exceptions identified.	7.1

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 2 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
7.1 Instrumentation and Controls – Introduction (continued)	SRP Appendix 7.1-D provides guidance for evaluating conformance to SRP acceptance criteria contained in IEEE Std. 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by Regulatory Guide 1.152, Revision 2, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants."		
7.1-T Table 7-1 Regulatory Requirements, Acceptance Criteria, and Guidelines for Instrumentation and Control Systems Important to Safety	SRP Table 7-1 identifies the regulatory requirements (denoted by “R”), and SRP acceptance criteria (denoted by “A”) and their applicability to the various sections of Chapter 7 of the safety analysis report (SAR).	Conformance with no exceptions identified.	7.1
Appendix 7.1-A: Acceptance Criteria and Guidelines for Instrumentation and Control Systems Important to Safety	(Note: This appendix is referenced from SRP 7.1 and contains many pages of acceptance criteria drawn from other regulatory documents such as 10CFRGeneral Design Criteria, Regulatory Guides, other SRP sections, Branch Technical Positions, national standards, etc.).	Conformance with no exceptions identified.	7.1
Appendix 7.1-B: Guidance for Evaluation of Conformance to IEEE Std. 279	(Note: This appendix is referenced from SRP 7.1 and contains many pages of acceptance criteria drawn from IEEE Standard 279, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations.")	Not applicable. US-APWR complies with IEEE Standard 603, which supersedes IEEE 279 for newer plants.	N/A
Appendix 7.1-C: Guidance for Evaluation of Conformance to IEEE Std. 603	(Note: This appendix is referenced from SRP 7.1 and contains many pages of acceptance criteria drawn from IEEE Standard 603, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations”)	Conformance with no exceptions identified.	7.1

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 3 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Appendix 7.1-D: Guidance for Evaluation of Conformance to IEEE Std. 7-4.3.2	(Note: This appendix is referenced from SRP 7.1 and contains many pages of acceptance criteria drawn from Standard IEEE/ANS 7-4.3.2-1982, "American Nuclear Society and IEEE Standard Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations.")	Conformance with no exceptions identified.	7.1
7.2 Reactor Trip System	1. SRP Appendix 7.1-C provides SRP acceptance criteria for safety system compliance with 10CFR50.55a(h). 2. SRP Appendix 7.1-B provides SRP acceptance criteria for protection system compliance with 10CFR50.55a(h) (2). 3. IEEE Std 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by Regulatory Guide 1.152, Revision 2, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants," provides guidance on applying the safety system criteria to computer-based safety systems. SRP Appendix 7.1-D provides SRP acceptance criteria for safety and protection systems using digital computer-based technology. 4. Item II.Q, "Defense Against Common-Mode Failures in Digital Instrument and Control Systems," of the Staff Requirements Memorandum on SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs," provides guidance on Diversity and Defense-in-Depth. SRP BTP 7-19 provides additional guidance.	Conformance with no exceptions identified.	7.2
7.3 Engineered Safety Features Systems	1. SRP Appendix 7.1-C provides SRP acceptance criteria for safety system compliance with 10CFR50.55a(h). 2. SRP Appendix 7.1-B provides SRP acceptance criteria for protection system compliance with 10CFR50.55a(h). 3. IEEE Std 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by Regulatory Guide 1.152, Revision 2, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants," provides guidance on applying the safety system criteria to computer-based safety systems. SRP Appendix 7.1-D provides SRP acceptance criteria for safety and protection systems using digital computer-based technology.	Conformance with no exceptions identified.	7.3

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 4 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
7.3 Engineered Safety Features Systems (continued)	4. Item II.Q, “Defense Against Common-Mode Failures in Digital Instrument and Control Systems,” of the Staff Requirements Memorandum on SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs,” provides guidance on Diversity and Defense-in-Depth. SRP BTP 7-19 provides additional guidance.		
7.4 Safe Shutdown Systems	1. SRP Appendix 7.1-C provides SRP acceptance criteria for safety system compliance with 10CFR50.55a(h). 2. SRP Appendix 7.1-B provides SRP acceptance criteria for protection system compliance with 10CFR50.55a(h). 3. SRP Appendix 7.1-D provides SRP acceptance criteria for the digital I&C compliance with IEEE Std 7-4.3.2-2003, “IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations,” as endorsed by Regulatory Guide 1.152, Revision 2, “Criteria for Digital Computers in Safety Systems of Nuclear Power Plants.”	Conformance with no exceptions identified.	7.4
7.5 Information Systems Important to Safety	1. SRP Appendix 7.1-C provides SRP acceptance criteria for safety system compliance with 10CFR50.55a(h). 2. SRP Appendix 7.1-B provides SRP acceptance criteria for protection system compliance with 10CFR50.5i5a(h). 3. SRP Appendix 7.1-D provides SRP acceptance criteria for the application of the requirements of IEEE Std. 603-1991 to digital I&C. Appendix 7.1-D discusses the application of the guidance in IEEE Std. 7-4.3.2-2003, “IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations,” as endorsed by Regulatory Guide 1.152, Revision 2. 4. Item II.Q, “Defense Against Common-Mode Failures in Digital Instrument and Control Systems,” of the Staff Requirements Memorandum on SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs,” provides guidance on Diversity and Defense-in-Depth. SRP BTP 7-19 provides additional guidance.	Conformance with no exceptions identified.	7.5

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 5 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
7.5 Information Systems Important to Safety (continued)	5. Regulatory Guide 1.97, Revision 2, 3, and 4, describe methods acceptable to the NRC staff for providing instrumentation to monitor variables for accident conditions. For plants with operating licenses issued before June 2006, Regulatory Guide 1.97, Revision 2 and 3, are still effective. Licensees of these plants may, however, convert to the criteria of Revision 4 or use the criteria of Revision 4 when performing modifications that do not involve a conversion. The guidance contained in Regulatory Position 1 of Regulatory Guide 1.97, Revision 4, should be followed in these cases. Plants that obtained an operating license after June 2006 should reference the guidance of Regulatory Guide 1.97, Revision 4. SRP BTP 7-10 provides guidance on the application of Regulatory Guide 1.97.		
7.6 Interlock Systems Important to Safety	1. SRP Appendix 7.1-C provides SRP acceptance criteria for safety system compliance with 10CFR50.55a(h). 2. SRP Appendix 7.1-B provides SRP acceptance criteria for protection system compliance with 10CFR50.55a(h). 3. SRP Appendix 7.1-D provides SRP acceptance criteria for digital I&C compliance with IEEE Std 7-4.3.2-2003, “IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations,” as endorsed by Regulatory Guide 1.152, Revision 2, “Criteria for Digital Computers in Safety Systems of Nuclear Power Plants.”	Conformance with no exceptions identified.	7.6
7.7 Control Systems	1. SRP Appendix 7.1-C provides SRP acceptance criteria for safety system compliance with 10CFR50.55a(h). Although compliance with IEEE Std. 603-1991 is required by 10CFR50.55a(h) only for safety systems, the criteria of IEEE Std. 603-1991 may be used as review guidance for any I&C system. Therefore, for control systems, the reviewer may use the concepts in IEEE Std. 603-1991 as a starting point. 2. SRP Appendix 7.1-B provides SRP acceptance criteria for protection system compliance with 10CFR50.55a(h). 3. SRP Appendix 7.1-D provides SRP acceptance criteria for digital I&C compliance with IEEE Std. 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by Regulatory Guide 1.152, Revision 2, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants."	Conformance with no exceptions identified.	7.7

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 6 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
7.7 Control Systems (continued)	4. Item II.Q, "Defense Against Common-Mode Failures in Digital Instrument and Control Systems," of the Staff Requirements Memorandum on SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs," provides guidance on Defense-in-Depth and Diversity. SRP BTP 7-19 provides additional guidance.		
7.8 Diverse Instrumentation and Control Systems	1. For plants with a digital RTS or ESFAS, the NRC position on D3 should be especially noted. This position is contained in Item II.Q, “Defense Against Common-Mode Failures in Digital Instrument and Control Systems,” of the Staff Requirements Memorandum on SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs.” SRM requirements applicable to diverse I&C functions are as follows: “If a postulated common-mode failure could disable a safety function, then a diverse means, with a documented basis that the diverse means is unlikely to be subject to the same common-mode failure [as the safety system], shall be required to perform either the same function [as the safety system function that is vulnerable to common mode failure] or a different function [that provides adequate protection]. The diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary functions under the associated event conditions.” “A set of displays and controls located in the main control room shall be provided for manual system-level actuation of critical safety functions and monitoring of parameters that support the safety functions. The displays and controls shall be independent and diverse from the safety computer system[s] …”	Conformance with no exceptions identified.	7.8

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 7 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
7.8 Diverse Instrumentation and Control Systems (continued)	2. SRP Appendix 7.1-C provides SRP acceptance criteria for safety system compliance with 10CFR50.55a(h). 3. SRP Appendix 7.1-B provides SRP acceptance criteria for protection system compliance with 10CFR50.55a(h). 4. SRP Appendix 7.1-D provides SRP acceptance criteria for digital I&C compliance with IEEE Std 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by Regulatory Guide 1.152, Revision 2.		
7.9 Data Communication Systems	1. SRP Appendix 7.1-C provides SRP acceptance criteria for safety system compliance with 10CFR50.55a(h). 2. SRP Appendix 7.1-B provides SRP acceptance criteria for protection system compliance with 10CFR50.55a(h). 3. SRP Appendix 7.1-D provides SRP acceptance criteria for digital I&C compliance with IEEE Std. 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by Regulatory Guide 1.152, Revision 2.	Conformance with no exceptions identified.	7.9
Appendix 7-A: General Agenda, Station Site Visits	This SRP Appendix establishes standards for general agendas and site visits. It does not establish any unique acceptance criteria.	Not applicable. SRP establishes no specific acceptance criteria.	N/A
Appendix 7-B Acronyms, Abbreviations, and Glossary	This SRP Appendix establishes standardized acronyms, abbreviations, and glossary. It does not establish any unique acceptance criteria.	Not applicable. SRP establishes no specific acceptance criteria.	N/A

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 8 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 7-1: Guidance on Isolation of Low-Pressure Systems From the High-Pressure Reactor Coolant System	The following measures should be incorporated in designs of the interfaces between low-pressure systems and the high pressure reactor coolant system: 1. At least two valves in series should be provided to isolate any subsystem whenever the primary system pressure is above the pressure rating of the subsystem. 2. For system interfaces where both valves are motor-operated, the valves should have independent and diverse interlocks to prevent both from opening unless the primary system pressure is below the subsystem design pressure. Also, the valve operators should receive a signal to close automatically whenever the primary system pressure exceeds the subsystem design pressure. 3. For those system interfaces where one check valve and one motor-operated valve are provided, the motor-operated valve should be interlocked to prevent the valve from opening whenever the primary pressure is above the subsystem design pressure, and to close automatically whenever the primary system pressure exceeds the subsystem design pressure. 4. Suitable valve position indication should be provided in the control room for the interface valves. 5. For those interfaces where the subsystem is required for emergency core cooling system operation, the above recommendations need not be implemented. System interfaces of this type should be evaluated on an individual basis, as discussed in GL 87-12 and GL 88-17.	Conformance with exceptions. Criteria 1 and 4-6: Conformance with no exceptions identified. Criterion 2: Conformance with exception (The CS/RHR pump hot leg isolation valves are interlocked so that they cannot be opened when the RCS pressure is above 400 psig. In US-APWR, CS/RHR pump suction relief valves provide the low-temperature over-pressure protection for RCS components. Therefore there is no interlock which automatically isolates RHRS from RCS when reactor coolant pressure exceeds the RHR design pressure to ensure performance of the low-temperature over-pressure protection function according to BTP 5-2.) Criterion 3: Not applicable to the US-APWR design certification (There are no such lines except for ECCS in the US-APWR.)	7.6.1

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 9 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 7-1: Guidance on Isolation of Low-Pressure Systems From the High-Pressure Reactor Coolant System (continued)	6. The system should satisfy the requirements of the General Design Criteria and Section 50.55a(h) of 10CFRPart 50b 10CFR50.55a(h), “Protection and Safety Systems,” requires compliance with IEEE Std 603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Station,” and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, the applicant/licensee may elect to comply instead with their plant specific licensing basis. For nuclear power plants with construction permits issued between January 1, 1971 and May 13, 1999, the applicant/licensee may elect to comply instead with the requirements stated in IEEE Std 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations". SRP Appendix 7.1-B provides procedures for reviewing systems against IEEE Std 279-1971. SRP Appendix 7.1-C provides procedures for reviewing systems against IEEE Std 603-1991.		
Branch Technical Position 7-2: Guidance on Requirements of Motor-Operated Valves in the Emergency Core Cooling System Accumulator Lines	The following features should be incorporated into the design of MOIV systems for safety injection tanks to meet the intent of IEEE Std 279-1971 or IEEE Std 603-1991: 1. Automatic opening of the valves when either primary coolant system pressure exceeds a preselected value (to be specified in the technical specifications), or a safety injection signal is present. Both primary coolant system pressure and safety injection signals should be provided to the valve operator. 2. Visual indication in the control room of the open or closed status of the valve. 3. Bypassed and inoperable status indication in accordance to Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety System." 4. Utilization of a safety injection signal to remove automatically (override) any bypass feature that may be provided to allow an isolation valve to be closed for short periods of time when the reactor coolant system is at pressure (in accordance with provisions of the technical specifications).	Conformance with no exceptions identified.	7.6.1

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 10 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 7-3: Guidance on Protection System Trip Point Changes for Operation With Reactor Coolant Pumps Out of Service	1. If more restrictive safety trip points are required for operation with a reactor coolant pump out of service, and if operation with a reactor coolant pump out of service is of sufficient likelihood to be a planned mode of operation, the change to the more restrictive trip points should be accomplished automatically. 2. Plants with designs not in accordance with the above should have included in the plant technical specifications a requirement that the reactor be shut down prior to changing the set points manually.	Not applicable. US-APWR design does not have a situation for RTS Restrictive setpoints.	7.2.3.4
Branch Technical Position 7-4: Guidance on Design Criteria for Auxiliary Feedwater Systems	The auxiliary feedwater system should be capable of satisfying the system functional requirements after a postulated break in the auxiliary feedwater piping inside containment together with a single electrical failure. The basis for the position is that an auxiliary feedwater piping break would result in tripping the unit and, in turn, might cause loss of offsite power. Standard staff assumptions for analyzing postulated accidents include the assumption of loss of offsite power if the affected unit generator is tripped by the accident. Such a circumstance would leave the plant without adequate means for removal of afterheat even though the reactor coolant pressure boundary was intact - an unacceptable result. Plant heat removal systems should, in any postulated piping break, be capable of removing afterheat to the ultimate heat sink assuming a single electrical (active) failure anywhere in the auxiliary feedwater system or in the onsite power system.	Conformance with no exceptions identified.	7.3.1.5

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 11 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 7-5: Guidance on Spurious Withdrawals of Single Control Rods in Pressurized Water Reactors	GDC 20 requires that the protection system shall be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences. GDC 25 requires that these limits shall not be exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection) of control rods. Within the context of GDC 20 the staff considers operator error to be an anticipated operational occurrence, in addition to the consideration of single malfunction requirements of GDC 25, for which conformance to these requirements is to be evaluated. The applicant should perform analyses of the reactivity control systems 1 and analyze the consequences of operator error to assess the impact of these events on fuel design limits. If the results of these analyses show that specified acceptable fuel design limits may be exceeded for these events, the protection system must be designed to detect and terminate these events prior to exceeding these limits. With regard to the evaluation of malfunctions within the reactivity control systems, consideration should be given to failures that cause actions as well as prevent actions, such that all possible effects are examined. Further, failures that could lead to single or multiple rod position changes or out-of-sequence rod patterns should be analyzed, as well as failures that could lead to reactivity changes by boron control systems.	Conformance with no exceptions identified.	7.7.1.1
Branch Technical Position 7-6: Guidance on Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recirculation Mode	1. A design that provides manual initiation at the system level of the transfer to the recirculation mode, while not ideal, is sufficient and satisfies the intent of IEEE Std 279-1971 or IEEE Std 603-1991, provided that adequate instrumentation and information display are available to the operator so that he or she can make the correct decision at the correct time. Furthermore, it should be shown that, in case of operator error, sufficient time and information are available so that the operator can correct the error, and that the consequences of such an error are acceptable. 2. Automatic transfer to the recirculation mode is preferable to manual transfer, for the reasons cited above, and should be provided for standard plant designs submitted for review on a generic basis under the Commission's standardization policy.	Not applicable. In the US-APWR, the refueling water storage pit inside the containment is used for the source of the emergency core cooling. It is not necessary to change from injection to recirculation mode.	N/A

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 12 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 7-8: Guidance for Application of Regulatory Guide 1.22	All portions of the protection and safety systems should be designed in accordance with IEEE Std 279-1971 or IEEE Std 603-1991, as required by 10CFR50.55a(h), "Protection and Safety Systems." All actuated equipment that is not tested during reactor operation should be identified, and a discussion of how each conforms to the provisions of paragraph D.4 of RG 1.22 should be submitted. In addition to compliance with RG 1.22, the review of this topic should also confirm that the proposed design and the justification for test intervals are consistent with the surveillance testing proposed as part of the plant technical specifications. 1. The protection system should satisfy the requirements of the General Design Criteria and Section 50.55a(h) of 10CFRPart 50. 10CFR50.55a(h) requires compliance with IEEE Std 603-1991, and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, the applicant/licensee may elect to comply instead with their plant-specific licensing basis. For nuclear power plants with construction permits issued between January 1, 1971 and May 13, 1999, the applicant/licensee may elect to comply instead with the requirements stated in IEEE Std 279-1971. SRP Appendix 7.1-B provides guidance for reviewing systems against IEEE Std 279-1971. SRP Appendix 7.1-C provides guidance for reviewing systems against IEEE Std 603-1991.	Conformance with no exceptions identified.	7.1.3.11, 7.1.3.14
Branch Technical Position 7-9: Guidance on Requirements for Reactor Protection System Anticipatory Trips	All reactor trips incorporated in the reactor protection system should be designed to meet the requirements of IEEE Std 279-1971, or IEEE Std 603-1991. This position applies to the entire trip function, from the sensor to the final actuated device. For sensors located in non-seismic areas, the installation (including circuit routing) and design should be such that the effects of credible faults (i.e., grounding, shorting, application of high voltage, or electromagnetic interference) or failures in these areas could not be propagated back to the reactor protection system and degrade the reactor protection system performance or reliability. The sensors should be qualified to operate in a seismic event, i.e., not fail to initiate a trip for conditions which would cause a trip.	Conformance with no exceptions identified.	7.1.2

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 13 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 7-10: Guidance on Application of Regulatory Guide 1.97	The design and qualification criteria identified in Regulatory Guide 1.97 should be supplemented by the considerations outlined below: • Environmental Qualification (Additional text follows on requirements) • Seismic Qualification (Additional text follows on requirements) • Redundancy (Additional text follows on requirements) • Independence of Redundant Instrumentation (Additional text follows on requirements) • Display and Recording (Additional text follows on requirements) • Range (Additional text follows on requirements) • Minimizing Measurements (Additional text follows on requirements) • Alternate Instrumentation (Additional text follows on requirements) • Guidance for Boiling Water Reactor (BWR) and Pressurized Water Reactor (PWR) Variables *(Additional text follows on requirements)*	Conformance with no exceptions identified.	7.5.1.1, 7.5.2.1

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 14 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 7-11: Guidance on Application and Qualification of Isolation Devices	General acceptance guidelines for application and qualification are provided in IEEE Std 603-1991, or IEEE Std 279-1971, and Regulatory Guide 1.75. Acceptance criteria for the descriptions of the device application, device design, test methods, and test results are as follows: • Description of Device Application (Additional text follows on requirements) • Description of Device Design (Additional text follows on requirements) • Description of Test Method (Additional text follows on requirements) • Description of Test Results (Additional text follows on requirements)	Conformance with no exceptions identified.	7.1.3.5
Branch Technical Position 7-12: Guidance on Establishing and Maintaining Instrument Setpoints	Setpoint Documentation - The following information on the licensee/applicant's setpoint program should be provided for review: *(Additional text follows on requirements)* • Analysis Supporting Establishment of Setpoints and Instrumentation Tolerances *(Additional text follows on requirements)* • Statistical Guidelines for Instrument Uncertainty *(Additional text follows on requirements)* • Guidelines for Graded Approach (Additional text follows on requirements) • Basis for Instrument Calibration Intervals *(Additional text follows on requirements)*	Conformance with no exceptions identified.	7.2.2.7, 7.3.2.7

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 15 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 7-13: Guidance on Cross-Calibration of Protection System Resistance Temperature Detectors	Supporting Analysis (Additional text follows on requirements) • Traceability of the Installed Reference RTD to Laboratory Calibration Data (Additional text follows on requirements) • Acceptable Methods for In-Situ Testing (Additional text follows on requirements) • Response Time Testing (Additional text follows on requirements) • Control/Protection Interaction and Common-Cause Failure During In-Situ Testing *(Additional text follows on requirements)*	Conformance with no exceptions identified.	7
Branch Technical Position 7-14: Guidance on Software Reviews for Digital Computer-Based Instrumentation and Controls Systems	B.3.1 Acceptance Criteria for Planning (Additional text follows on requirements) B.3.2 Acceptance Criteria for Implementation *(Additional text follows on requirements)* B.3.3 Acceptance Criteria for Design Outputs *(Additional text follows on requirements)*	Conformance with no exceptions identified.	7.1.3.17
Branch Technical Position 7-16 (Withdrawn) Guidance on Level of Detail Required for Design Certification Applications Under 10CFRPart 52	This BTP has been withdrawn.	Not applicable. SRP has been withdrawn by NRC.	N/A

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 16 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 7-17: Guidance on Self-Test and Surveillance Test Provisions	Surveillance test and self-test features for digital computer-based protection systems should conform to the guidance of Regulatory Guide 1.22 and Regulatory Guide 1.118. Bypasses necessary to enable testing should conform with the guidance of Regulatory Guide 1.47. • Failure Detection (Additional text follows on requirements) • Self-Test Features (Additional text follows on requirements) • Surveillance Testing (Additional text follows on requirements) • Actions on Failure Detection (Additional text follows on requirements)	Conformance with no exceptions identified.	7.1.3.11, 7.1.3.14
Branch Technical Position 7-18: Guidance on the Use of Programmable Logic Controllers in Digital Computer-Based Instrumentation and Control Systems	Purchased PLC hardware; embedded and operating systems software, programming tools, and peripheral components should be qualified to a level commensurate with the system they are designed to support. EPRI TR-106439 and EPRI TR-107330 describe an acceptable process for qualifying commercial systems. NUREG/CR-6421 provides additional information on the characteristics of an acceptable process for qualifying existing software, and discusses the use of engineering judgment and compensating factors for purchased PLC software. See the discussion of the commercial dedication of predeveloped software (PDS) in SRP Appendix 7.0-A. PLC hardware, embedded and operating system software, and peripheral components built specifically for nuclear power plant applications should meet the appropriate quality criteria. The embedded and operating system software should meet the acceptance criteria contained in SRP BTP 7-14, appropriately graded for the application in which the PLC will be used. The application software (ladder logic or other) should meet the acceptance criteria contained in SRP BTP 7-14 commensurate with the system it is designed to support. Application software should conform with the recommended practices of NUREG/CR-6463. Tools for developing application software or loading it into the PLC should be qualified to a level commensurate with the system they are designed to support.	Not applicable. No Programmable Logic Controllers are used in US-APWR design.	N/A

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 17 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 7-18: Guidance on the Use of Programmable Logic Controllers in Digital Computer-Based Instrumentation and Control Systems (continued)	PLC-based functions should conform with the guidance regarding real-time performance and testing outlined in SRP BTP 7-21 and SRP BTP 7-17. Administrative or hardware lockout controls that prevent unauthorized modification of the PLC software should be in place. This is particularly important because many PLCs are designed so that their software is easy to modify. All software changes should be under configuration management control. In particular, administrative procedures for maintaining control of the software implemented in the PLC should be detailed in the configuration management plan.		
Branch Technical Position 7-19: Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems	1. For each anticipated operational occurrence in the design basis occurring in conjunction with each single postulated common-cause failure, the plant response calculated using best-estimate (realistic assumptions) analyses should not result in radiation release exceeding 10 percent of the 10CFR100 guideline value or violation of the integrity of the primary coolant pressure boundary. The applicant/licensee should (1) demonstrate that sufficient diversity exists to achieve these goals, (2) identify the vulnerabilities discovered and the corrective actions taken, or (3) identify the vulnerabilities discovered and provide a documented basis that justifies taking no action. 2. For each postulated accident in the design basis occurring in conjunction with each single postulated common-cause failure, the plant response calculated using best-estimate (realistic assumptions) analyses should not result in radiation release exceeding the 10CFR100 guideline values, violation of the integrity of the primary coolant pressure boundary, or violation of the integrity of the containment (i.e., exceeding coolant system or containment design limits). The applicant/licensee should (1) demonstrate that sufficient diversity exists to achieve these goals, (2) identify the vulnerabilities discovered and the corrective actions taken, or (3) identify the vulnerabilities discovered and provide a documented basis that justifies taking no action.	Conformance with no exceptions identified.	7.1.3.1, 7.8

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 18 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 7-19: Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems (continued)	3. When a failure of a common element or signal source shared by the control system and RTS is postulated and the common-cause failure results in a plant response that requires reactor trip and also impairs the trip function, then diverse means that are not subject to or failed by the postulated failure should be provided to perform the RTS function. The diverse means should assure that the plant response calculated using best-estimate (realistic assumptions) analyses does not result in radiation release exceeding 10 percent of the 10CFR100 guideline value or violation of the integrity of the primary coolant pressure boundary. 4. When a failure of a common element or signal source shared by the control system and ESFAS is postulated and the common-cause failure results in a plant response that requires engineered safety features (ESF) and also impairs the ESF function, then diverse means that are not subject to or failed by the postulated failure should be provided to perform the ESF function. The diverse means should assure that the plant response calculated using best-estimate (realistic assumptions) analyses does not result in radiation release exceeding 10 percent of the 10CFR100 guideline value or violation of the integrity of the primary coolant pressure boundary. 5. No failure of monitoring or display systems should influence the functioning of the RTS or ESFAS. If plant monitoring system failure induces operators to attempt to operate the plant outside safety limits or in violation of the limiting conditions of operation, the analysis should demonstrate that such operator-induced transients will be compensated for by protection system function.		

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 19 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 7-21: Guidance on Digital Computer Real-Time Performance	If the following criteria are met, the Staff may conclude that the design or completed system will meet timing requirements, can be verified as correct and timely, or that a prototype system accurately reflects the performance and correctness expected of the actual plant. Some of the criteria described herein may be met by submissions describing a software development process or verification methods that include real-time concerns. • Limiting Response Times (Additional text follows on requirements) • Digital Computer Timing Requirements (Additional text follows on requirements) • Architecture (Additional text follows on requirements) • Design Commitments (Additional text follows on requirements) • Performance Verification (Additional text follows on requirements) • Use of Cyclic Real-Time Executive (Additional text follows on requirements) • Use of Part-Scale Prototypes (Additional text follows on requirements)	Conformance with no exceptions identified.	7.2.2.7.2, 7.9.2.3

Table 1.9.2-8 US-APWR Conformance with Standard Review Plan Chapter 8 Electrical Power (sheet 1 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
8.1 Electric Power – Introduction	Specific SRP acceptance criteria are contained in SRP Sections 8.2, 8.3.1, 8.3.2, and 8.4. (This SRP does not contain any unique acceptance criteria, but references other SRP sections.)	Not applicable. SRP establishes no specific acceptance criteria.	N/A (Discussed, however, in 8.1.1)
8.2 Offsite Power System	1. GDC 2 is satisfied as it relates to SSCs of the offsite power system being capable of withstanding the effects of natural phenomena such as high and low atmospheric temperatures, high wind, rain, lightning discharges, ice and snow conditions, and weather events causing regional effects as established in Chapter 3 of the SAR, and reviewed by the organizations with primary responsibility for the reviews of plant systems, civil engineering and geosciences, and mechanical engineering. 2. GDC 4 is satisfied as it relates to SSCs of the offsite power system being protected against dynamic effects, including the effects of missile that may result from equipment failures during normal operation, maintenance, testing, and postulated accidents, as established in Chapter 3 of the SAR and reviewed by the organizations with primary responsibility for the reviews of plant systems, materials, and chemical engineering. 3. GDC 5 is satisfied as it relates to: sharing of SSCs of the preferred power systems; guidelines of Regulatory Guide 1.32 as related to its endorsement of Section 7 of IEEE Std 308, relating to sharing of SSCs of the Class 1E power system at multi-unit stations; and guidance related to the sharing of SSCs of the offsite power system (preferred power supply) at multi-unit stations, previously addressed in the 1980 and earlier versions of IEEE Std 308, but now covered in the industry standard for preferred power supply (Reference 52). 4. GDC 17 is satisfied as it relates to the preferred power system's (i) capacity and capability to permit functioning of SSCs important to safety; (ii) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies; (iii) physical independence; (iv) availability and the guidelines of Regulatory Guide 1.32 (see also IEEE Std 308) as related to the availability and number of immediate access circuits from the transmission	Conformance with no exceptions identified.	8.2

Table 1.9.2-8 US-APWR Conformance with Standard Review Plan Chapter 8 Electrical Power (sheet 2 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
8.2 Offsite Power System (continued)	network; and (v) capability to meet the guidelines of Appendix A to SRP Section 8.2 as related to acceptability of generator circuit breakers and generator load break switches. For evolutionary light water reactor design applications, as documented in SECY 94-084 for designs such as the CE-ABB System 80+ and the GE ABWR, the design should provide at least one offsite circuit to each redundant safety division that is supplied directly from an offsite power source with no intervening non-safety buses, thereby permitting the offsite source to supply power for safety buses in the event the non-safety bus(es) fails. The design should also include an alternate power source to non-safety loads, unless it can be demonstrated that existing design margins will ensure that transients for loss of non-safety power events are no more severe than those associated with the turbine-trip-only event specified in current plant designs (References 33 and 35). These issues are reviewed in detail in SRP Section 8.3.1. For passive reactor design applications, the passive safety-related systems only require electric power for valves and related instrumentation, which can be supplied from the onsite Class 1E batteries and associated dc and ac distribution systems. The acceptability of this design for the AP 1000 is documented in SECY-05-0227 and FSER NUREG-1793. If no offsite power is available, it is expected that the non-safety-related diesel generators would be available for important plant functions, but this non-safety related ac power is not relied on to maintain core cooling or containment integrity. Therefore, this passive reactor design supports an exemption to the requirement of GDC 17 for two physically independent offsite circuits, by providing safety-related passive safety systems for core cooling and containment integrity (see also References 33, 34, 35). However, one offsite power source with sufficient capacity and capability from the transmission network must be provided to power the safety-related systems and all other auxiliary systems under normal, abnormal, and accident conditions. The offsite power source should be designed to minimize to the extent practical the likelihood of its failure under normal, abnormal, and accident conditions.		

Table 1.9.2-8 US-APWR Conformance with Standard Review Plan Chapter 8 Electrical Power (sheet 3 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
8.2 Offsite Power System (continued)	5. GDC 18 is satisfied as it relates to the inspection and testing of the offsite electric power system. 6. GDCs 33, 34, 35, 38, 41, and 44 are satisfied as they relate to the operation of the offsite electric power system, encompassed in GDC 17, to ensure that the safety functions of the systems described in GDC's 33, 34, 35, 38, 41, and 44 are accomplished, assuming a single failure where applicable. 7. 10CFR50.63 is satisfied as it relates to an AAC power source (as defined in 10CFR50.2) provided for safe shutdown in the event of a station blackout (non-DBA), and the guidelines of Regulatory Guide 1.155 are followed as they relate to the adequacy of the AAC source and the independence of the AAC power source from the offsite power system and onsite power system and sources. Except for passive reactor designs described in subsection II(2) above, new applications must provide an adequate AAC source of diverse design (with respect to ac onsite emergency sources) that is consistent with the guidance in Regulatory Guide 1.155 and capable of powering at least one complete set of normal safe shutdown loads. These issues are reviewed in detail in SRP Section 8.4. 8. 10CFR50.65, Section 50.65(a)(4), as it relates to the requirements to assess and manage the increase in risk that may result from proposed maintenance activities before performing the maintenance activities. Acceptance is based on meeting the following specific guidelines: A. Regulatory Guide 1.160, as related to the effectiveness of maintenance activities for onsite emergency ac power sources including grid-risk-sensitive maintenance activities (i.e., activities that tend to increase the likelihood of a plant trip, increase LOOP frequency, or reduce the capability to cope with a LOOP or SBO). B. Regulatory Guide 1.182, as related to implementing the provisions of 10CFR50.65 (a)(4) by endorsing Section 11 to NUMARC 93-01, "Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, February 22, 2000.		

Table 1.9.2-8 US-APWR Conformance with Standard Review Plan Chapter 8 Electrical Power (sheet 4 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
8.3.1 ac/AC Power Systems (Onsite)	In general, the onsite ac power system is acceptable when it can be concluded that this system has the required redundancy, meets the single failure criterion, is protected from the effects of postulated accidents, is testable, and has the capacity, capability, and reliability to supply power to all safety loads and other required equipment in accordance with GDCs 2, 4, 5, 17, 18, and 50. Table 8-1 of SRP 8.1 lists GDCs, regulations, regulatory guides, and branch technical positions used as the bases for arriving at this conclusion. 1. GDC 2 is satisfied as it relates to SSCs of the onsite ac power system being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods, as established in Chapter 3 of the SAR, and reviewed by the organizations with primary responsibility for the reviews of plant systems, civil engineering and geosciences, and mechanical engineering. 2. GDC 4 is satisfied as it relates to SSCs of the ac power system being capable of withstanding the effects of missiles and environmental conditions associated with normal operation and postulated accidents, as established in Chapter 3 of the SAR and reviewed by the organizations with primary responsibility for the reviews of plant systems, materials, and chemical engineering. 3. GDC 5 is satisfied as it relates to the sharing of SSCs of the ac power system and the following guidelines: (Additional text follows on requirements) 4. GDC 17 is satisfied as it relates to the onsite ac power system's: (a) capacity and capability to permit functioning of SSCs important to safety; (b) independence, redundancy, and testability to perform its safety function assuming a single failure; and (c) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or the loss of power from the transmission network. Acceptance is based on meeting the following specific guidelines: (Additional text follows on requirements)	Conformance with no exceptions identified. Note of clarification: The US-APWR design utilizes gas turbine generators to fulfill the functional requirements for emergency power. Some of the SRP descriptions contain an assumption that diesel generators will be utilized.	8.3

Table 1.9.2-8 US-APWR Conformance with Standard Review Plan Chapter 8 Electrical Power (sheet 5 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
8.3.1 ac/AC Power Systems (Onsite) (continued)	5. GDC 18 is satisfied as it relates to the testability of the onsite ac power system, and the following guidelines: (Additional text follows on requirements) 6. The design requirements for an onsite ac power supply for systems covered by GDCs 33, 34, 35, 38, 41, and 44 are encompassed in GDC 17. 7. GDC 50 is satisfied as it relates to the design of containment electrical penetrations containing circuits of the ac power system, and the guidelines of Regulatory Guide 1.63 are followed (see also IEEE Stds 242, 317, and 741), as related to the capability of electric penetration assemblies in containment structures to withstand a LOCA without loss of mechanical integrity and the external circuit protection for such penetrations, as well as to ensure that electrical penetrations will withstand the full range of fault current (minimum to maximum) available at the penetration. 8. 10CFR50.63, as it relates to use of the redundancy and reliability of diesel generator units as a factor in limiting the potential for station blackout events. Acceptance is based on meeting the following specific guidelines: (Additional text follows on requirements) 9. 10CFR50.65, Section 50.65(a)(4), as it relates to the requirements to assess and manage the increase in risk that may result from proposed maintenance activities before performing the maintenance activities. Acceptance is based on meeting the following specific guidelines: (Additional text follows on requirements)		

Table 1.9.2-8 US-APWR Conformance with Standard Review Plan Chapter 8 Electrical Power (sheet 6 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
8.3.1 ac/AC Power Systems (Onsite) (continued)	10. 10CFR50.55a(h) as it relates to protection systems for plants with construction permits issued after January 1, 1971, but before May 13, 1999, which must meet the requirements stated in either IEEE Std. 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," or IEEE Std. 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, protection systems must be consistent with their licensing basis or may meet the requirements of IEEE Std 279-1971. Nuclear power plants with applications filed on or after May 13, 1999 for preliminary and final design approvals (10CFRPart 52, Appendix O), design certification, construction permits, operating licenses, and combined licenses that do not reference a final design approval or design certification, must meet the requirements for safety systems in IEEE Std 603-1991 and the correction sheet dated January 30, 1995.		
8.3.2 DC Power Systems (Onsite)	1. Regulatory Guide 1.6 positions D.1, D.3, and D.4, as they relate to the independence between redundant onsite dc power sources and between their distribution systems. 2. Regulatory Guide 1.32, as it relates to the design, operation, and testing of the safety-related portions of the onsite dc power system. Except for sharing of safety-related dc power systems in multi-unit nuclear power plants, RG 1.32 endorses IEEE Std. 308-2001. 3. Regulatory Guide 1.75, as it relates to the physical independence of the circuits and electrical equipment that comprise or are associated with the onsite dc power system. 4. Regulatory Guide 1.81, as it relates to the sharing of SSCs of the dc power system. Regulatory Position C.1 states that multi-unit sites should not share dc systems. 5. Regulatory Guide 1.128, as it relates to the installation of vented lead-acid storage batteries in the onsite dc power system.	Conformance with no exceptions identified.	8.3.2

Table 1.9.2-8 US-APWR Conformance with Standard Review Plan Chapter 8 Electrical Power (sheet 7 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
8.3.2 DC Power Systems (Onsite) (continued)	6. Regulatory Guide 1.129, as it relates to maintenance, testing, and replacement of vented lead-acid storage batteries in the onsite dc power system. 7. Regulatory Guide 1.118, as it relates to the capability to periodically test the onsite dc power system. 8. Regulatory Guide 1.153, as it relates to the design, reliability, qualification, and testability of the power, instrumentation, and control portions of safety systems of nuclear plants, including the application of the single failure criterion in the onsite dc power system. As endorsed by Regulatory Guide 1.153, IEEE Std. 603 provides a method acceptable to the staff to evaluate all aspects of the electrical portions of the safety-related systems, including basic criteria for addressing single failures. However, as stated in 10CFR55a(h), all plants are not required to comply with IEEE Std. 603. Only applications filed on or after May 13, 1999, for preliminary and final design approvals (10CFRPart 52, Appendix O), design certification, and construction permits; operating licenses and combined licenses that do not reference a final design approval or design certification must meet the requirements for safety systems in IEEE Std. 603-1991 and the correction sheet dated January 30, 1995. Operating nuclear power plants are encouraged, but not required to, comply with IEEE Std. 603 for future system-level modifications. 9. Regulatory Guide 1.53, as it relates to the application of the single-failure criterion. 10. Regulatory Guide 1.63, as it relates to the capability of electric penetration assemblies in containment structures to withstand a loss of coolant accident without loss of mechanical integrity and the external circuit protection for such penetrations. 11. Regulatory Guide 1.155, as it relates to the capability and the capacity of the onsite dc power system for an SBO, including batteries associated with the operation of the alternate ac (AAC) power source(s) (if used).		

Table 1.9.2-8 US-APWR Conformance with Standard Review Plan Chapter 8 Electrical Power (sheet 8 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
8.3.2 DC Power Systems (Onsite) (continued)	12. The guidelines of Regulatory Guide 1.160, as they relate to the effectiveness of maintenance activities for dc power systems. Compliance with the maintenance rule, including verification that appropriate maintenance activities are covered therein, is reviewed under SRP Chapter 17. 13. The guidelines of Regulatory Guide 1.182, as they relate to conformance to the requirements of 10CFR50.65(a)(4) for assessing and managing risk when performing maintenance.		
8.4 Station Blackout	1. The guidelines of RG 1.155, as they relate to compliance to 10CFR50.63. NUMARC-8700, Revision 0, also provides guidance acceptable to the staff for meeting these requirements. Table 1 of RG 1.155 provides a cross-reference to NUMARC-8700, Revision 0, and notes when the RG takes precedence. 2. The guidelines and criteria of SECY-90-016 and SECY-94-084 (Ref. 25), as they relate to the use of AAC power sources and RTNSS at plants provided with passive safety systems. 3. The guidelines of RGs 1.9 (Ref. 6) and 1.155, as they relate to the reliability program implemented to ensure that the target reliability goals for onsite EAC power sources (typically diesel generator units) are adequately maintained. 4. The guidelines of RG 1.160 (Ref. 8), as they relate to the effectiveness of maintenance activities for onsite EAC power sources, including grid-risk-sensitive maintenance activities (i.e., activities that tend to increase the likelihood of a plant trip, increase LOOP frequency, or reduce the capability to cope with a LOOP or SBO). Compliance with the maintenance rule, including verification that appropriate maintenance activities are covered therein, is reviewed under SRP Chapter 17. 5. The guidelines of RG 1.182 (Ref. 9), as they relate to conformance to the requirements of 10CFR50.65(a)(4) for assessing and managing risk when performing maintenance.	Conformance with no exceptions identified.	8.4
Appendix 8-A General Agenda, Station Site Visits	This SRP Appendix establishes standards for general agenda and station site visits, and does not contain any unique acceptance criteria.	Not applicable. SRP establishes no specific acceptance criteria.	N/A

Table 1.9.2-8 US-APWR Conformance with Standard Review Plan Chapter 8 Electrical Power (sheet 9 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 8-1: Requirements on Motor-Operated Valves in the ECCS Accumulator Lines	To meet the intent of IEEE Std. 279, the design of the MOIV system should incorporate the following features for safety injection tanks: 1. Automatic opening of the valves when either primary coolant system pressure exceeds a preselected value (to be specified in the technical specifications) or a safety injection signal is present. Both primary coolant system pressure and safety injection signals should be provided to the valve operator. 2. Visual indication in the control room of the open or closed status of the valve. 3. An audible and visual alarm, independent of item 2, above, that is actuated by a sensor on the valve when the valve is not in the fully open position. 4. Use of a safety injection signal to remove automatically (override) any bypass feature that may be provided to allow an isolation valve to be closed for short periods of time when the reactor coolant system is at pressure (in accordance with provisions of the technical specifications). Conformance with the relevant criteria for operating bypasses described in IEEE Std. 603, as endorsed in RG 1.153, constitutes an acceptable alternative approach.	Conformance with no exceptions identified.	8.1.5
Branch Technical Position 8-2: Use of Diesel-Generator Sets for Peaking	The staff's position regarding the use of onsite emergency power diesel-generator sets for purposes other than that of supplying standby power when needed is that such use should be prohibited. In particular, emergency power diesel-generator sets should not be used for peaking service.	Conformance with no exceptions identified. US-APWR has no diesel generators, but will use gas turbine generators for emergency power in the standard design.	8.1 (Note: MHI has generated a position on the use of gas turbine generators for emergency power that meets the intent of the SRP)
Branch Technical Position 8-3: Stability of Offsite Power Systems	1. The staff has concluded, from a review of appropriate reliability data, that power systems with supporting grid interties meet the grid availability criterion with some margin. This conclusion is applicable to the review of most plants located on the U.S. mainland.	Not applicable to US-APWR design certification. Consideration of this topic will be site-specific and the responsibility of the COL Applicant.	N/A (BTP and offsite power system discussed in section 8.1)

Table 1.9.2-8 US-APWR Conformance with Standard Review Plan Chapter 8 Electrical Power (sheet 10 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 8-3: Stability of Offsite Power Systems (continued)	2. A strong indication exists that an isolated system large enough to justify inclusion of a nuclear unit will also meet this criterion. However, as a conservative approach, the staff will examine the generating capacity of a system, including interties if available, available to withstand outage of the largest unit. If the available capacity is judged marginal in its ability to provide adequate stability of the grid, additional measures should be taken. These may include provisions for additional capability and margin for the onsite power system beyond the normal requirements or other measures that may be appropriate in a particular case. The additional measures to be taken should be determined on an individual case basis.		
Branch Technical Position 8-4: Application of the Single Failure Criterion to Manually Controlled Electrically Operator Valves	1. Failures of components in electrical systems, including valves and other fluid system components, in both the “fail to function” sense and the “undesirable function” sense, should be considered in designing against a single failure, even though the valve or other fluid system component may not be called upon to function in a given safety operational sequence. 2. When it is determined that failure of an electrical system component can cause undesired mechanical motion of a valve or other fluid system component, and this motion results in loss of the system safety function, it is acceptable, in lieu of design changes that also may be acceptable, to disconnect power to the electric systems of the valve or other fluid system component. The plant technical specifications should include a list of all electrically operated valves, and the required positions of these valves, to which the requirement for removal of electric power is applied in order to satisfy the single failure criterion.	Conformance with no exceptions identified.	8.1.5

Table 1.9.2-8 US-APWR Conformance with Standard Review Plan Chapter 8 Electrical Power (sheet 11 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 8-4: Application of the Single Failure Criterion to Manually Controlled Electrically Operator Valves (continued)	3. Electrically operated valves that are classified as “active” valves (i.e., are required to open or close in various safety system operational sequences, but are manually controlled) should be operated from the main control room. Such valves may not be included among those valves from which power is removed in order to meet the single failure criterion unless (1) electrical power can be restored to the valves from the main control room, (2) valve operation is not necessary for at least 10 minutes following occurrence of the event requiring such operation, and (3) it is demonstrated that there is reasonable assurance that all necessary operator actions will be performed within the time shown to be adequate by the analysis. The plant technical specifications should include a list of the required positions of manually controlled, electrically operated valves and should identify those valves to which the requirement for removal of electric power is applied in order to satisfy the single failure criterion. 4. When the single failure criterion is satisfied by removal of electrical power from valves described in items 2 and 3, above, these valves should have redundant position indication in the main control room, and the position indication system should, itself, meet the single failure criterion. 5. The phrase “electrically operated valves” includes both valves operated directly by an electrical device (e.g., a motor-operated valve or a solenoid-operated valve) and those valves operated indirectly by an electrical device (e.g., an air-operated valve with an air supply controlled by an electrical solenoid valve).		

Table 1.9.2-8 US-APWR Conformance with Standard Review Plan Chapter 8 Electrical Power (sheet 12 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 8-5: Supplemental Guidance for Bypass and Inoperable Status Indication for Engineered Safety Features Systems	The design criteria for bypass and inoperable status indication systems for ESFs should reflect the importance of providing accurate information for the operator and reducing the possibility for the indicating equipment to adversely affect the monitored safety systems. In developing the design criteria, the following should be considered: 1. The bypass indicators should be arranged to enable the operator to determine the status of each safety system and whether continued reactor operation is permissible. 2. When a protective function of a shared system can be bypassed, indication of that bypass condition should be provided in the control room of each affected unit. 3. The means by which the operator can cancel erroneous bypass indications, if provided, should be justified by demonstrating that the postulated cases of erroneous indications cannot be eliminated by another practical design. 4. Unless the indication system is designed in conformance with criteria established for safety systems, it should not be used to perform functions that are essential to safety. Administrative procedures should not require immediate operator action based solely on the bypass indications. 5. The indication system should be designed and installed in a manner that precludes the possibility of adverse effects on plant safety systems. Failure or bypass of a protective function should not be a credible consequence of failures occurring in the indication equipment, and the bypass indication should not reduce the required independence between redundant safety systems. 6. The indication system should include a capability of assuring its operable status during normal plant operation to the extent that the indicating and annunciating function can be verified.	Conformance with no exceptions identified.	8.1.5

Table 1.9.2-8 US-APWR Conformance with Standard Review Plan Chapter 8 Electrical Power (sheet 13 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 8-6: Adequacy of Station Electric Distribution System Voltages	1. In addition to the undervoltage scheme provided to detect LOOP at the Class 1E buses, a second level of undervoltage protection with time delay should be provided to protect the Class 1E equipment. This second level of undervoltage protection should satisfy the following criteria: (Additional text follows on requirements) 2. The Class 1E bus load shedding scheme should automatically prevent shedding during sequencing of the emergency loads to the bus. The load shedding feature should, however, be reinstated upon completion of the load sequencing action. The technical specifications must include a test requirement to demonstrate the operability of the automatic load shedding features at least once every refueling outage/cycle. An adequate basis must be provided if the load shedding feature is retained during the above load sequencing of the emergency loads to the bus. 3. The voltage levels at the safety-related buses should be optimized for the maximum and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite power sources by appropriate adjustment of the voltage tap settings of the intervening transformers. The tap settings selected should be based on an analysis of the voltage at the terminals of the Class 1E loads. The analyses performed to determine minimum operating voltages should typically consider maximum unit steady-state and transient loads for events, such as a unit trip, loss-of-coolant accident, startup or shutdown, with the offsite power supply (grid) at minimum anticipated voltage and only the offsite source being considered available. Maximum voltages should be analyzed with the offsite power supply (grid) at maximum expected voltage concurrent with minimum unit loads (e.g., cold shutdown, refueling). A separate set of the above analyses should be performed for each available connection to the offsite power supply.	Conformance with no exceptions identified.	8.1.5, 8.3.1

Table 1.9.2-8 US-APWR Conformance with Standard Review Plan Chapter 8 Electrical Power (sheet 14 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 8-6: Adequacy of Station Electric Distribution System Voltages (continued)	4. The analytical techniques and assumptions used in the voltage analyses cited in item 3 above must be verified by actual measurement. The verification and test should be performed before initial full-power reactor operation on all sources of offsite power by taking the following actions: (Additional text follows on requirements)		
Branch Technical Position 8-7: Criteria for Alarms and Indications Associated with Diesel-Generator Unit Bypassed and Inoperable Status	1. Diesel-generator unit bypass or deliberately induced inoperability status should be automatically indicated in the control room when the bypass or deliberately induced inoperable condition can be expected to occur more frequently than once per year and can render the unit unavailable to adequately respond to an automatic or operator-initiated emergency demand. Manually induced indication may be desirable and is permitted for diesel-generator unit bypass or deliberately induced inoperability status for those conditions expected to occur less frequently than once per year. 2. All status indication should be sufficiently precise to prevent misinterpretation. Furthermore, disabling or bypass indicators should be separate from nondisabling indicators and should be physically arranged to enable the operator to clearly determine the status of each diesel-generator unit. An acceptable design includes a separate alarm for each disabling condition or a single shared alarm with reflash capability. The alarms should be displayed in the control room and at the diesel-generator unit for all disabling conditions, with wording that indicates that the diesel-generator unit is incapable of adequately responding to an emergency demand. 3. When a shared diesel-generator unit can be bypassed, indication of that bypass condition should be provided in the control room of each affected unit.	Conformance with no exceptions identified. US-APWR has no diesel generators, but will use gas turbine generators for emergency power in the standard design.	8.1.5, 8.3.1

Table 1.9.2-8 US-APWR Conformance with Standard Review Plan Chapter 8 Electrical Power (sheet 15 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 8-7: Criteria for Alarms and Indications Associated with Diesel-Generator Unit Bypassed and Inoperable Status (continued)	4. The indication system should be designed and installed to preclude the possibility of adverse effects on the diesel-generator units. Failures in the indication equipment should not result in diesel-generator unit failure or bypass of the diesel-generator unit, and the bypass indication should not reduce the required independence between redundant diesel-generator units. 5. The indication system should be capable of ensuring its operable status during normal plant operation to the extent that the indicating and annunciating function can be verified. 6. RG 1.9, positions C.1.6 through C.1.8, contains further guidance to be addressed regarding status and anomalous conditions indication and alarms for diesel-generators.		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 1 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.1.1 Criticality Safety of Fresh and Spent Fuel Storage and Handling	1. The criteria for GDC 62 are specified in American National Standards Institute (ANSI)/American Nuclear Society (ANS) 57.1, ANSI/ANS 57.2, and ANSI/ANS 57.3, as they relate to the prevention of criticality accidents in fuel storage and handling.	Conformance with no exceptions identified.	9.1.1
9.1.2 New and Spent Fuel Storage	1. Acceptance for meeting the relevant aspect of GDC 2 is based on compliance with positions C.1 and C.2 of Regulatory Guide (RG) 1.13 and applicable portions of RG 1.29, and RG 1.117. For the spent fuel storage facility, additional guidance acceptable for meeting this criterion is found in American Nuclear Society (ANS) 57.2, 9.1.2-5 paragraphs 5.1.1, 5.1.3, 5.1.12.9, and 5.3.2. For the new fuel storage facility, additional guidance acceptable for meeting this criterion is found in ANS 57.3, paragraphs 6.2.1.3(2), 6.2.3.1, 6.3.1.1, 6.3.3.4, and 6.3.4.2. 2. Acceptance for meeting the relevant aspect of GDC 4 is based on positions C.2 and C.3 of RG 1.13, and RG 1.115 and 1.117. 3. GDC 5 is met by sharing the SSCs important to safety between the units in a manner that does not degrade the performance of their safety functions. 4. Acceptance for meeting the relevant aspect of GDC 61 for the spent fuel storage facility is based on compliance with positions C.4, C.6, C.10, C.11, and C.12 of RG 1.13 and the appropriate paragraphs of ANS 57.2. Acceptance for meeting this criterion for the new fuel storage facility is based on compliance with the appropriate paragraphs of ANS 57.3. Acceptance is also based on meeting the fuel storage capacity requirements noted in subsection III.1 of this SRP section. The following design considerations are evaluated:	Conformance with exceptions. Criterion 3 is not applicable for US-APWR design certification. (US-APWR is a single unit.)	9.1.2

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 2 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.1.2 New and Spent Fuel Storage (continued)	A. Provisions for periodic inspections of components important to safety. B. Suitable shielding for radiation protection, including adequate water levels. C. Appropriate containment and confinement systems. D. Residual heat removal capability by effective coolant flow through the storage racks for spent fuel assemblies. E. Prevention of reduction in fuel storage coolant inventory under accident conditions. 5. Acceptance for meeting the relevant aspect of GDC 63 for spent fuel storage is based on compliance with position C.7 of RG 1.13 and paragraph 5.4 of ANS 57.2. Acceptance for meeting this criterion for the dry storage of new fuel is based on radiation monitoring pursuant to 10CFR70.24 or acceptable prevention of an increase in effective multiplication factor (Keff) beyond safe limits as described in 10CFR50.68. 6. In meeting the requirements of 10CFR20.1101(b), positions C.2.f (2) and C.2.f (6) of RG 8.8 are the bases for acceptance with respect to provisions for decontamination. For spent fuel storage, paragraph 5.1.5 of ANS 57.2 and appropriate positions of RG 1.13 are the bases for acceptance. For new fuel storage, paragraphs 6.3.3.7 and 6.3.4 of ANS 57.3 are the bases for acceptance. 7. 10CFR50.68 allows the applicant to follow the guidelines of 10CFR70.24 for criticality monitors or the guidelines described therein for significant margins of subcriticality.		
9.1.3 Spent Fuel Pool Cooling and Cleanup System	1. General Design Criterion (GDC) 2 contained in Appendix A to 10CFRPart 50, as related to structures housing the system and the system itself being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, and hurricanes.	Conformance with no exceptions identified.	9.1.3

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 3 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.1.3 Spent Fuel Pool Cooling and Cleanup System (continued)	Acceptance for meeting this criterion is based on conformance to positions C.1, C.2, C.6, and C.8 of RG 1.13 and position C.1 of RG 1.29 for safety-related and position C.2 of RG 1.29 for nonsafety-related portions of the system. This criterion does not apply to the cleanup portion of the system and need not apply to the cooling system if the fuel pool makeup water system and its source meet this criterion, the fuel pool building and its ventilation and filtration system meet this criterion, and the ventilation and filtration system meets the guidelines of RG 1.52. The cooling and makeup system should be designed to Quality Group C requirements in accordance with RG 1.26. However, when the cooling system is not designated Category I it need not meet the requirements of ASME Section XI for inservice inspection of nuclear plant components. 2. GDC 4 with respect to the capability of the system and the structure housing the system to withstand the effects of external missiles. Acceptance is based on meeting position C.2 of RG 1.13. This criterion does not apply to the cleanup system and need not apply to the cooling water system if the makeup system, its source, the building, and its ventilation and filtration system are tornado protected, and the ventilation and filtration system 3. GDC 5 as related to shared systems and components important to safety being capable of performing required safety functions. 4. GDC 61 as related to the system design for fuel storage and handling of radioactive materials, including the following elements:meets the guidelines of RG 1.52.		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 4 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.1.3 Spent Fuel Pool Cooling and Cleanup System (continued)	A. The capability for periodic testing of components important to safety B. Provisions for containment. C. Provisions for decay heat removal that reflects its importance to safety. D. The capability to prevent reduction in fuel storage coolant inventory under accident conditions. E. The capability and capacity to remove corrosion products, radioactive materials, and impurities from the pool water and reduce occupational exposures to radiation. 5. GDC 63 as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal, to detect excessive radiation levels, and to initiate appropriate safety actions. 6. 10CFR20.1101(b) as it relates to radiation doses being kept ALARA. In meeting this regulation, RG 8.8, positions C.2.f (2) and C.2.f (3) can be used as a basis for acceptance. 7. 10CFR52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations. 8. 10CFR52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations. Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations are included in the Requirements subsection, above.		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 5 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.1.4 Light Load Handling System (Related to Refueling)	1. Acceptance for meeting the relevant aspects of GDC 2 is based on RG 1.29, Positions C.1 and C.2. 2. Acceptance for meeting the relevant aspects of GDC 5 is embodied within the other acceptance criteria 3. Acceptance for meeting the relevant aspects of GDC 61 is based in part on the guidelines of American National Standards Institute/American Nuclear Society (ANSI/ANS) 57.1-1992. 4. Acceptance for meeting the relevant aspects of GDC 62 is based in part on ANSI/ANS 57.1-1992.	Conformance with exceptions. Criterion 2 is not applicable for US-APWR design certification. (US-APWR is a single unit.)	9.1.4
9.1.5 Overhead Heavy Load Handling Systems	1. Acceptance for meeting the relevant aspects of GDC 1 is based in part on NUREG-0554 for overhead handling systems and ANSI N14.6 or ASME B30.9 for lifting devices. 2. Acceptance for meeting the relevant aspects of GDC 2 is based in part on position C.2 of RG 1.29 and Section 2.5 of NUREG-0554. 3. Acceptance for meeting the relevant aspects of GDC 4 is based in part on position C.5 of RG 1.13. 4. Acceptance for meeting the relevant aspects of GDC 5 is embodied within the other acceptance criteria.	Conformance with exceptions. Criterion 4 is not applicable for US-APWR design certification. (US-APWR is a single unit.)	9.1.5
9.2.1 Station Service Water System	1. Protection Against Natural Phenomena. Information that addresses the requirements of GDC 2 regarding the capability of structures housing the EESWS and the ESWS itself to withstand the effects of natural phenomena will be considered acceptable if the guidance of Regulatory Guide (RG) 1.29, Position C.1 for safety-related portions of the ESWS and Position C.2 for nonsafety-related portions of the ESWS are appropriately addressed. 2. Environmental and Dynamic Effects. Information that addresses the requirements of GDC 4 regarding consideration of environmental and dynamic effects will be considered acceptable if the acceptance criteria in the following SRP sections, as they apply to the EESWS, are met: SRP Sections 3.5.1.1, 3.5.1.4, 3.5.2, and SRP Section 3.6.1. In addition, the information will be considered acceptable if the design provisions presented in GL 96-06 and to GL 96-06, Supplement 1 are appropriately addressed.	Conformance with no exceptions identified.	9.2.1

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 6 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.2.1 Station Service Water System (continued)	3. Sharing of SSCs. Information that addresses the requirements of GDC 5 regarding the capability of shared systems and components important to safety to perform required safety functions will be considered acceptable if the use of the ESWS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the unaffected unit(s). In addition, the information will be considered acceptable if the provisions GL 89-13 and GL 91-13 are appropriately addressed. 4. Cooling Water System. Information that addresses the requirements of GDC 44 regarding consideration of the cooling water system will be considered acceptable if a system to transfer heat from SSCs important to safety to an ultimate heat sink is provided. In addition, the ESWS can transfer the combined heat load of these SSCs under normal operating and accident conditions, assuming loss of offsite power and a single failure, and that system portions can be isolated so the safety function of the system is not compromised. 5. Cooling Water System Inspection. Information that addresses the requirements of GDC 45 regarding the inspection of cooling water systems will be considered acceptable if the design of the ESWS permits inservice inspection of safety-related components and equipment and operational functional testing of the system and its components. 6. Cooling Water System Testing. Information that addresses the requirements of GDC 46 regarding the testing of cooling water systems will be considered acceptable if the ESWS is designed for testing to detect degradation in performance or in the system pressure boundary so that the ESWS will function reliably to provide decay heat removal and essential cooling for safety-related equipment.		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 7 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.2.2 Reactor Auxiliary Cooling Water Systems	1. Protection Against Natural Phenomena. Information that addresses the requirements of GDC 2 regarding the capability of structures housing the reactor auxiliary CWS and the reactor auxiliary CWS itself to withstand the effects of natural phenomena will be considered acceptable if the guidance of Regulatory Guide (RG) 1.29, Position C.1 for safety-related portions of the reactor auxiliary CWS and Position C.2 for nonsafety-related portions of the reactor auxiliary CWS are appropriately addressed. 2. Environmental and Dynamic Effects. Information that addresses the requirements of GDC 4 regarding consideration of environmental and dynamic effects will be considered acceptable if the acceptance criteria in the following SRP sections, as they apply to the reactor auxiliary CWS, are met: SRP Sections 3.5.1.1, 3.5.1.4, 3.5.2, and SRP Section 3.6.1. In addition, the information will be considered acceptable if the design provisions presented in GL 96-06 and GL 96-06, Supplement 1 are appropriately addressed. 3. Sharing of SSCs. Information that addresses the requirements of GDC 5 regarding the capability of shared systems and components important to safety to perform required safety functions will be considered acceptable if the use of the reactor auxiliary CWS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the unaffected unit(s).	Conformance with no exceptions identified.	9.2.2

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 8 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.2.2 Reactor Auxiliary Cooling Water Systems (continued)	4. Cooling Water System. Information that addresses the requirements of GDC 44 regarding consideration of the cooling water system will be considered acceptable if the reactor auxiliary CWS and its components will continue to perform their required safety functions, assuming a single, active failure or a moderate-energy line crack as defined in Branch Technical Position ASB 3-1 and to seismic Category I, Quality Group C, and American Society of Mechanical Engineers (ASME) Section III Class 3 requirements concurrent with the loss of offsite power. In addition, the information will be considered acceptable based on appropriate application of IEEE Std 603, as endorsed by RG 1.153, and appropriate application of RG 1.155, Position C.3.3.4. 5. Cooling Water System Inspection. Information that addresses the requirements of GDC 45 regarding the inspection of cooling water systems will be considered acceptable if the periodic inspection of important reactor auxiliary CWS components ensures system integrity and capability to perform design safety functions. 6. Cooling Water System Testing. Information that addresses the requirements of GDC 46 regarding the testing of cooling water systems will be considered acceptable if periodic system pressure and function testing of the reactor auxiliary CWS will ensure the leak-tight integrity and operability of its components, as well as the operability of the system as a whole, at conditions as close to the design basis as practical		
9.2.3 (Withdrawn) Demineralized Water Makeup System	This SRP has been withdrawn.	Not applicable. SRP has been withdrawn by NRC.	N/A

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 9 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.2.4 Potable and Sanitary Water Systems	1. Control of Releases of Radioactive Materials to the PWSW. Information that addresses the requirements of GDC 60 in regards to controlling radioactive effluent releases is considered acceptable if the following are met: A. There are no interconnections between the PESWS and systems having the potential for containing radioactive material. B. The potable water system is protected by an air gap, where necessary. C. An evaluation of potential radiological contamination, including accidental, and safety implications of sharing (for multi-unit facilities) indicates that the system will not result in contamination beyond acceptable limits.	Conformance with exceptions. Safety implications of sharing (for multi-unit facilities) of criteria 1C is N/A.)	9.2.4, 9.2.5
9.2.5 Ultimate Heat Sink	1. GDC 2 as to capability of structures housing the system and the system itself to withstand the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods. 2. GDC 5 as to capability of shared systems and components important to safety to perform required safety functions. 3. GDC 44 as to: A. The capability to transfer heat loads from safety-related SSCs to the heat sink under both normal operating and accident conditions. B. Suitable component redundancy so that safety functions can be performed assuming a single, active component failure coincident with loss of offsite power. C. The capability to isolate components, systems, or piping if required so safety functions are not compromised. 4. GDC 45 as to the design provisions to permit inservice inspection of safety-related components and equipment. 5. GDC 46 as to the design provisions to permit operation functional testing of safety related systems or components.	Conformance with no exceptions identified.	9.2.5, 14.3, also Tier 1 section 2.7 for ITAAC

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 10 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.2.5 Ultimate Heat Sink (continued)	6. 10CFR52.47(b) (1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations. 7. 10CFR52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
9.2.6 Condensate Storage Facilities	1. Protection Against Natural Phenomena. Acceptance for meeting the relevant aspects of GDC 2 is based in part on meeting the guidance of Position C.1 of Regulatory Guide 1.29 if any portion of the system is deemed to be safety related and the guidance of Position C.2 for nonsafety-related portions. Also, acceptance is based in part on (1) meeting the guidance of Regulatory Guide 1.117 with respect to identifying portions of the system that should be protected from tornadoes and (2) meeting the guidance of Regulatory Guide 1.102 with respect to identifying portions of the system that should be protected from flooding. 2. Sharing of SSCs. Information that addresses the requirements of GDC 5 regarding the capability of shared systems and components important to safety to perform required safety functions will be considered acceptable if the use of the CSF in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the unaffected unit(s).	Not applicable. US-APWR design does not have a condensate storage system.	N/A

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 11 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.2.6 Condensate Storage Facilities (continued)	3. Condensate Storage Facility. Information that addresses the requirements of GDC 44 regarding consideration of the cooling water system will be considered acceptable if a system to transfer heat from SSCs important to safety to an ultimate heat sink is provided. In addition, the CSF can transfer the combined heat load of these SSCs under normal operating and accident conditions, assuming loss of offsite power and a single failure, and that system portions can be isolated so the safety function of the system is not compromised. 4. Condensate Storage Facility Inspection. Information that addresses the requirements of GDC 45 regarding the inspection of cooling water systems will be considered acceptable if the design of the CSF permits inservice inspection of safety-related components and equipment and operational functional testing of the system and its components. 5. Condensate Storage Facility Testing. Information that addresses the requirements of GDC 46 regarding the testing of cooling water systems will be considered acceptable if the CSF is designed for testing to detect degradation in performance or in the system pressure boundary so that the CSF will function reliably to provide decay heat removal and essential cooling for safety-related equipment. 6. Control of Radioactive Releases to the Environment. Acceptance for meeting the relevant aspects of GDC 60 is based on meeting the guidance of Regulatory Guide 1.143. 7. Loss of All Alternating Current Power. Acceptance for meeting the relevant aspects of 10CFR50.63 is based on meeting the guidance of Regulatory Guide 1.155.		
9.3.1 Compressed Air System	1. Acceptance for meeting the relevant aspect of GDC 1 is based on compliance with the criteria specified in American National Standards Institute/Instrument Society of America (ANSI/ISA) S7.3-R1981 related to minimum instrument air quality standards.	Conformance with exceptions. Criterion 3, the instrument air system of the US-APWR is not shared. Criterion 4, US-APWR can cope with a station blackout [SBO] without air supply from the instrument air system.	9.3.1

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 12 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.3.1 Compressed Air System (continued)	2. Acceptance for meeting the relevant requirements of GDC 2 as it relates to seismic classification is based on compliance to guidance provided in RG 1.29, Positions C.1 and C.2. 3. Acceptance for meeting the relevant requirements of GDC 5 as it relates to the sharing of safety-related SSCs is based on the criteria set forth here for CAS SSCs shared among multiple units. 4. Acceptance for meeting the relevant requirements of 10CFR50.63 as it relates to the CAS design and the ability of a plant to withstand for a specified duration and recover from a station blackout is based on RG 1.155.		
9.3.2 Process and Post-Accident Sampling Systems	1. The applicant's design is such that the PSS has the capability to sample all normal process systems and principal components, including provisions for obtaining samples from at least the points indicated below. The guidelines of Regulatory Guide (RG) 1.21, Position C.2, the Electric Power Research Institute (EPRI) BWR Water Chemistry Guidelines, and the Electric Power Research Institute (EPRI) PWR Water Chemistry Guidelines are used to meet the requirements of the relevant GDC. 2. The plant Technical Specifications include the required analysis and frequencies. 3. The following guidelines should be used to determine the acceptability of the PSS functional design: A. Provisions should be made to ensure representative samples from liquid process streams and tanks. For tanks, provisions should be made to sample the bulk volume of the tank and to avoid sampling from low points or from potential sediment traps. For process stream samples, sample points should be located in turbulent flow zones. The guidelines of Regulatory Position C.6 in RG 1.21 are followed to meet these criteria.	Conformance with no exception identified. (DCD Chapter16 should include the required analysis and frequencies per criteria 2.)	9.3.2

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 13 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.3.2 Process and Post-Accident Sampling Systems (continued)	B. Provisions should be made to ensure representative samples from gaseous process streams and tanks in accordance with American National Standards Institute/Health Physics Society (ANSI/HPS) Standard N13.1-1999. The guidelines of Regulatory Position C.6 in RG 1.21 are followed to meet this criterion. C. Provisions should be made for purging sampling lines and for reducing plateout in sample lines (e.g., heat tracing). The guidelines of Regulatory Position C.7 in RG 1.21 are followed to meet this criterion. D. Provisions should be made to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system in accordance with the requirements of 10CFR20.1101(b) to keep radiation exposures at ALARA levels. The guidelines of Regulatory Positions 2.d.(2), 2.f.(3), and 2.f.(8) in RG 8.8 are followed to meet this criterion. E. Isolation valves should fail in the closed position, in accordance with the requirements of GDC 60 to control the release of radioactive materials to the environment. F. Passive flow restrictions to limit reactor coolant loss from a rupture of the sample line should be provided in accordance with the requirements of 10CFR20.1101(b) to keep radiation exposures to ALARA levels and the requirements of GDC 60 to control the release of radioactive materials to the environment. The guidelines of Regulatory Position 2.i. (6) in RG 8.8 should be followed to meet this criterion. Redundant environmentally qualified, remotely operated isolation valves may replace passive flow restrictions in the sample lines to limit potential leakage. The automatic containment isolation valves should close on containment isolation signals or safety injection signals.		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 14 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.3.2 Process and Post-Accident Sampling Systems (continued)	4. To meet the requirements of GDCs 1 and 2, the applicant's seismic design and quality group classification of sampling lines, components, and instruments for the PSS should conform to the classification of the system to which each sampling line and component is connected (e.g., a sampling line connected to a Quality Group A and seismic Category I system should be designed to Quality Group A and seismic Category I classification), in accordance with Regulatory Positions C.1, C.2, and C.3 in RG 1.26; Regulatory Positions C.1, C.2, C.3, and C.4 in RG 1.29, and the guidelines of RG 1.97. Components and piping downstream of the second isolation valve may be designed to Quality Group D and nonseismic Category I requirements, in accordance with Regulatory Position C.3 in RG 1.26.		
9.3.3 Equipment and Floor Drainage System	1. Protection Against Natural Phenomena. Information that addresses the requirements of GDC 2 regarding the capability of safety-related system portions of the EFDS to withstand the effects of natural phenomena. Comprehensive compliance with GDC 2 is reviewed under other SRP sections as specified in subsection I of this SRP section. If no portion is safety-related, the EFDS need not meet GDC 2. 2. Environmental and Dynamic Effects. Information that addresses the requirements of GDC 4 regarding the capability to withstand the effects of and to be compatible with the environmental conditions (flooding) of normal operation, maintenance, testing, and postulated accidents (pipe break, tank ruptures) will be considered acceptable if the EFDS is designed to prevent flooding that could affect SSCs important to safety (i.e., necessary for safe shutdown, accident prevention, or accident mitigation) adversely. 3. Control of Releases of Radioactive Material to the Environment. Information that addresses the requirements of GDC 60 regarding the suitable control of the release of radioactive materials in liquid effluent, including anticipated operational occurrences will be considered acceptable if the EFDS is designed to prevent the inadvertent transfer of contaminated fluids to a non-contaminated drainage system for disposal.	Conformance with no exceptions identified.	9.3.3

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 15 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.3.4 Chemical and Volume Control System (PWR) (Including Boron Recovery System)	1. The CVCS safety-related functional performance should be maintained in the event of adverse environmental phenomena such as earthquakes, tornadoes, hurricanes, and floods, or in the event of certain pipe breaks or loss of offsite power. For compliance with GDC 29, 33 and 35, the CVCS should provide sufficient pumping capacity to supply borated water to the RCS, maintain RCS water inventory within the allowable pressurizer level range for all normal modes of operation, and function as part of the ECCS, if so designed, to supply reactor coolant makeup in the event of small pipe breaks assuming a single active failure coincident with the loss of offsite power. In addition, Regulatory Guide 1.155 describes a means acceptable to the NRC staff for meeting the requirements of 10CFR50.63, “Loss of all ac/AC power.” If the CVCS is necessary to support a plant SBO coping capability as required by 10CFR50.63, the positions in Regulatory Guide 1.155 regarding CVCS design provide an acceptable method for showing compliance. 2. SECY-77-439 describes the concept of single failure criteria and the application of the single failure criterion that involves a systematic search for potential single failure points and their effects on prescribed missions. Application of the single failure assumption in system design and analysis provides redundancy and defense-in-depth to ensure functional performance of the CVCS. Also, the requirements of GDC 5 prohibiting the sharing among nuclear units the SSCs important to safety would be met by the use of a separate CVCS for each unit.	Conformance with no exceptions identified.	9.3.4

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 16 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.3.4 Chemical and Volume Control System (PWR) (Including Boron Recovery System) (continued)	3. 10CFR50.55(a) requires that components of the RCPB be designed, fabricated, erected, and tested in accordance with the requirements for Class 1 components of Section III of the ASME Boiler and Pressure Vessel Code or equivalent quality standards. Regulatory Guide 1.26 describes a quality classification system that may be used to determine quality standards acceptable to the NRC staff for satisfying GDC 1 for other safety related components containing water, steam, or radioactive materials in light-water-cooled nuclear power plants. RG 1.29 describes a method acceptable to the NRC staff for identifying and classifying those features of LWRs that should be designed to withstand the effects of the safe shutdown earthquake (SSE). The requirements of GDC 1 regarding the quality standard are met by acceptable application of quality group classifications and application of quality standards as described in RG 1.26. The requirement of GDC 2 regarding the protection against natural phenomena are met by meeting the guidance of RG 1.29, Position C.1, for safety-related portions of the system and Position C.2 for nonsafety-related portion. 4. The CVCS design and arrangement should be that all components and piping that can contain boric acid will either be heat traced or will be located within heated rooms to prevent precipitation of boric acid. As additional specific criteria used to review the CVCS and BRS design, the CVCS should include provisions for monitoring: (a) temperature upstream of the demineralizer to assure that resin temperature limits are not exceeded, and (b) filter demineralizer differential pressure to assure that pressure differential limits are not exceeded. In addition, the CVCS should have provision for automatically diverting or isolating the CVCS flow to the demineralizer in the event the demineralizer influent temperature exceeds the resin temperature limit.		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 17 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.3.4 Chemical and Volume Control System (PWR) (Including Boron Recovery System) (continued)	5. 10CFR50.34(f)(2)(xxvi), as applicable, specifies the provisions regarding detection of reactor coolant leakage outside containment. These requirements will be met, in part, by providing leakage control and detection systems in the CVCS and implementation of appropriate leakage control program. 6. Implementation of Action 1 specified in Bulletin 80-05 provides an acceptable means for the system to prevent the CVCS holdup tanks, which can contain radioactive release, from the formation of such vacuum conditions that could cause wall inward buckling and failure. The requirements of GDC 60 and 61 can be met, in part, by providing in the CVCS appropriately designed venting and draining closed systems to confine the radioactivity associated with the effluents. 7. 10CFR52.47(a)(1)(vi) specifies that the application of a design certification should contain proposed ITAAC necessary and sufficient to assure the plant is built and will operate in accordance with the design certification. 10CFR52.97(b)(1) specifies that the COL identifies the ITAAC necessary and sufficient to assure that the facility has been constructed and will be operated in conformity with the license. SRP 14.3 provides guidance for reviewing the ITAAC. The requirements of 10CFR52.47(a)(1)(vi) and 10CFR52.97(b)(1) will be met, in part, by identifying inspections, tests, analyses, and acceptance criteria of the top-level design features of the CVCS in the design certification application and the combined license, respectively.		
9.3.5 Standby Liquid Control System (BWR)	This SRP applies to boiling water reactors (BWRs) and is not applicable to the US-APWR.	Not applicable. The SRP applies to BWRs only.	N/A

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 18 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.4.1 Control Room Area Ventilation System	1. Protection Against Natural Phenomena. Information that addresses the requirements of GDC 2 regarding the capability of structures housing the CRAVS and the CRAVS itself to withstand the effects of natural phenomena will be considered acceptable if the guidance of Regulatory Guide (RG) 1.29, Position C.1 for safety-related portions of the CRAVS and Position C.2 for nonsafety-related portions of the CRAVS are appropriately addressed. 2. Environmental and Dynamic Effects. Information that addresses the requirements of GDC 4 regarding consideration of environmental and dynamic effects will be considered acceptable if the acceptance criteria in the following SRP sections, as they apply to the CRAVS, are met: SRP Sections 3.5.1.1, 3.5.2, and 3.6.1. 3. Sharing of SSCs. Information that addresses the requirements of GDC 5 regarding the capability of shared systems and components important to safety to perform required safety functions will be considered acceptable if the use of the CRAVS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the remaining unit(s). 4. Control Room. Information that addresses the requirements of GDC 19 regarding the capability of the control room to remain functional to the degree that actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain the plant in a safe condition under accident conditions, including loss-of-coolant accidents will be considered acceptable if adequate protection against radiation and hazardous chemical releases are provided to permit access to and occupancy of the control room under accident conditions. RG 1.78 provide guidance acceptable to the staff for meeting these control room occupancy protection requirements.	Conformance with exceptions. Criteria 1, 2, 5, 6: Conformance with no exceptions identified. Criterion 3: Not applicable to US-APWR design certification (Not multiple unit plants) Criterion 4: The postulated hazardous chemical release is COLA Specific.	9.4.1

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 19 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.4.1 Control Room Area Ventilation System (continued)	5. Control of Releases of Radioactive Material to the Environment. Information that addresses the requirements of GDC 60 regarding the suitable control of the release of gaseous radioactive effluents to the environment will be considered acceptable if the guidance of RGs 1.52 and 1.140 as related to design, inspection, testing, and maintenance criteria for post-accident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and adsorption units of light-water-cooled nuclear power plants are appropriately addressed. For RG 1.52 rev 2, the applicable regulatory position is C.2. For RG 1.52 rev 3, the applicable regulatory position is C.3. For RG 1.140 rev 1, the applicable regulatory positions are C.1 and C.2. For RG 1.140 rev 2, the applicable regulatory positions are C.2 and C.3. 6. Loss of All Alternating Current Power. Information that addresses the requirements of 10CFR50.63 regarding the necessary support systems providing sufficient capacity and capability for coping with a station blackout event will be considered acceptable if the guidance of RG1.155, including position C.3.2.4 is applied appropriately.		
9.4.2 Spent Fuel Pool Area Ventilation System	1. For GDC 2, acceptance is based on the guidance of RG 1.29, Position C.1 for safety-related portions and Position C.2 for nonsafety-related portions. 2. For GDC 5, acceptance is based on the determination that the use of the SFPAVS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the remaining unit(s).	Conformance with exceptions. Criterion 2 is N/A.(Not multiple unit plants) Criterion 3 is N/A. (Not air .cleanup system) Criterion 4 is N/A.(satisfy the limit offsite dose consequences from fuel handling area without ESF ventilation (filtration) system.)	9.4.2

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 20 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.4.2 Spent Fuel Pool Area Ventilation System (continued)	3. For GDC 60, acceptance is based on the guidance of RGs 1.52 and 1.140 as related to design, inspection, testing, and maintenance criteria for post-accident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and adsorption units of light-water-cooled nuclear power plants. For RG 1.52 rev 2, the applicable regulatory position is C.2. For RG 1.52 rev 3, the applicable regulatory position is C.3. For RG 1.140 rev 1, the applicable regulatory positions are C.1 and C.2. For RG 1.140 rev 2, the applicable regulatory positions are C.2 and C.3. 4. For GDC 61, acceptance is based on the guidance of RG 1.13 as to the design of the ventilation system for the spent fuel storage facility, Position C.4.		
9.4.3 Auxiliary and Radwaste Area Ventilation System	1. For GDC 2, acceptance is based on the guidance of RG 1.29, Position C.1 for safety-related portions, and Position C.2 for nonsafety-related portions. 2. For GDC 5, acceptance is based on the determination that the use of the ARAVS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the remaining unit(s). 3. For GDC 60, acceptance is based on the guidance of RGs 1.52 and 1.140 as related to design, inspection, testing, and maintenance criteria for post-accident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and adsorption units of light-water-cooled nuclear power plants. For RG 1.52 rev 2, the applicable regulatory position is C.2. For RG 1.52 rev 3, the applicable regulatory position is C.3. For RG 1.140 rev 1, the applicable regulatory positions are C.1 and C.2. For RG 1.140 rev 2, the applicable regulatory positions are C.2 and C.3.	Conformance with exceptions. Criterion 2:Not multiple unit plants Criterion 3: Air clean up function is provided for TSC HVAC system only.	9.4.3

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 21of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.4.4 Turbine Area Ventilation System	1. For GDC 2, acceptance is based on the guidance of RG 1.29, Position C.1 for safety-related portions and Position C.2 for nonsafety-related portions. 2. For GDC 5, acceptance is based on the determination that the use of the TAVS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the remaining unit(s) 3. For GDC 60, acceptance is based on guidance of RGs 1.52 and 1.140 as related to design, inspection, testing, and maintenance criteria for post-accident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and adsorption units of light-water-cooled nuclear power plants. For RG 1.52 Revision 2, the applicable regulatory position is C.2. For RG 1.52 Revision 3, the applicable regulatory position is C.3. For RG 1.140 Revision 1, the applicable regulatory positions are C.1 and C.2. For RG 1.140 Revision 2, the applicable regulatory positions are C.2 and C.3.	Not applicable. Criterion 1: Turbine Building Area Ventilation System does not need design and manufacture considered SSE because the failure of Turbine Building Area Ventilation System has no influence on safety-related portion and the main control room comfort. Criterion 2: The standard design of US-APWR is single unit and will not be system sharing basically even in case of multiple-unit. Criterion 3: The filter system required is not installed in Turbine Building because Turbine Building has not possibility of contamination by radio-active particle.	9.4.4
9.4.5 Engineered Safety Feature Ventilation System	1. For GDC 2, acceptance is based on the guidance of RG 1.29, Position C.1, for safety-related portions and Position C.2 for nonsafety-related portions. 2. For GDC 4, acceptance is based on meeting the acceptance criteria in the following SRP sections, as they apply to the ESFVS: SRP Sections 3.5.1.1, 3.5.1.4, 3.5.2, and SRP Section 3.6.1. 3. For GDC 5, acceptance is based on the determination that the use of the ESFVS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the remaining unit(s). 4. For GDC 17, acceptance is based on the guidance of item 2 under Subsection A and item 1 under Subsection C of the NUREG-CR/0660 section “Recommendations” for protection of essential electrical components from failure due to the accumulation of dust and particulate materials.	Conform with exceptions Criterion 3: Not multiple unit plants. Criterion 4: Gas turbine has own cooling system. Criterion 5: Air cleanup function is provided for annulus exhaust system only.	9.4.5

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 22 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.4.5 Engineered Safety Feature Ventilation System (continued)	5. For GDC 60, acceptance is based on the guidance of RGs 1.52 and 1.140 as related to design, inspection, testing, and maintenance criteria for post-accident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and adsorption units of light-water-cooled nuclear power plants. For RG 1.52 rev 2, the applicable regulatory position is C.2. For RG 1.52 rev 3, the applicable regulatory position is C.3. For RG 1.140 rev 1, the applicable regulatory positions are C.1 and C.2. For RG 1.140 rev 2, the applicable regulatory positions are C.2 and C.3. 6. For 10CFR50.63, acceptance is based on the applicable guidance of RG 1.155, including Position C.3.2.4.		
9.5.1 Fire Protection Program	1. RG 1.174, Revision 1, “An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis,” as it applies to the use of PRA in support of changes to the fire protection licensing basis for nuclear power plants. Appropriate techniques for performing a Fire PRA are presented in NUREG/CR-6850 (EPRI TR-1011989), “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities.” 2. RG 1.188, Revision 1, “Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses,” as it applies to FPP considerations for license renewal such as equipment aging issues. This RG endorses the guidance in Nuclear Energy Institute (NEI) document, NEI 95-10, Revision 6, “Industry Guideline for Implementing the Requirements of 10CFRPart 54 – The License Renewal Rule.” 3. RG 1.189, Revision 1, “Fire Protection for Nuclear Power Plants,” which provides comprehensive staff positions and guidelines on fire protection for nuclear power plants. 4. RG 1.191, “Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown,” which establishes the fire protection objectives and staff positions for implementing fire protection for those nuclear power plants that have submitted the necessary certifications for license termination under 10CFRPart 50.82(a).	Conformance with exceptions. Some information of the Fire Protection Program such as the fire protection organization; administrative policies; maintenance, and QA is provided in COLA.	9.5.1

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 23 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.5.1 Fire Protection Program (continued)	5. Regulatory Guide 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition),” as it applies to the FPP of any new reactor COL application submitted in accordance with 10CFRPart 52. 6. Enhanced fire protection criteria for new reactor designs as documented in SECY 90-016, SECY 93-087, and SECY 94-084. SECY 90-016 established enhanced fire protection criteria for evolutionary light water reactors. SECY 93-087 recommended that the enhanced criteria be extended to include passive reactor designs. SECY 90 016 and SECY 93-087 were approved by the Commission in staff requirements memoranda (SRM). SECY 94-084, in part, establishes criteria defining safe-shutdown conditions for passive light water reactor designs. 7. For COL reviews, the description of the operational program and proposed implementation milestone(s) for the fire protection program are reviewed in accordance with 10 50.48. The operational program for fire protection should be fully implemented prior to fuel receipt at the plant site.		
9.5.2 Communications Systems	1. Information regarding the requirements of Appendix E to 10CFRPart 50, Part IV.E(9), will be found acceptable if adequate provisions are made and described for emergency facilities and equipment, including: at least one onsite and one offsite communications system; each system shall have a backup power source. 2. For those applicants subject to either 10CFR50.34(f) or the TMI Action Plan, information regarding the requirements of 10CFR50.34(f) (2)(xxv) and TMI Action Plan Item III A.1.2 will be found acceptable if provisions are made for an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a nearsite Emergency Operations Facility. 3. Information regarding the requirements of 10CFR50.47(a) (8) will be found acceptable if adequate emergency facilities and equipment to support the response are provided and maintained.	Conform with exceptions. Criteria #1, #2, #3, #9, #12, #13, and #14 refer to site-specific emergency response and security requirements that will be the responsibility of the COL Applicant. As indicated in section 9.5.2, details of the security communication system design and procedures are the responsibility of the COL Applicant.	9.5.2

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 24 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.5.2 Communications Systems (continued)	4. Information regarding the requirements of 10CFR50.55a will be found acceptable if SSCs are designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. 5. Information regarding the requirements of GDC 1 will be found acceptable if SSCs important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these SSCs will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of SSCs important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit. 6. Information regarding the requirements of GDC 2 will be found acceptable if SSCs important to safety are designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these SSCs shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 25 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.5.2 Communications Systems (continued)	7. Information regarding the requirements of GDC 3 will be found acceptable if SSCs important to safety are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on SSCs important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these SSCs. 8. Information regarding the requirements of GDC 4 will be found acceptable if SSCs important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. 9. Information regarding the requirements of GDC 19 will be found acceptable if equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls (I&C) to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 26 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.5.2 Communications Systems (continued)	10. Information regarding the requirements of 10CFR73.45(e)(2)(iii) will be found acceptable if communications subsystems and procedures are provided for notification of an attempted unauthorized or unconfirmed removal of strategic special nuclear material so that response can be such as to prevent the removal and satisfy the general performance objective and requirements of § 73.20(a). 11. Information regarding the requirements of 10CFR73.45(g)(4)(i) will be found acceptable if communications networks are provided to transmit rapid and accurate security information among onsite forces for routine security operation, assessment of a contingency, and response to a contingency. 12. Information regarding the requirements of 10CFR73.46(f) will be found acceptable if each guard, watchman, or armed response individual on duty shall be capable of maintaining continuous communication with an individual in each continuously manned alarm station required by 10CFR73.46(e)(5), who shall be capable of calling for assistance from other guards, watchmen, and armed response personnel and from law enforcement authorities; each alarm station required by 10CFR73.46(e)(5) shall have both conventional telephone service and radio or microwave transmitted two-way voice communication, either directly or through an intermediary, for the capability of communication with the law enforcement authorities; and non-portable communications equipment controlled by the licensee and required by 10CFR73.46(f) shall remain operable from independent power sources in the event of the loss of normal power.		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 27 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.5.2 Communications Systems (continued)	13. Information regarding the requirements of 10CFR73.55(e) will be found acceptable if all alarms required by 10CFR73.55 annunciate in a continuously manned central alarm station located within the protected area and in at least one other continuously manned station not necessarily onsite, so that a single act cannot remove the capability of calling for assistance or otherwise responding to an alarm. The onsite central alarm station must be considered a vital area and its walls, doors, ceiling, floor, and any windows in the walls and in the doors must be bullet-resisting. The onsite central alarm station must be located within a building in such a manner that the interior of the central alarm station is not visible from the perimeter of the protected area. This station must not contain any operational activities that would interfere with the execution of the alarm response function. Onsite secondary power supply systems for alarm annunciator equipment and non-portable communications equipment as required 10CFR73.55(f) of this section must be located within vital areas. All alarm devices including transmission lines to annunciators shall be tamper indicating and self-checking, e.g., an automatic indication is provided when failure of the alarm system or a component occurs, or when the system is on standby power. The annunciation of an alarm at the alarm stations shall indicate the type of alarm (e.g., intrusion alarms, emergency exit alarm, etc.) and location. All emergency exits in each protected area and each vital area shall be alarmed.		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 28 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.5.2 Communications Systems (continued)	14. Information regarding the requirements of 10CFR73.55(f) will be found acceptable if each guard, watchman or armed response individual on duty is capable of maintaining continuous communication with an individual in each continuously manned alarm station required by 10CFR73.55(e)(1), who shall be capable of calling for assistance from other guards, watchmen, and armed response personnel and from local law enforcement authorities. The alarm stations required by 10CFR73.55(e)(1) shall have conventional telephone service for communication with the law enforcement authorities as described in 10CFR73.55(f)(1). To provide the capability of continuous communication, radio or microwave transmitted two-way voice communication, either directly or through an intermediary, shall be established, in addition to conventional telephone service, between local law enforcement authorities and the facility and shall terminate in each continuously manned alarm station required by 10CFR73.55(e)(1). Non-portable communications equipment controlled by the licensee and required by 10CFR73.55 shall remain operable from independent power sources in the event of the loss of normal power.		
9.5.3 Lighting Systems	1. Acceptance criteria of the design of the normal and emergency lighting systems, as described in the applicant's safety analysis report (SAR), is based in part on the degree of similarity of the systems design with those for previously reviewed plants with satisfactory operating experience. 2. The normal lighting system(s) is acceptable if the integrated design of the system(s) will provide adequate station lighting in all areas, from power sources described in Section 8.2 of the SRP that are required for control and maintenance of equipment and plant access routes during normal plant operations.	Conformance with no exceptions identified	9.5.3

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 29 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.5.3 Lighting Systems (continued)	3. The emergency lighting system(s) is acceptable if the integrated design of the system(s) will provide adequate emergency station lighting in all areas, required for fire fighting, control and maintenance of equipment used for implementing safe shutdown of the plant during all plant operating conditions, and the access routes to and from these areas. 4. The lighting systems designs will be acceptable if they conform to the lighting levels recommended in NUREG-0700, which is based on the Illuminating Engineering Society of North America (IESNA) Lighting Handbook (Reference 2) as related to systems design and illumination levels recommended for industrial facilities.		
9.5.4 Emergency Diesel Engine Fuel Oil Storage and Transfer System	1. GDC 2 requirements for which SSCs must be protected from…. 2. GDC 4 requirements for which SSCs must be protected from…Comprehensive compliance with GDC 2 is reviewed under other SRP sections as specified in subsection I of this SRP section. 3. GDC 5 requirements for sharing of SSCs important to safety … 4. GDC 17 requirements for the capability of the cooling water system … 5. GDC 44 requirements are met when the EDECWS has… 6. GDC 45 as to design provisions for periodic inspection… 7. GDC 46 as to design provisions for appropriate functional testing…	Conformance with no exception identified. US-APWR has no diesel generators, but uses gas turbine generators for emergency power in the standard design.	9.5.4
9.5.5 Emergency Diesel Engine Cooling Water System	1. GDC 2 requirements for which SSCs must be protected from…. 2. GDC 4 requirements for which SSCs must be protected from…. 3. GDC 5 requirements for sharing of SSCs important to safety… 4. GDC 17 requirements for the capability of the cooling water system… 5. GDC 44 requirements are met when the EDECWS has… 6. GDC 45 as to design provisions for periodic inspection… 7. GDC 46 as to design provisions for appropriate functional testing…	Not applicable. Emergency power will be provided for US-APWR by gas turbine generators in lieu of diesel generators. The gas turbine generators have no functional equivalent of a cooling water system.	N/A

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 30 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.5.6 Emergency Diesel Engine Starting System	1. GDC 2 requirements for SSCs to withstand or be protected from… 2. GDC 4 requirements for SSCs to be protected against… 3. GDC 5 requirements for sharing of SSCs important to safety 4. GDC 17 as to the capability of the diesel engine air starting system…	Conformance with no exception identified. US-APWR has no diesel generators, but uses gas turbine generators for emergency power in the standard design.	9.5.6
9.5.7 Emergency Diesel Engine Lubrication System	1. GDC 2 requirements for SSCs to withstand or be protected… 2. GDC 4 requirements for SSCs to be protected against… 3. GDC 5 requirements for sharing of SSCs important to safety 4. GDC 17 requirements of independence and redundancy criteria…	Conformance with no exception identified. US-APWR has no diesel generators, but uses gas turbine generators for emergency power in the standard design..	9.5.7
9.5.8 Emergency Diesel Engine Combustion Air Intake and Exhaust System	1. GDC 2 requirements for SSCs to withstand or be protected from… 2. GDC 4 requirements of SSCs to be protected against 3. GDC 5 requirements for sharing of SSC important to safety… 4. GDC 17 as related to the capabilities of the diesel engine combustion…	Conformance with no exception identified. US-APWR has no diesel generators, but uses gas turbine generators for emergency power in the standard design.	9.5.8

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 1 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
10.2 Turbine Generator	1. Specific criteria necessary to meet the requirements of GDC 4 are as follows: A. A turbine control and overspeed protection system should control turbine action under all normal or abnormal operating conditions and should ensure that a full load turbine trip will not cause the turbine to overspeed beyond acceptable limits. Under these conditions, the control and protection system should permit an orderly reactor shutdown by use of either the turbine bypass system and main steam relief system or other engineered safety systems. The overspeed protection system should meet the single failure criterion and should be testable when the turbine is in operation. B. The turbine main steam stop and control valves and the reheat steam stop and intercept valves should protect the turbine from exceeding set speeds and should protect the reactor system from abnormal surges. The reheat stop and intercept valves should be capable of closure concurrent with the main steam stop valves, or of sequential closure within an appropriate time limit, to ensure that turbine overspeed is controlled within acceptable limits. The valve arrangements and valve closure times should be structured so that a failure of any single valve to close will not result in excessive turbine overspeed in the event of a TGS trip signal. C. The TGS should have the capability to permit periodic testing of components important to safety while the unit is operating at rated load.	Conformance with exceptions. Item “3” is not applicable because there is no safety-related equipment in this room.	10.2

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 2 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
10.2 Turbine Generator (continued)	2. An inservice inspection program for main steam and reheat valves should be established and should include the following provisions: A. At intervals of approximately 3-1/3 years, during refueling or maintenance shutdowns coinciding with the inservice inspection schedule required by Section XI of the American Society of Mechanical Engineers (ASME) Code for reactor components, at least one main steam stop valve, one main steam control valve, one reheat stop valve, and one reheat intercept valve should be dismantled, and visual and surface examinations should be conducted of valve seats, disks, and stems. If this process detects unacceptable flaws or excessive corrosion in a valve, all other valves of that type should be dismantled and inspected. Valve bushings should be inspected and cleaned and bore diameters should be checked for proper clearance. B. Main steam stop and control valves should be exercised at a frequency recommended by the turbine vendor or valve manufacturer. 3. The arrangement of connection joints between the low-pressure turbine exhaust and the main condenser should prevent adverse effects on any safety-related equipment in the turbine room in the event of a rupture (it is preferable not to locate safety-related equipment in the turbine room).		
10.2.3 Turbine Rotor Integrity	1. Materials Selection The turbine forged or welded rotor should be made from a material and by a process that tends to minimize flaw occurrence and maximize fracture toughness properties, such as a NiCrMoV alloy processed by vacuum melting or vacuum degassing. The material should be examined and tested to meet the following criteria: A. Chemical analysis should be performed for each forging. Elements that have a deleterious effect on toughness, such as sulfur and phosphorus, should be controlled to low levels.	Conformance with exceptions. Criterion “3C” is not applicable because no bore rotor is scheduled to be applied.	10.2.3

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 3 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
10.2.3 Turbine Rotor Integrity (continued)	B. The 50% fracture appearance transition temperature (FATT) as obtained from Charpy tests performed in accordance with specification ASTM A-370 should be no higher than -18EC (OEF) for low-pressure turbine rotors. The nil-ductility transition (NDT) temperature obtained in accordance with specification ASTM E-208 may be used in lieu of FATT. NDT temperatures should be no higher than -35EC (-30EF). C. The Charpy V-notch (Cv) energy at the minimum operating temperature of each low-pressure rotor in the tangential direction should be at least 8.3 kg-m (60 ft-lbs). A minimum of three Cv specimens should be tested in accordance with specification ASTM A-370. 2. Fracture Toughness The low-pressure turbine disk forged or welded rotor fracture toughness properties are acceptable if the following criteria are met. The ratio of the fracture toughness (KIc) of the rotor material to the maximum tangential stress at speeds from normal to design overspeed should be at least 10 /mm (2 /in), at minimum operating temperature. Bore stress calculations should include components due to centrifugal loads, interference fit, and thermal gradients. Sufficient warmup time should be specified in the turbine operating instructions to ensure that toughness will be adequate to prevent brittle fracture during startup. Fracture toughness properties can be obtained by any of the following methods: A. Testing of the actual material of the turbine rotor to establish the KIc value at normal operating temperature. B. Testing of the actual material of the turbine rotor with an instrumented Charpy machine and a fatigue precracked specimen to establish the KIc (dynamic) value at normal operating temperature. If this method is used, KIc (dynamic) shall be used in lieu of KIc (static) in meeting the toughness criteria above.		

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 4 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
10.2.3 Turbine Rotor Integrity (continued)	C. Estimating of KIc values at various temperatures from conventional Charpy and tensile data on the rotor material using methods are presented in J. A. Begley and W. A. Logsdon, Scientific Paper 71-1E7-AMSLRF-P1. This method of obtaining KIc should be used only on materials which exhibit a well-defined Charpy energy and fracture appearance transition curve and are strain-rate insensitive. The staff should review the test data and the calculated toughness curve submitted by the applicant. D. Estimating "lower bound" values of KIc at various temperatures using the equivalent energy concept developed by F. J. Witt and T. R. Mager, ORNL-TM- 3894. The staff should review the load-displacement data from the compact tension specimens and the calculated toughness data submitted by the applicant. 3. Pre-service Inspection The applicant's pre-service inspection program is acceptable if it meets the following criteria: A. Forged or welded rotors should be rough machined prior to heat treatment. B. Each finished forged or welded rotor should be subjected to 100% volumetric (ultrasonic), surface, and visual examinations using procedures and acceptance criteria equivalent to those specified for Class 1 components in the ASME Boiler and Pressure Vessel Code, Sections III and V. Before welding and/or brazing, all surfaces prepared for welding and/or brazing should be surface examined. After welding and/or brazing, all surfaces exposed to steam should be surface examined, giving particular attention to stress risers and welds. Welds should be ultrasonically examined in the radial and radial-tangential sound beam directions.		

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 5 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
10.2.3 Turbine Rotor Integrity (continued)	C. Finish machined bores, keyways, and drilled holes should be subjected to magnetic particle or liquid penetrant examination. No flaw indications in keyway or hole regions are allowed. D. Each turbine rotor assembly should be spin tested at 5% above the maximum speed anticipated during a turbine trip following loss of full load. 4. Turbine Rotor Design The turbine assembly should be designed to withstand normal conditions, anticipated transients, and accidents resulting in a turbine trip without loss of structural integrity. The design of the turbine assembly should meet the following criteria: A. The design overspeed of the turbine should be 5% above the highest anticipated speed resulting from a loss of load. The staff should review the basis for the assumed design overspeed. B. The combined stresses of low-pressure turbine rotor at design overspeed due to centrifugal forces, interference fit, and thermal gradients should not exceed 0.75 of the minimum specified yield strength of the material, or 0.75 of the measured yield strength in the weak direction of the materials if appropriate tensile tests have been performed on the actual rotor material. C. The turbine shaft bearings should be able to withstand any combination of the normal operating loads, anticipated transients, and accidents resulting in a turbine trip. D. The natural critical frequencies of the turbine shaft assemblies existing between zero speed and 20% overspeed should be controlled in the design and operation stages so as to cause no distress to the unit during operation.		

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 6 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
10.2.3 Turbine Rotor Integrity (continued)	E. The turbine rotor design should facilitate inservice inspection of all high stress regions, including bores and keyways, without the need for removing the disks from the shaft. 5. Inservice Inspection The applicant's inservice inspection program is acceptable if it meets the following criteria: The inservice inspection program for the steam turbine assembly should provide assurance that rotor flaws that might lead to brittle failure of a rotor at speeds up to design speed will be detected. The inservice inspection and maintenance program for the turbine assembly should comply with the manufacturer's recommendations. Inservice inspection and maintenance activities may be performed during plant shutdown coinciding with the inservice inspection schedule as required by ASME Boiler and Pressure Vessel Code, Section XI, and should include complete inspection of all significant turbine components, such as couplings, coupling bolts, turbine shafts, low-pressure turbine blades, low-pressure rotors, and high-pressure rotors. This inspection should consist of visual, surface, and volumetric examinations, as required by the code.		
10.3 Main Steam Supply System	1. Acceptance of GDC 2 is based on meeting the guidance of Regulatory Guide 1.29, Position C.1 for safety-related portions and Position C.2 for nonsafety-related portions. 2. Acceptance of GDC 4 is based on the guidance of Regulatory Guide 1.115, Position C.1, as it relates to the protection of SSCs important to safety from the effects of turbine missiles. In addition, the system design should adequately consider water (steam) hammer and relief valve discharge loads to assure that system safety functions can be performed and should assure that operating and maintenance procedures include adequate precautions to prevent water (steam) hammer and relief valve discharge loads. The system design should also include protection against water entrainment	Conformance with no exceptions identified.	10.3

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 7 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
10.3 Main Steam Supply System (continued)	3. Compliance with GDC 5 requires that SSCs important to safety shall not be shared by nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their intended safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. Meeting the requirements of GDC 5 provides assurance that the main steam system and its associated components will continue performing their required safety functions even if they are shared by multiple nuclear power units. 4. Acceptance of GDC 34 is based on the following: A. The positions in Branch Technical Position 5-4, as they relate to the design requirements for residual heat removal (RHR) B. Issue Number 1 of NUREG-0138, as it relates to credit being taken for all valves downstream of the main steam isolation valves (MSIVs) to limit blowdown of a second steam generator if a steamline were to break upstream of the MSIV 5. Acceptance of 10CFR50.63 is based on meeting Regulatory Guide 1.155 as it relates to the MSSS design. 6. Regulatory Guide 1.29, Positions C.1.a, C.1.e, C.1.f, C.2 and C.3, as it relates to the seismic design classification of system components. 7. Regulatory Guide 1.117, Appendix Position 2 and 4, as it relates to the protection of SSCs important to safety from the effects of tornado missiles.		

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 8 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
10.3 Main Steam Supply System (continued)	8. SECY 93-087, as it applies to BWR plants that do not incorporate an MSIVLCS and for which main steamline fission product holdup and retention are credited in the analysis of design-basis accident radiological consequences as follows: A. Seismic Category I is the classification for the main steamlines extending from the outermost containment isolation valve to the seismic interface restraint and connected piping up to the first normally closed valve. B. The nonseismic Category I classification can apply to the main steamlines from the seismic interface restraint up to, but not including, the turbine stop valve (including connected piping to the first normally closed valve) if the following criteria are met: i. A dynamic seismic analysis method analyzed the lines to demonstrate their structural integrity under SSE loading conditions. ii. All pertinent quality assurance requirements of Appendix B to 10CFRPart 50 are applied. iii. For lines used as an MSIV leakage path to the condenser, reliable power sources must be available for control and isolation valves so that a control operator can establish the flowpath, assuming a single act C. Main steamlines and other main steam system components are assigned a quality group classification in accordance with the criteria of Branch Technical Position 3-1.		

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 9 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
10.3.6 Steam and Feedwater System Materials	1. Materials Selection and Fabrication of Class 2 and 3 Components A. The materials specified for use in Class 2 and 3 components should conform to Appendix I to Section III of the Code and to Parts A, B, and C of Section II of the Code. B. Regulatory Guide 1.84, describes acceptable Code Cases that may be used in conjunction with the above specifications. Appendix IV to Section III of the Code provides requirements for approval of new materials. C. Regulatory Guide 1.71 provides the following guidelines for assuring the integrity of welds in locations of restricted direct physical and visual accessibility. i. The performance qualification should require testing of the welder under simulated conditions when conditions of accessibility to production welds are less than 30 to 35 cm (12 to 14 inches) in any direction from the joint. ii. Requalification should be required for significantly different restricted accessibility conditions or when any essential welding variables listed in Code Section IX are changed. D. Regulatory Guide 1.50 provides methods to control preheat temperatures for welding low alloy steel. For carbon steel and low alloy steel materials, Section III, Appendix D, Article D-1000 of the ASME Code specifies preheat temperatures. E. Regulatory Guide 1.37 and ANSI Standard N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," describe acceptable procedures for cleaning and handling Class 2 and 3 components of the steam and feedwater systems. F. Acceptance criteria for nondestructive examination of tubular products are provided in the relevant paragraphs of Subsections NC and ND of Section III of the ASME Code.	Conformance with no exceptions identified.	10.3.6

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 10 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
10.3.6 Steam and Feedwater System Materials (continued)	2. Fracture Toughness of Class 2 and 3 Components The fracture toughness properties of the ferritic materials of these components should meet the following requirements of the editions and addenda of Section III of the Code, as specified in 10CFR50.55a: A. NC-2300, "Fracture Toughness Requirements for Material" (Class 2) B. ND-2300, "Fracture Toughness Requirements for Material" (Class 3)		
10.4.1 Main Condensers	1. The requirements of GDC 60 are met when the MC design includes provisions to prevent excessive releases of radioactivity to the environment which may result from a failure of a structure, system or component in the MC. Acceptance is based on meeting the following: A. SECY 93-087 gives guidance for new BWR plants that do not incorporate an MSIVLCS and for which MC holdup and plateout of fission products is credited in the analysis of design basis accident radiological consequence. It states that seismic analyses are to be performed to ensure that the condenser anchorages and the piping inlet nozzle to the condenser are capable of maintaining their structural integrity during and after an SSE. B. If there is a potential for explosive mixtures to exist, the MC is designed to withstand the effects of an explosion and instrumentation is provided to detect and annunciate the buildup of potentially explosive mixtures, dual instrumentation is provided to detect, annunciate, and effect control measures to prevent the buildup of potentially explosive mixtures, as outlined in SRP Section 11.3, subsection II, “Acceptance Criteria,” SRP Acceptance Criteria, Item 6.	Not applicable. Criterion 1.A: This criterion is applicable for BWR. Criterion 1.B: No hydrogen buildup is anticipated for US-APWR.	10.4.1

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 11 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
10.4.2 Main Condenser Evacuation System	1. The requirements of General Design Criteria 60 (GDC 60) are met when the MCES design includes provisions to prevent excessive releases of radioactivity to the environment which may result from a failure of a structure, system or component in the MC. Acceptance is based on meeting the following: A. If there is a potential for explosive mixtures to exist, the MCES is designed to withstand the effects of an explosion and instrumentation is provided to detect and annunciate the buildup of potentially explosive mixtures, dual instrumentation is provided to detect, annunciate, and effect control measures to prevent the buildup of potentially explosive mixtures, as outlined in SRP Section 11.3, subsection II, "Acceptance Criteria," SRP Acceptance Criteria, Item 6. Such a potential does not exist on systems designed to maintain the steam content above 58% by volume in hydrogen-air mixtures or nitrogen content above 92% by volume in hydrogen-oxygen mixtures in all MCES components. The design pressure and normal operational absolute pressure should be provided for MCES components containing potentially explosive mixtures.	Not applicable. There is no potential for explosive mixtures to exit in US-APWR.	10.4.2
10.4.3 Turbine Gland Sealing System	GDC 60 requires the TGSS to be designed to provide for the collection and condensation of sealing steam and the venting and treatment of noncondensables. Additional acceptance criteria and review procedures are contained in the SRP sections referenced in the "review interfaces" section of this SRP. There is no specific acceptance criteria associated with this SRP section.	Conformance with no exceptions identified.	10.4.3

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 12 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
10.4.4 Turbine Bypass System	1. Piping Failures The requirements of GDC 4 related to the ability of SSCs important to safety to meet environmental conditions associated with normal operation, maintenance, testing, and postulated accident conditions is met by demonstrating that failure of the TBS due to a pipe break or malfunction of the TBS will not adversely affect essential systems or components (i.e., those necessary for safe shutdown or accident prevention or mitigation). 2. Residual Heat Removal The requirements of GDC 34 related to providing a reliable system that removes residual heat during normal plant shutdown is met by demonstrating the ability to use the turbine bypass system for shutting down the plant during normal operations. The operation of the TBS eliminates the need to rely solely on safety systems, which are required to meet the redundancy and power source requirements of this criterion. 3. MSIV Alternate Leakage Path (ALP) For BWR plants that do not incorporate an MSIVLCS and for which TBS holdup and plateout of fission products is credited in the analysis of design basis accident radiological consequences, guidance from SECY 93-087 is applicable. Specifically, the turbine bypass lines from the first valve up to the condenser inlet do not need to be classified as seismic category I if the following criteria are met: A. They have been analyzed using a dynamic seismic analysis method to demonstrate their structural integrity under SSE loading conditions. B. All pertinent QA requirements of Appendix B to 10CFRPart 50 are applied.	Conformance with exceptions. Criterion #3 applies to BWRs only.	10.4.4

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 13 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
10.4.4 Turbine Bypass System (continued)	C. For lines utilized as an MSIV leakage path to the condenser, reliable power sources must be available for control and isolation valves so that a control operator can establish the flow path assuming a single active failure. In addition, the TBS lines and other components utilized as an MSIV leakage path to the condenser are assigned a quality group classification in accordance with the criteria of Branch Technical Position 3-1.		
10.4.5 Circulating Water System	1. The requirements of GDC 4 are met when the circulating water system design includes provisions to accommodate the effects of discharging water that may result from a failure of a component or piping in the CWS. Acceptance is based on meeting the following: A. Means should be provided to prevent or detect and control flooding of safety-related areas so that the intended safety function of a system or component will not be precluded due to leakage from the CWS. B. Malfunction or a failure of a component or piping of the CWS, including an expansion joint, should not have unacceptable adverse effects on the functional performance capabilities of safety-related systems or components.	Conformance with no exceptions identified.	10.4.5
10.4.6 Condensate Cleanup System	1. For direct cycle (boiling-water reactor (BWR)) plants, SRP Section 5.4.8 provides the criteria for acceptable water purity. SRP Section 5.4.8 refers to the guidelines provided in the latest version in the Electric Power Research Institute (EPRI) report series, “BWR Water Chemistry Guidelines,” and the technical specifications for the water chemistry of BWR reactor coolant systems. (This criterion is specific to boiling water reactors (BWRs), and is not applicable to US-APWR.) 2. For indirect cycle (pressurized-water reactor (PWR)) plants, SRP Section 5.4.2.1 provides the criteria for acceptable secondary water chemistry. SRP Section 5.4.2.1 refers to the guidelines provided in the latest version in the EPRI report series, “PWR Secondary Water Chemistry Guidelines.”	Conformance with no exceptions identified.	10.3.5

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 14 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
10.4.7 Condensate and Feedwater System	1. Seismic Events The requirements of GDC 2 are met by demonstrating that SSCs important to safety will be designed to withstand the effects of natural phenomena such as earthquakes. Acceptance is based on meeting the guidance of Regulatory Guide 1.29, Position C.1 for safety-related portions and Position C.2 for nonsafety-related portions. 2. Fluid Instabilities The requirements of GDC 4 as related to protecting SSCs against the dynamic effects associated with possible fluid flow instabilities (e.g., water hammers) during normal plant operation as well as during upset or accident conditions are met by: A. Meeting the guidance contained in the Branch Technical Position 10-2, "Design Guidelines for Avoiding Water Hammers in Steam Generators," for reducing the potential for water hammers in steam generators; and B. Meeting the guidance related to feedwater-control-induced water hammer. Guidance for water hammer prevention and mitigation is found in NUREG-0927, Revision 1. 3. Sharing of SSCs The requirements of GDC 5 are met by demonstrating the capability of important to safety components in the CFS which are shared by multiple units to perform their required safety functions. 4. Heat Removal Capability The requirements of GDC 44, as related to the capability to transfer heat from SSCs important to safety to an ultimate heat sink are met by demonstrating that the CFS is capable of providing heat removal under both normal operating and accident conditions. Sufficient redundancy of components is demonstrated so that under accident conditions the safety function can be performed assuming a single active component failure (which may be coincident with the loss of offsite power for certain events.) The system demonstrates capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained.	Conformance with exceptions. Criterion 7 is defined as COL item in DCD subsection 10.3.6. Criterion 8 is for BWR.	10.4.7

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 15 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
10.4.7 Condensate and Feedwater System (continued)	5. Inspection The requirements of GDC 45 are met by demonstrating that the design contains provisions to permit periodic inservice inspection of system components and equipment. 6. Testing The requirements of GDC 46 are met by demonstrating that the design contains provisions to permit appropriate functional testing of the system and components to ensure structural integrity and leak-tightness, operability and performance of active components, and capability of the integrated system to function as intended during normal, shutdown, and accident conditions. 7. Flow Accelerated Corrosion Piping system designs, including material standards and inspection programs, shall incorporate adequate considerations to avoid erosion and corrosion. Guidance for acceptable inspection programs is found in Generic Letter 89-08 and in EPRI NP-3944, "Erosion/Corrosion in Nuclear Plant Steam Piping: Causes and Inspection Guidelines." 8. Feedwater Nozzle Design For BWRs, feedwater nozzle design, inspection, and testing procedures, and CFS operating procedures are adequate to minimize nozzle cracking at low feedwater flow. The review criteria for this issue are stated in NUREG-0619 and in associated Generic Letters 80-95 and 81-11.		
10.4.8 Steam Generator Blowdown System	1. The requirements of GDC 1 and GDC 2 are met when the design of the SGBS includes the following: A. The design is seismic Category I and Quality Group B, from its connection to the steam generator inside primary containment up to and including the first isolation valve outside containment. B. The design is in accordance with the provisions of Regulatory Guide 1.143, Position C.1.1 downstream of the outer containment isolation valves.	Conformance with no exceptions identified. Note: As for R.G1.143 mentioned in criteria 2, SGBDS is not designed as Radioactive waste management system. (The potion of outer containment valve excluding itself is designed as class 4.)	10.4.8

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 16 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
10.4.8 Steam Generator Blowdown System (continued)	2. The requirements of GDC 13 are met when the SGBS design includes provisions to monitor system parameters and maintain them within a range that allows the system to perform its impurity removal function and thereby assist in maintaining the integrity of the reactor coolant pressure boundary. 3. The requirements of GDC 14 are met when the SGBS design includes provisions to control secondary water chemistry to maintain the integrity of the primary coolant boundary. Acceptance is based on meeting the following: A. The SGBS is sized to accommodate the design blowdown flow needed to maintain secondary coolant chemistry for normal operation, including anticipated operational occurrences. B. Equipment capacities are based on design blowdown flow rates and are such that temperature limits for heat-sensitive processes are not exceeded.		
10.4.9 Auxiliary Feedwater System (PWR)	1. Acceptance for meeting the relevant aspects of GDC 2 is based in part on meeting the guidance of Position C.1 of Regulatory Guide 1.29 if any portion of the system is deemed to be safety related and the guidance of Position C.2 for nonsafety-related portions. Also, acceptance is based in part on (1) meeting the guidance of Regulatory Guide 1.117 with respect to identifying portions of the system that should be protected from tornadoes and (2) meeting the guidance of Regulatory Guide 1.102 with respect to identifying portions of the system that should be protected from flooding. 2. Acceptance for meeting the relevant aspects of GDC 4 is based on identification of essential portions of the system as protected from dynamic effects including internal and external missiles. In part, this information should be consistent with the guidance of Regulatory Guide 1.117 with respect to identifying portions of the system that should be protected from tornado missiles and the guidance of BTP 3-3 with respect to identifying portions of the system that should be protected from the dynamic effects of pipe breaks.	Conformance with exceptions. Criterion #3 refers to shared systems and US-APWR is a single unit design.	10.4.9

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 17 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
10.4.9 Auxiliary Feedwater System (PWR) (continued)	3. Acceptance of GDC 5 is based on provision of information that addresses the capability of shared portions of the AFW system to perform required safety functions during an accident in one unit such that the capability to conduct a safe and orderly shutdown and cool-down in the unaffected unit(s) is not significantly affected. 4. Acceptance of GDC 19 is based on meeting BTP 5-4 with regards to cold shutdown from the control room using only safety grade equipment. 5. Acceptance of GDC 34 and 44 is based on the system having sufficient flow capacity so that the system can remove residual heat over the entire range of reactor operation and cool the plant to the decay heat removal system cut-in temperature and the system design conforming to the guidance of BTP 10-1 as it relates to AFW pump drive and power supply diversity. In addition, the recommendations of NUREG-0611 and NUREG-0635 shall also be met. TMI Action Plan item II.E.1.1 of NUREG 0737 and 10CFR50.34(f)(1)(ii) for applicants subject to 10CFR50.34(f) require an AFWS reliability analysis. An acceptable AFWS should have an unreliability in the range of 10-4 to 10-5 per demand exclusive of station blackout scenarios. Compensating factors (e.g., other methods of accomplishing AFWS safety functions of the AFWS or other reliable methods for cooling the reactor core during abnormal conditions) may be considered to justify a larger AFWS unavailability. 6. Acceptance of GDC 45 is based on provision of information describing how the design of the AFW system permits inservice inspection of safety-related components and equipment. 7. Acceptance of GDC 46 is based on provision of information describing how the design of the AFW system, including instrumentation, permits periodic operational functional testing of safety-related components and equipment.		

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 18 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
10.4.9 Auxiliary Feedwater System (PWR) (continued)	8. Acceptance of 10CFR50.62 is based on design provisions for automatic initiation of the AFW system in an ATWS. 9. Acceptance of 10CFR50.63 is based on conformance with the guidance of RG 1.155 as related to the AFWS design.		
Branch Technical Position 10-1: Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants	1. The AFWS should have at least two full-capacity, independent systems with diverse power sources. 2. Other AFWS powered components also should have separate and multiple sources of motive energy (e.g., two separate auxiliary feedwater trains, each capable of removing the reactor system after-heat load, one separate train powered from either of two ac/AC sources and the other powered wholly by steam and direct current electric power). 3. The piping arrangements, both intake and discharge, for each train should be designed for the pumps to supply feedwater to any combination of steam generators. This arrangement should be designed for pipe failure, active component failure, power supply failure, or control system failure that could prevent system function. One acceptable arrangement is crossover piping with valves operable by remote manual control from the control room applying the power diversity principle to the valve operators and actuation systems. 4. The AFWS design should have suitable redundancy to offset the consequences of any single-active component failure; however, each train need not have redundant active components. 5. For a high-energy line break, the system should be arranged to assure the capability to supply necessary emergency feedwater to the steam generators despite the postulated rupture of any high-energy section of the system, assuming a concurrent, single, active failure.	Conformance with no exceptions identified.	10.4.9

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 19 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 10-2: Design Guidelines for Avoiding Water Hammers in Steam Generators	Top-Feed Steam Generator Designs 1. Prevent or delay water draining from the feed ring following a drop in steam generator water level by means such as top discharge J-Tubes and limiting feed ring seal assembly leakage. 2. Minimize the volume of feedwater piping external to the steam generator which could pocket steam using the shortest possible (less than 2.1 m (7 ft)) horizontal run of inlet piping to the steam generator feed ring. 3. Perform tests acceptable to NRC to verify that unacceptable feedwater hammer will not occur using the plant operating procedures for normal and emergency restoration of steam generator water level following loss of normal feedwater and possible draining of the feed ring. Provide the procedures for these tests for approval before conducting the tests and submit the results from such tests. 4. Implement pipe refill flow limits where practical. Preheat Steam Generator Designs 1. Minimize the horizontal lengths of feedwater piping between the steam generator and the vertical run of piping by providing downward turning elbows immediately upstream of the main and auxiliary feedwater nozzles. 2. Provide a check valve upstream of the auxiliary feedwater connection to the top feedwater line. 3. Maintain the top feedwater line full at all times.	Conformance with exception. Criteria relating to “Preheat Steam Generator Designs” and “Once Through Steam Generator Designs” are not applicable to US-APWR.	5.4.2.1, 10.4.7, 14.2.12

Table 1.9.2-10 US-APWR Conformance with Standard Review Plan Chapter 10 Steam and Power Conversion System (sheet 20 of 20)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 10-2: Design Guidelines for Avoiding Water Hammers in Steam Generators (continued)	4. Perform tests acceptable to NRC to verify that unacceptable feedwater hammer will not occur using plant operating procedures for normal and emergency restoration of steam generator water level following loss of normal feedwater. Also perform a water hammer test at the power level at which feedwater flow is transferred from the auxiliary feedwater nozzle to the main feedwater nozzle. The test shall be performed by pumping feedwater through the auxiliary feedwater (top) nozzle at the lowest feedwater temperature that the plant standard operating procedure (SOP) allows and then switching the feedwater at that temperature from the auxiliary feedwater nozzle to the main feedwater (bottom) nozzle by following the SOP. Submit the results of such tests. Once Through Steam Generator (OTSG) Designs 1. Provide auxiliary feedwater to the steam generator through an externally mounted supply top discharge header. 2. Perform tests acceptable to NRC to verify that unacceptable feedwater hammer will not occur using the plant operating procedures for normal and emergency restoration of steam generator water level following loss of normal feedwater. Provide the procedures for these tests for approval before conducting the tests, and submit the results of such tests.		

Table 1.9.2-11 US-APWR Conformance with Standard Review Plan Chapter 11 Radioactive Waste Management (sheet 1 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
11.1 Source Terms	1. All normal and potential sources of radioactive effluent delineated above in Subsection I will be considered. 2. For each source of liquid and gaseous waste considered above in Subsection I.1, the volumes and concentrations of radioactive material given for normal operation and anticipated operational occurrences should be consistent with those given in NUREG-0016 or NUREG-0017. 3. Decontamination factors for inplant control measures used to reduce gaseous effluent releases to the environment, such as iodine removal systems and high-efficiency particulate air (HEPA) filters for building ventilation exhaust systems and containment internal cleanup systems should be consistent with those given in Regulatory Guide 1.140. The building mixing efficiency for containment internal cleanup should be consistent with NUREG-0017. 4. Decontamination factors for inplant control measures used to reduce liquid effluent releases to the environment, such as filters, demineralizers and evaporators, should be consistent with those given in NUREG-0016 or NUREG-0017. 5. Radwaste augments used in the calculation of effluent releases to the environment are consistent with the findings of a cost-benefit analysis, which may be performed using the guidance of Regulatory Guide 1.110. The provisions that require a cost-benefit analysis are stated in Section II.D of Appendix I to 10CFRPart 50. 6. Effluent concentration limits at the boundary of the unrestricted area do not exceed the values specified in Table 2 of Appendix B to 10CFRPart 20. 7. The source terms result in meeting the design objectives for doses in unrestricted areas as set forth in Appendix I to 10CFRPart 50.	Conformance with exceptions. Per expectation expressed in the SRP, cost-benefit analysis, site specific effluent calculation and dose calculation will be contained in the COLA.	11.1

Table 1.9.2-11 US-APWR Conformance with Standard Review Plan Chapter 11 Radioactive Waste Management (sheet 2 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
11.1 Source Terms (continued)	8. For evaluating the source terms, the applicant should provide the relevant information in the SAR as required by 10CFR50.34, and 10CFR50.34a. This technical information should include all the basic data listed in Appendix A (BWRs) and Appendix B (PWRs) to Regulatory Guide 1.112 in order to calculate the releases of radioactive material in liquid and gaseous effluents (the source terms). An acceptable method for satisfying the criteria given in items 1 through 5 consists of using the Gaseous and Liquid Effluent (GALE) Computer Code and the source term parameters given in NUREG-0016 or NUREG-0017 for BWRs and PWRs, respectively. Complete listings of the GALE Computer Codes for BWRs and PWRs are given in NUREG-0016 and NUREG-0017, respectively. 9. If the applicant's calculational technique or any source term parameter differs from that given in ANSI/ANS 18.1-1999, NUREG-0016, or NUREG-0017, they should be described in detail and the bases for the methods and/or parameters used should be provided.		
11.2 Liquid Waste Management System	1. The LWMS should have the capability to meet the dose design objectives and include provisions to treat liquid radioactive wastes such that the following is true: A. The calculated annual total quantity of all radioactive materials released from each reactor at the site to unrestricted areas will not result in an estimated annual dose or dose commitment from liquid effluents for any individual in an unrestricted area from all pathways of exposure in excess of 0.03 millisievert (mSv) (3 millirem (mrem)) to the total body or 0.1 mSv (10 mrem) to any organ. Regulatory Guides 1.109, 1.112, and 1.113 provide acceptable methods for performing this analysis. B. In addition to 1.A, the LWMS should include all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return for a favorable cost-benefit ratio, can effect reductions in doses to the population reasonably expected to be within 80 kilometers (km) (50 miles [mi]) of the reactor. Regulatory Guide 1.110 provides an acceptable method for performing this analysis.	Conformance with exceptions. Criterion 1:Cost-benefit analysis, site specific effluent calculation and dose calculation will be contained in the COLA Criterion 6: This applies to an ESP application.	11.2

Table 1.9.2-11 US-APWR Conformance with Standard Review Plan Chapter 11 Radioactive Waste Management (sheet 3 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
11.2 Liquid Waste Management System (continued)	C. The concentrations of radioactive materials in liquid effluents released to unrestricted areas should not exceed the concentration limits in Table 2, Column 2, of Appendix B, to 10CFRPart 20. 2. The LWMS should be designed to meet the anticipated processing requirements of the plant. Adequate capacity should be provided to process liquid wastes during periods when major processing equipment may be down for maintenance (single failures) and during periods of excessive waste generation. Systems that have adequate capacity to process the anticipated wastes and that are capable of operating within the design objectives during normal operation, including anticipated operational occurrences, are acceptable. To meet these processing demands, interconnections between subsystems, redundant equipment, mobile equipment, and reserve storage capacity will be considered. 3. The seismic design of structures housing LWMS components, the quality group classification of liquid radwaste treatment equipment, and provisions to prevent and collect spills from indoor and outdoor storage tanks should conform to the guidelines of Regulatory Guide 1.143 for liquids and liquid wastes produced during normal operation and anticipated operational occurrences. For the purpose of this SRP, the dose limit cited in Section 5 of Regulatory Guide 1.43, addressing unmitigated releases of radioactive materials, is revised to be consistent with that of 10CFRPart 20.1301. The annual dose limit of Part 20.1301 is 100 mrem for members of the public located in unrestricted areas. 4. System designs should contain provisions to control leakage and facilitate operation and maintenance in accordance with the guidelines of Regulatory Guide 1.143 and industry standards cited in this regulatory guide for liquids and liquid wastes produced during normal operation and anticipated operational occurrences.		

Table 1.9.2-11 US-APWR Conformance with Standard Review Plan Chapter 11 Radioactive Waste Management (sheet 4 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
11.2 Liquid Waste Management System (continued)	5. System designs should describe features that will minimize, to the extent practicable, contamination of the facility and environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste, in accordance with the guidelines of Regulatory Guide 1.143, for liquids and liquid wastes produced during normal operation and anticipated operational occurrences, and the requirements of 10CFR20.1406, or the DC application, update in the SAR, or the COL application, to the extent not addressed in a referenced certified design. 6. For an ESP application, the dose estimates to a hypothetical maximally exposed member of the public from liquid effluents using radiological exposure models are developed based on Regulatory Guides 1.109, 1.111, and 1.113, and appropriate computer codes, such as the LADTAP II computer code (NUREG/CR-4013) for liquid effluents.		
11.3 Gaseous Waste Management System	1. The GWMS should have the capability to meet the dose design objectives and should include provisions to treat gaseous radioactive wastes such that the following is true: *(Additional text follows on requirements)* 2. The GWMS should be designed to meet the anticipated processing requirements of the plant. Adequate capacity should be provided to process gaseous wastes during periods when major processing equipment may be down for maintenance (single failures) and during periods of excessive waste generation. Systems that have adequate capacity to process the anticipated wastes and that are capable of operating within the design objectives during normal operation, including anticipated operational occurrences, are acceptable. To meet these processing demands, the reviewer will consider shared systems, redundant equipment, mobile equipment, and reserve storage capacity.	Conformance with exceptions. Criterion 5: The gaseous radwaste system charcoal delay bed function differs from the scope of Regulatory Guide 1.140. The function is to delay the release of noble gases, not to remove iodine. Criterion 6.A: The US-APWR GRS is not designed to withstand a hydrogen explosion. Criterion 8:This applies to an ESP application)	11.3

Table 1.9.2-11 US-APWR Conformance with Standard Review Plan Chapter 11 Radioactive Waste Management (sheet 5 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
11.3 Gaseous Waste Management System (continued)	3. The seismic design and quality group classification of components used in the GWMS and structures housing the system should conform to Regulatory Guide 1.143. The design should include precautions to stop continuous leakage paths (i.e., to provide liquid seals downstream of rupture discs) and to prevent permanent loss of the liquid seals in the event of an explosion due to gaseous wastes produced during normal operation and anticipated operational occurrences. 4. System designs should describe features that will minimize, to the extent practicable, contamination of the facility and environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste in accordance with Regulatory Guide 1.143, for gaseous wastes produced during normal operation and anticipated operational occurrences, and the requirements of 10CFR20.1406 or the DC application, update in the SAR, or the COL application to the extent not addressed in a referenced certified design. 5. System designs should use the guidelines in Regulatory Guide 1.140 for the design testing and maintenance of HEPA filters and charcoal absorbers installed in normal ventilation exhaust systems. If decontamination factors for radioiodines that differ from those specified in Regulatory Guide 1.140 are used for design purposes, they should be supported by test data under operating or simulated operating conditions (temperature, pressure, humidity, expected iodine concentrations, and flow rate). The test data should also support the effects of aging and poisoning by airborne contaminants. 6. If the potential for explosive mixtures of hydrogen and oxygen exists, the GRS portion of the GWMS should either be designed to withstand the effects of a hydrogen explosion or be provided with dual gas analyzers with automatic control functions to preclude the formation or buildup of explosive mixtures. The GRS is normally the only portion of the system that is vulnerable to potential hydrogen explosion. *(Additional text follows on requirements)*		

Table 1.9.2-11 US-APWR Conformance with Standard Review Plan Chapter 11 Radioactive Waste Management (sheet 6 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
11.3 Gaseous Waste Management System (continued)	7. Branch Technical Position (BTP) 11-5, as it relates to potential releases of radioactive materials (noble gases) as a result of postulated leakage or failure of a waste gas storage tank or offgas charcoal delay bed. 8. For an ESP application, the dose estimates to a hypothetical maximally exposed member of the public from gaseous effluents using radiological exposure models are developed based on Regulatory Guides 1.109 and 1.111, and appropriate computer codes, such as the GASPAR II computer code (NUREG/CR-4653) for gaseous effluents.		
11.4 Solid Waste Management System	1. The SWMS design parameters are based on expected radionuclide distributions and concentrations consistent with reactor operating experience for similar designs, as evaluated under SRP Section 11.1. 2. Processing equipment is sized to handle the design SWMS inputs, that is, the types of liquid, wet, and solid wastes; radionuclide distributions and concentrations; radionuclide removal efficiencies and decontamination factors; waste volume reduction and increase factors; waste volumes; and waste generation rates. 3. All liquid and wet wastes will be stabilized in accordance with a PCP before offsite shipment, or provisions will be made to verify the absence of free liquid in each container and procedures to reprocess containers in which free liquid is detected in accordance with the requirements of Branch Technical Position (BTP) 11-3. 4. Other forms of wet wastes will be stabilized or dewatered (subject to the licensed disposal facility's waste acceptance criteria) in accordance with a PCP, or provisions will be made to verify the absence of free liquid in each container and procedures to reprocess containers in which excess water is detected in accordance with the requirements of BTP 11-3. 5. SWMS design objectives, design criteria, treatment methods, expected effluent releases, process and effluent radiation monitoring and control instrumentation, and methods for establishing process and effluent instrumentation control set points, as they relate to the PCP and ODCM under this SRP Section and SRP Section 11.5.	Conformance with no exceptions identified.	11.4

Table 1.9.2-11 US-APWR Conformance with Standard Review Plan Chapter 11 Radioactive Waste Management (sheet 7 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
11.4 Solid Waste Management System (continued)	6. Waste containers, shipping casks, and methods of packaging wastes meet all applicable Federal regulations (e.g., 10CFRPart 71, addressing the packaging and transportation of radioactive materials; 10CFR20.2006 and Appendix G to 10CFRPart 20, addressing the transfer and manifesting of radioactive waste shipments; and 49CFRParts 171–180, addressing U.S. Department of Transportation (DOT) regulations for the shipment of radioactive materials); and 10CFRPart 61 or corresponding State regulations addressing applicable waste acceptance criteria of the disposal facility or waste processors. 7. Onsite waste storage facilities provide sufficient storage capacity to allow time for shorter lived radionuclides to decay before shipping in accordance with the requirements of BTP 11-3. The SAR should give the bases for determining the duration of the storage. 8. SWMS components and piping systems, as well as structures housing SWMS components, are designed in accordance with the provisions of Regulatory Guide 1.143, as it relates to the seismic design and quality group classification of components, and BTP 11-3 for wastes produced during normal operation and anticipated operational occurrences. 9. The SWMS contains provisions to reduce leakage and facilitate operations and maintenance in accordance with the provisions of Regulatory Guide 1.143 and BTP 11-3, as they relate to wastes produced during normal operation and anticipated operational occurrences. 10. For long-term onsite storage (e.g., for several years, but within the operational life of the plant), the storage facility should be designed to the guidelines of Appendix 11.4-A to this SRP section, including updated guidance from SECY 93-323 and SECY 94-198. 11. Liquid, wet, and dry solid wastes will be processed and disposed of in accordance with 10CFR61.55 and 10CFR61.56 requirements for waste classification and characteristics and with the waste acceptance criteria of the chosen licensed radioactive waste disposal site. The PCP should present the process and methods used to meet these 10CFRPart 61 requirements.		

Table 1.9.2-11 US-APWR Conformance with Standard Review Plan Chapter 11 Radioactive Waste Management (sheet 8 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
11.4 Solid Waste Management System (continued)	12. Mixed wastes (characterized by the presence of hazardous chemicals and radioactive materials) will be processed and disposed in accordance with 10CFR20.2007, as it relates to compliance with other applicable Federal, State, and local regulations governing any other toxic or hazardous properties of radioactive wastes. 13. All effluent releases (gaseous and liquid) associated with the operation (normal and anticipated operational occurrences) of the SWMS will comply with 10CFRPart 20 and Regulatory Guide 1.143, as they relate to the definition of the boundary of the SWMS beginning at the interface from plant systems, including multiunit stations, to the points of controlled liquid and gaseous effluent discharges to the environment or designated onsite storage locations, as defined in the PCP and ODCM. 14. Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestone for the PCP aspect of the Process and Effluent Monitoring and Sampling Program are reviewed in accordance with 10CFR20.1301 and 20.13.2, 10CFR50.34a, 10CFR50.36a, and 10CFR50, Appendix I, section II and IV. Its implementation is required by a license condition.		

Table 1.9.2-11 US-APWR Conformance with Standard Review Plan Chapter 11 Radioactive Waste Management (sheet 9 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
11.5 Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	1. Provisions should be made for the installation of instrumentation and monitoring equipment and/or sampling and analyses of all normal and potential effluent pathways for release of radioactive materials to the environment, including nonradioactive systems that could become radioactive through interfaces with radioactive systems. For GDC 64 and the requirements specified in 10CFR50.34(f)(2)(xvii) and 10CFR50.34(f)(2)(xxvii), the system designs should meet the provisions of Regulatory Guide 1.21 (Position C and Appendix A), Regulatory Guide 1.97 (Position C and Table 1 or 2, as applicable), Regulatory Guide 4.15 (Position C), and Appendix A to Regulatory Guide 1.33. SRP Branch Technical Position (BTP) 7-10 (see SRP Section 7.5) provides additional guidance on the application of Regulatory Guide 1.97. (Additional text follows on requirements) 2. Provisions should be made for the installation of instrumentation and monitoring equipment and/or periodic or continuous sampling and analysis of radioactive waste process systems. For GDC 60 and 63, as they relate to radioactive waste systems, detection of excessive radiation levels, and initiation of appropriate safety actions, the design of systems should meet the guidelines of Appendix 11.5-A, Regulatory Guide 1.21 (Position C, as applicable), Regulatory Guide 1.97 (Position C and Table 1 or 2, as applicable), Regulatory Guide 4.15 (Position C), and Appendix A to Regulatory Guide 1.33. SRP BTP 7-10 (see SRP Section 7.5) provides additional guidance on the application of Regulatory Guide 1.97. (Additional text follows on requirements)	Conformance with exceptions. Per expectation expressed in the SRP, operational programs will be provided by the COL applicant.	11.5

Table 1.9.2-11 US-APWR Conformance with Standard Review Plan Chapter 11 Radioactive Waste Management (sheet 10 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
11.5 Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems (continued)	3. Provisions should be made for administrative and procedural controls for the installation of necessary auxiliary or ancillary equipment, for the inclusion of special features in instrumentation and radiological monitoring sampling systems, and for the analysis of process and effluent streams. For GDC 63 and 64 (including the requirements specified in 10CFR50.34(f)(2)(xvii) and 10CFR50.34(f)(2)(xxvii)), as they relate to radioactive waste process systems and effluent discharge paths, the design of systems and the implementation of administrative and procedural controls should meet the guidelines of Appendix 11.5-A, Regulatory Guide 1.21 (Position C), Regulatory Guide 4.15 (Position C), and Appendix A to Regulatory Guide 1.33. (Additional text follows on requirements) 4. Provisions should be made for monitoring instrumentation, sampling, and sample analyses for all identified gaseous effluent release paths in the event of a postulated accident. For GDC 64, as it relates to potential gaseous effluent release paths, the design of systems should meet the provisions of NUREG-0718 and NUREG-0737 (item II.F.1 and Attachments 1 and 2), 10CFR50.34(f)(2)(vxii) and 10CFR50.34(f)(2)(vxxii), Appendix 11.5-A, and Regulatory Guide 1.97 (Position C). SRP BTP 7-10 (see SRP Section 7.5) provides additional guidance on the application of Regulatory Guide 1.97. In addition, the design of the gaseous waste collection and processing system should meet the guidelines referenced in SRP Sections 9.3.2, 11.3, and 13.3, as well as the following conditions: (Additional text follows on requirements) 5. Provisions should be made for monitoring instrumentation, sampling, and sample analysis for all identified liquid effluent release paths in the event of a postulated accident. These provisions should be in accordance with GDC 64 and the requirements of 10CFR50.34(f)(2)(vxii) and 10CFR50.34(f)(2)(vxxii), as they relate to postulated accidents and identified liquid effluent release paths. In addition, the design of the liquid waste collection and processing system should meet the guidelines referenced in SRP Sections 9.3.2, 11.2, and 13.3, as well as the following conditions: : (Additional text follows on requirements)		

Table 1.9.2-11 US-APWR Conformance with Standard Review Plan Chapter 11 Radioactive Waste Management (sheet 11 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
11.5 Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems (continued)	6. Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestone for the RETS/SREC, ODCM and REMP aspects of the Process and Effluent Monitoring and Sampling Program are reviewed in accordance with 10CFR20.1301 and 20.13.2, 10CFR50.34a, 10CFR50.36a, and 10CFRPart 50, Appendix I, Section II and IV. Its implementation is required by a license condition.		
Branch Technical Position 11-3: Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants	1. Processing Requirements (Additional text follows on requirements) 2. Assurance of Complete Stabilization or Dewatering (Additional text follows on requirements) 3. Waste Storage (Additional text follows on requirements) 4. Portable Solid Waste Systems (Additional text follows on requirements) 5. Additional Design Features (Additional text follows on requirements)	Conformance with exceptions. Operational and design information relating to portable solid waste systems, as mentioned in Criterion #4, will be provided by the COL Applicant.	11.4
Branch Technical Position 11-5: Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure	1. Waste Gas System Leak or Failure Analysis A. Criteria The SAR (Section 11.3) should provide an analysis of the radiological consequences of a single failure of an active component in the waste gas system. The analysis should provide reasonable assurance that, in the event of a postulated failure or leak of the waste gas system, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 25 mSv (2.5 rem) for systems designed to withstand explosions and earthquakes, or 1 mSv (0.1 rem) for systems not designed to withstand explosions and earthquakes. The bases for the analysis should include the assumption that the waste gas system fails to meet its design intent as required by 10CFR50.34a(c) and GDC 60 of Appendix A to 10CFRPart 50.	Conformance with no exceptions identified.	11.3

Table 1.9.2-11 US-APWR Conformance with Standard Review Plan Chapter 11 Radioactive Waste Management (sheet 12 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 11-5: Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure (continued)	B. Source Term The safety analysis on the radiological consequences of a single failure of an active component in the waste gas system should use a system design-basis source term for light-water-cooled nuclear power plants. The NRC staff method of calculation for this analysis is based on conservative assumptions to maximize the design capacity source term (sustained power operation). These assumptions are given below: *(Additional text follows on requirements)* C. Release The NRC staff considers that the release to the environment resulting from the postulated event will occur via a pathway not normally used for planned releases, and the release will require a reasonable time to detect and take remedial action to terminate the release. The NRC staff considers that the release of a compressed gas storage tank of a batch-type waste gas system or the inadvertent bypass of the main decay portion of a continuous-type waste gas system (such as charcoal delay beds in a BWR-augmented offgas system) will provide a conservative assumption for the release, while the input to the waste gas system is at the system design-basis source term. Only the radioactive noble gases (xenon and krypton) are to be considered since the assumed transit time is long enough to permit major radioactive decay of oxygen and nitrogen isotopes. Particulates and radioiodines are assumed to be removed by pretreatment, gas separation, and intermediate radwaste treatment equipment. The release should be assumed to occur within the building structure housing the waste gas system storage tank or the main decay position of the system. It should further be assumed that the effluent resulting from the postulated event will be released to the environs without continuous effluent radiation monitoring to automatically isolate and/or terminate the effluent release.		

Table 1.9.2-11 US-APWR Conformance with Standard Review Plan Chapter 11 Radioactive Waste Management (sheet 13 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 11-5: Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure (continued)	In addition, ground-level release without credit for a building wake factor should be assumed, and a conservative (5 percent) short-term diffusion estimate (X/Q), as determined by a method outlined in the acceptance criteria in SRP Section 2.3.4, should be assumed. No deposition is assumed to occur during downwind transport.		
Branch Technical Position 11-6: Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	1. Site Geology and Hydrology and Conceptual Transport Models The staff will review the site's geologic and hydrologic features in assessing the potential consequences of a release radioactive materials associated with the failure of a tank and its components on current and likely future users of ground or surface water. The review of information on surface and ground water hydrology, parameters governing the movement of liquids and mobility of radioactivity through soils, and potential dilution in water is performed under SRP Section 2.4.13. Briefly, these sections of the SRP address information describing streams and lakes, regional and local ground water aquifers, sources, and sinks, local and regional ground water users, known and likely future withdrawal rates, regional flow rates, travel time, gradients, and velocities, subsurface properties that affect movement of contaminants in ground water, ground water levels including their seasonal and climatic fluctuations, ground water monitoring and protection requirements, man-made changes that may affect regional ground water characteristics over time, and local practices in using water resources.	Conformance with exceptions. Site-specific geology and hydrology are to be provided by the COL Applicant.	11.1, 11.2, 11.5

Table 1.9.2-11 US-APWR Conformance with Standard Review Plan Chapter 11 Radioactive Waste Management (sheet 14 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 11-6: Postulated Radioactive Releases Due to Liquid-Containing Tank Failures (continued)	2. Radioactive Source Term The proposed radionuclide concentrations assumed for the postulated failure of a tank and its components will be reviewed by the staff using the information presented by the applicant. The analysis assumes that a tank and its components fail to meet the design bases as required by 10CFRPart 50.34a, and General Design Criteria 60 and 61. The staff will evaluate the basis and assumptions used in developing the source terms, radionuclide distributions and concentrations to ensure that the highest potential radioactive material inventory is selected among the expected types of liquid and wet waste streams processed by the LWMS. The radionuclide inventory for the tank and its components assumed to fail is based on 80% of the volume capacity of that tank and its component. The radionuclides selected for the radioactive source term and total inventory should include those that have the highest potential exposure consequences to users of water resources, including long-lived fission and activation products and environmentally mobile radionuclides. The radionuclide concentrations and total inventory of radioactive materials is based on the expected failed fuel fraction, i.e., 0.12% of the fuel producing power in a pressurized water reactor (PWR) as per NUREG-0017, or consistent with an offgas release rate of 0.555 MBq/sec per MWt (15 ìCi/sec per MWt) after a 30-minute delay for a boiling water reactor (BWR) as per NUREG-0016. The radionuclide inventory in failed components is calculated based on the methods given in Chapter 4 and Appendices A and B of NUREG 0133, or by using equivalently documented techniques. The staff will confirm that the initial inventory of radioactive materials corresponds to the highest expected concentrations and inventory of radioactivity in systems and components used to process, treat, or store liquid and wet wastes products associated with normal operation and anticipated operational occurrences, The reviewer will determine whether the tank and its components, for which a failure is assumed, will result in the highest concentrations of radioactive materials at the nearest potable water supply located in an unrestricted area.		

Table 1.9.2-11 US-APWR Conformance with Standard Review Plan Chapter 11 Radioactive Waste Management (sheet 15 of 15)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 11-6: Postulated Radioactive Releases Due to Liquid-Containing Tank Failures (continued)	3. Mitigating Design Features The staff will determine whether the analysis has considered the use of design features, e.g., steel liners or walls in areas housing components, dikes for outdoor tanks, and overflow provisions incorporated to mitigate the effect of a postulated tank failure. The types of failed components are typically waste collector tanks or sample tank, among others. However, the components selected for the analysis should realistically reflect the specific design features of the plant, as described in Sections 11.2 and 11.4 of the application. The staff will coordinate this part of the evaluation with the organization responsible for the review of systems and components that are part of the balance of plant. The purpose of this review is to ensure that the analysis considered the proper selection of the failed equipment, and appropriate release mechanisms from the selected equipment and buildings housing such systems. Credit for liquid retention by unlined building foundations will not be given regardless of the building seismic category because of the potential for cracks. Credit is not allowed for retention by coatings or leakage barriers outside the building foundation. 4. Specifications on Tank Waste Radioactivity Concentration Levels The reviewer will evaluate the proposed technical specification limiting the radioactivity content (becquerel, curie) of liquid-containing tanks to ensure that the technical specification is consistent with the safety evaluation. Chapter 16 of the SRP identifies the requirements for this technical specification. The radioactivity content (becquerel, curie) is based on that quantity which would not exceed the concentration limits of 10CFRPart 20, Appendix B, Table 2, Column 2, at the nearest potable water supply, located in an unrestricted area, in the event of an uncontrolled release of the tank's contents.		

Table 1.9.2-12 US-APWR Conformance with Standard Review Plan Chapter 12 Radiation Protection (sheet 1 of 7)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
12.1 Assuring that Occupational Radiation Exposures Are As Low as is Reasonably Achievable	1. Policy Considerations Acceptability will be based on evidence that a policy for ensuring that ORE will be ALARA has been formulated in accordance with the training requirements in 10CFR19.12 and the ALARA provisions of 10CFR20.1101(b), and that the policy has been described, displayed, and will be implemented in accordance with the provisions of Regulatory Guides 8.8 (Regulatory Position C.1) and 8.10 (Regulatory Position C.1) and NUREG-1736, as it relates to maintaining doses ALARA. A specific individual(s) will be designated and assigned responsibility and authority for implementing ALARA policy. Alternative proposed policies will be evaluated on the basis of a comparison with the above regulatory guides and NUREG-1736. 2. Design Considerations Acceptability will be based on evidence that the design methods, approach, and interactions are in accordance with the ALARA provisions of 10CFR20.1101(b) and Regulatory Guide 8.8 (Regulatory Position C.2) and will include incorporation of measures for reducing the need for time spent in radiation areas; maintenance; measures to improve the accessibility to components requiring periodic maintenance or inservice inspection; measures to reduce the production, distribution, and retention of activated corrosion products throughout the primary system; measures for assuring that ORE during decommissioning will be ALARA; reviews of the design by competent radiation protection personnel; instructions to designers and engineers regarding ALARA design; experience from operating plants and past designs; and continuing facility design reviews. Alternative proposed design policies will be evaluated on the basis of a comparison with the design guidance in Regulatory Guide 8.8 (Regulatory Position C.2).	Conformance with exceptions. 1 and 2 are conformance to US-APWR design certification. Other numbers are not applicable to US-APWR design certification.	12.1

Table 1.9.2-12 US-APWR Conformance with Standard Review Plan Chapter 12 Radiation Protection (sheet 2 of 7)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
12.1 Assuring that Occupational Radiation Exposures Are As Low as is Reasonably Achievable (continued)	3. Operational Considerations Acceptability will be based on evidence that the applicant has a program to develop plans and procedures in accordance with Regulatory Guides 1.33, 1.8, 8.8, and 8.10 that can incorporate the experiences obtained from facility operation into facility and equipment design and operations planning and that will implement specific exposure control techniques. 4. Radiation Protection Considerations Acceptability will be based on evidence that overall facility operations, as well as the radiation protection program, integrate the procedures necessary to ensure that radiation doses are ALARA, including work scheduling, work planning, design modifications, and radiological considerations.		

Table 1.9.2-12 US-APWR Conformance with Standard Review Plan Chapter 12 Radiation Protection (sheet 3 of 7)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
12.2 Radiation Sources	Descriptions should be provided for all radiation sources that require (1) shielding, (2) special ventilation systems, (3) special storage locations and conditions, (4) traffic or access control, (5) special plans or procedures, or (6) monitoring equipment. The source descriptions should include all pertinent information required for (1) input to shielding codes used in the design process, (2) establishment of related facility design features, (3) development of plans and procedures, and (4) assessment of occupational exposure. For contained sources, the description should include plan scale drawings of each floor of the plant that show all sources identified so that they can easily be related to tables containing the pertinent and necessary quantitative source parameters. Their position should be located accurately, indicating the approximate size and shape. Neutron and gamma streaming into containment from the annulus between the reactor pressure vessel and the biological shield should be analyzed to determine the radiation fields that could occur in areas that may require occupancy. Relevant experience from operating reactors may be used. Airborne sources that are created by leakage, opening formerly closed containers, storage of leaking fuel elements, and other mechanisms should be identified by location and magnitude so that they can be used for designing appropriate ventilation systems and in specifying appropriate monitoring systems. Airborne radioactivity concentrations in frequently occupied areas should be a small fraction of the concentrations related to 10CFR20.1203, 10CFR20.1204, and Appendix B to 10CFRPart 20. The assumptions made in arriving at quantitative values for these various sources should be specified, either in this section or by reference to SAR Chapter 11.	Conformance with no exceptions identified.	12.2

Table 1.9.2-12 US-APWR Conformance with Standard Review Plan Chapter 12 Radiation Protection (sheet 4 of 7)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
12.2 Radiation Sources (continued)	Shielding and ventilation design fission product source terms will be acceptable if developed using these bases: • An offgas rate of 370 MBq/s (100,000 μCi/s) after a 30-minute delay for BWRs. • 0.25-percent fuel cladding defects for PWRs. • Postaccident shielding (for vital area access, including work in the area) source terms from NUREG-0737, Item II.B.2, or Regulatory Guide 1.183. Coolant and corrosion activation products source terms should be based on applicable reactor operating experience. The buildup of activated corrosion products in various components and systems should be addressed. Any allowances made in design source terms for the buildup of activated corrosion products should be explained. Neutron and prompt gamma source terms should be based on reactor core physics calculations and applicable reactor operating experience. The tables of source parameters, which can be placed in SAR Chapter 12 or referenced to SAR Chapter 11, will be acceptable if the accompanying text either in this section or other referenced sections makes it clear how the values are used in a shield design calculation or in a ventilation system design. In addition, the quantities will be acceptable if the specific values given in the tables are consistent with ANSI/ANS Standard 18.1 and Regulatory Guide 1.112 for coolant and corrosion activation products source terms. For PWRs designed for the recycling of tritiated water, tritium concentrations in contained sources and airborne concentrations in the regions specified in item I.2 above should be based on a primary coolant concentration of 1.3x10 Bq/gm (3.5 μCi/gm).		

Table 1.9.2-12 US-APWR Conformance with Standard Review Plan Chapter 12 Radiation Protection (sheet 5 of 7)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
12.3-12.4 Radiation Protection Design Features	1. Facility Design Features The acceptability of the facility design features will be based on evidence that the applicant has fulfilled the dose limiting requirements of 10CFR20.1201, 10CFR20.1202, 10CFR20.1203, 10CFR20.1204, and 10CFR20.1207, as well as the radiation protection aspects of GDC 19 and 61, and 10CFR50.34. This includes evidence that major exposure accumulating functions (maintenance, refueling, radioactive material handling and processing, inservice inspection, calibration, decommissioning, and recovery from accidents) have been considered in plant design and that radiation protection features incorporated into the design will keep potential radiation exposure from these activities ALARA in accordance with 10CFR20.1101(b), the definition of ALARA in 10CFR20.1003, and Regulatory Guides 8.8 and 8.10. Such features may include (1) the ease of accessibility to work, inspection, and sampling areas, (2) the ability to reduce source intensity, (3) design measures to reduce the production, distribution, and retention of activated corrosion products, (4) the ability to reduce time required in radiation fields, and (5) a provision for portable shielding and remote handling tools. Access control will be judged for acceptability in accordance with the requirements of 10CFR20.1601, 10CFR20.1602, 10CFR20.1901, 10CFR20.1902, and 10CFR20.1903 or access control alternatives in Standard Technical Specifications (NUREG-1430, NUREG-1431, NUREG-1432, NUREG-1433, and NUREG-1434). *(Additional text follows on criteria)* 2. Shielding The staff will evaluate the shielding design in terms of the assumptions used to calculate shield thickness, the calculational methods used, and the parameters chosen. A number of acceptable shielding calculational codes are available that are effective for determining the necessary shield thickness for gamma ray and combination neutrongamma sources. *(Additional text follows on criteria)*	Conformance with exceptions. Criteria 8 and 12: The details of radiation monitoring program will be provided by COL applicants)	12.3, 12.4

Table 1.9.2-12 US-APWR Conformance with Standard Review Plan Chapter 12 Radiation Protection (sheet 6 of 7)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
12.3-12.4 Radiation Protection Design Features (continued)	3. Ventilation The ventilation system will be acceptable for radiation protection purposes if the criteria and bases for ventilation rates within the areas covered in SAR Section 12.2.2 will ensure that air will flow from areas of low potential airborne radioactivity to areas of higher airborne radioactivity and then to filters or vents, that the concentrations of radioactive material in areas normally occupied can be maintained in accordance with the requirements 10CFR20.1701, and that the dose limits of 10CFR20.1201 are met consistent with the requirements of 10CFR20.1202, 10CFR20.1203, and 10CFR20.1204. The system has adequate capability to reduce concentrations of airborne radioactivity to 1.0 derived air concentration (DAC), as specified in Appendix B to 10CFRPart 20, in areas not normally occupied where maintenance or inservice inspection must be performed. The system is designed so that filters containing radioactivity can be easily maintained and will not create an additional radiation hazard to personnel maintaining them, or those in adjacent occupied areas. Acceptability of the ventilation system, relative to radioactive gases and particulates, will also be based on evidence that the applicant has applied the guidance of Regulatory Guide 8.8 or proposed acceptable alternatives. Regulatory Guide 1.52, particularly Sections C.3.10 and 4.10, provides guidance that can be used in this review, although the guide relates to mitigating accidents involving airborne radioactivity. Good practices in that regard apply to normal operation as well, since the release of radioactivity in normal operational occurrences is usually different only in quantity from some of the accident cases. 4. Area Radiation and Airborne Radioactivity Monitoring Systems (Additional text follows on criteria)		

Table 1.9.2-12 US-APWR Conformance with Standard Review Plan Chapter 12 Radiation Protection (sheet 7 of 7)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
12.3-12.4 Radiation Protection Design Features (continued)	5. Dose Assessment The dose assessment will be acceptable if it documents in appropriate detail the assumptions made, calculations used, results for each radiation zone (including numbers and types of workers involved in each), expected and design dose rates, and projected person-Sievert (person-rem) doses, in accordance with Regulatory Guide 8.19.		
12.5 Operational Radiation Protection Program	1. Organization Acceptance will be based on a determination that the organization described, and the duties, qualifications, and training of the individuals responsible for ensuring that ORE will be ALARA; (1) are in accordance with 10CFR20.1101(b) and the definition of ALARA in 10CFR20.1103; Regulatory Guides 1.8, 8.2, 8.8, and 8.10; and 10CFR19.12; and (2) are such that doses resulting from licensed activities fall within the limits of 10CFR20.1201, 10CFR20.1202, 10CFR20.1203, 10CFR20.1204, 10CFR20.1301, 10CFR20.1302, 10CFR50.120, NUREG-0731, and NUREG-1736. Alternatives will be evaluated on the basis of a comparison with the referenced regulatory guides. 2. Equipment, Instrumentation, and Facilities Acceptance will be based on a determination of the following: *(Additional text follows on criteria)*	Not applicable. The details of the operational radiation protection program will be provided by COL applicants.	N/A

Table 1.9.2-13 US-APWR Conformance with Standard Review Plan Chapter 13 Conduct of Operations (sheet 1 of 9)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
13.1.1 Management and Technical Support Organization	Specific criteria are described below for meeting 10CFR50.40(b) with respect to the CP, OL, COL reviews and 10CFR50.80 with respect to license transfer reviews. A. CPs and COLs i. The applicant has identified and functionally described the organizational groups responsible for implementing the project. ii. The applicant has described how it will carry out its responsibilities to consider safety first in designing and construct the project and during the transition to operation and to control major contractors. iii. The organizational units involved in the design and construction of the project communicate fully and frankly among each other and with management, and management clearly and unambiguously controls the project. iv. Manpower with suitable experience is available to implement the project. v. The applicant has clearly described the role and function of the AE and the NSSS vendor during both design and construction and has demonstrated appropriate control over the project-related activities of the AE and NSSS vendor. vi. The applicant has designated the organizations responsible for the test program, and early plans give reasonable assurance that the designated organizations can collectively provide staff with the skills and experience necessary to develop and conduct the test program. vii. The applicant plans to utilize the plant operating and technical staff in developing and conducting the test program and in reviewing test results. viii. For COL Applicants subject to 10CFR50.34(f)1, the applicant has identified plans for the organization and staffing to oversee design and construction of the nuclear facility, in accordance with the guidelines of Item II.J.3.1 of NUREG-0718, as related to the requirements of 10CFR50.34(f)(3)(vii). As referenced in SRP Section 18.0, the review criteria for the human factors engineering (HFE) design team are provided in NUREG-0711, Chapter 2, "Element 1 - HFE Program Management."	Conformance with exceptions. The details are Combined License applicant responsibility.	13.1.1

Table 1.9.2-13 US-APWR Conformance with Standard Review Plan Chapter 13 Conduct of Operations (sheet 2 of 9)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
13.1.2-13.1.3 Operating Organization	1. Specific Requirements. Specific criteria to meet the relevant requirements of 10CFR50.40(b), 10CFR50.80, and 10CFR50.54(j), (k), (l), and (m) are as follows: A. ANSI N18.7/ANS-3.2, Section 3.4, “Operating Organization,” as endorsed by Regulatory Guide 1.33, should be met. In addition, the following criteria should be satisfied: (Additional text follows on requirements) B. Responsibilities and authorities of operating organization personnel… (Additional text follows on requirements) C. Assignments of onsite shift operating crews… (Additional text follows on requirements) D. Any deviation from the Specific Criterion B.3.a-f and/or the staffing- related requirements of 10CFRPart 50… (Additional text follows on requirements) E. The total complement of licensed and unlicensed personnel… (Additional text follows on requirements) F. The plant operating and technical staff should be used as much as possible in the initial test program for the facility… (Additional text follows on requirements) G. Assignments of personnel to the fire brigade… (Additional text follows on requirements) H. Regulatory Guide 1.8, “Qualification and Training of Personnel for Nuclear Power Plants,” sets forth the staff position on plant personnel qualifications and training. In addition, although the qualification levels of the standards are endorsed as acceptable minimums for each position, it is expected that the collective qualifications of the plant staff will be greater than the sum of the minimum individual requirements described in the standard, particularly in the area of nuclear power plant experience and in supervisory and managerial positions involved in operating the facility. If the collective qualifications do not exceed the sum of the minimums for individual positions, additional technical support for the plant staff may be required. This will be determined on a case-by-case basis.	Conformance with exceptions. The details are Combined License applicant responsibility.	13.1.2

Table 1.9.2-13 US-APWR Conformance with Standard Review Plan Chapter 13 Conduct of Operations (sheet 3 of 9)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
13.1.2-13.1.3 Operating Organization (continued)	For review of a DC application, the reviewer should consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.		
13.2.1 Reactor Operator Requalification Program; Reactor Operator Training	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestone for the Reactor Operator Requalification Program are reviewed in accordance with 10CFR50.34(b), 10CFR50.54 (i), and 10CFR55.59. The implementation milestone is within 3 months after issuance of license or the date that the Commission makes the finding under 10CFR52.103(g) per 10CFR50.54 (i-1). The description of the operational program for the Reactor Operator Training Program is reviewed in accordance with 10CFR55.13, 10CFR55.31, 10CFR55.41, 10CFR55.43, and 10CFR55.45. Its implementation is required by a license condition. 4. For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.	Conformance with exceptions. The details are Combined License applicant responsibility.	13.2
13.2.2 Non-Licensed Plant Staff Training	3. Design Certification. The development of training programs will be designated as the responsibility of a COL Applicant. 3. For Design Certification. The development of the non-licensed plant staff training programs is identified as a COL action item. There are no acceptance criteria in this SRP that are applicable to the Design Certification phase of the US-APWR.	Conformance with exceptions. The details are Combined License applicant responsibility.	13.2

Table 1.9.2-13 US-APWR Conformance with Standard Review Plan Chapter 13 Conduct of Operations (sheet 4 of 9)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
13.3 Emergency Planning	22. 10CFR52.47(b)(1) allows an applicant for a design certification to include proposed ITAAC, including those applicable to emergency planning, which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations. For a design certification application, the review is conducted against the requirements in 10CFR52.47 and 10CFR52.48, and only addresses those design features, facilities, functions, and equipment that are technically relevant to the design and are not site-specific, and which affect some aspect of emergency planning or the capability of a licensee to cope with plant emergencies. The review addresses such areas as a habitable technical support center (TSC) with adequate space, data retrieval capabilities and dedicated communications equipment, and an operational support center (OSC) with adequate communications. Additional design-related features associated with emergency planning, such as EALs, may also be included in the application for review. There is no minimum amount of design-related emergency planning for the proposed reactor that must be addressed in an application. The applicant may choose the extent to which emergency planning features are included in the application to be reviewed as part of the certified design. Standard Design Certification	Conformance with exceptions. However, details of emergency planning are responsibility of the Combined License applicant. Therefore, US-APWR Design Certification Document includes the design bases of design features, facilities, functions, and equipment necessary for emergency planning for standard plant.	13.3

Table 1.9.2-13 US-APWR Conformance with Standard Review Plan Chapter 13 Conduct of Operations (sheet 5 of 9)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
13.3 Emergency Planning (continued)	1. The reviewer should examine the requirements in 10CFR52.47 and 10CFR52.48, relating to the application contents and standards for review, respectively. Emergency planning basically consists of facilities, equipment, personnel and training. The majority of emergency planning requirements are programmatic in nature and supplement physical facilities and equipment. The reviewer should confirm that any emergency planning features addressed in the application are technically relevant to the design (i.e., facilities and equipment) proposed for the facility and not site-specific (i.e., programmatic in nature), and are usable for a multiple number of units or at a multiple number of sites. In general, programmatic aspects of emergency planning and preparedness are the responsibility of a COL Applicant that references the certified standard design. The application may, but is not required to, identify such programmatic responsibilities as COL action or information items. Although the COL Applicant will address most aspects of emergency planning, the standard design may consider design features, facilities, functions, and equipment necessary to support emergency preparedness and response. 2. If applicable, the reviewer should confirm that the application identifies the technically relevant portions of the requirements in 10CFR50.34(f)(1) through 10CFR50.34(f)(3), and determine whether the application demonstrates compliance with them (see 10CFR52.47(a)(17)). 3. The reviewer should examine the relevant sections of the SAR that address facilities, equipment, and systems that support the emergency preparedness and response capabilities of the proposed reactor design. The application may, but is not required to, address facilities that support emergency response. These facilities include, but are not limited to, the TSC, OSC, and decontamination facilities. The reviewer should determine whether the proposed facilities satisfactorily meet the relevant acceptance criteria, which address location, size, and habitability during an emergency.		

Table 1.9.2-13 US-APWR Conformance with Standard Review Plan Chapter 13 Conduct of Operations (sheet 6 of 9)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
13.3 Emergency Planning (continued)	4. The reviewer should determine whether the proposed equipment and system designs that support the facilities satisfactorily meet the relevant acceptance criteria. For example, the reviewer should examine, at a minimum, the proposed ventilation system that ensures the habitability of the TSC. To the extent that the TSC shares a common ventilation system with the control room or other area of the plant, the reviewer should also examine that aspect of the design to determine any impact on TSC habitability. In addition, if addressed in the application, and to the extent that it is related to the non-site-specific design, the reviewer should also examine the ERDS, SPDS, voice and data communications capabilities, and radiological protection, monitoring and decontamination equipment. The application may, but is not required to, address these additional equipment and system descriptions. Further, the application may, but is not required to, identify these additional descriptions as COL action or information items. 5. The reviewer should examine the proposed ITAAC, and should determine whether the ITAAC are necessary and sufficient to provide reasonable assurance that, if the tests, inspections, and analyses are performed and the acceptance criteria met, a plant which references the design will be built, and will operate, in accordance with the design certification. 6. The procedures above should be followed, as modified by the procedures in SRP Section 14.3, to verify that the design set forth in the standard SAR (including ITAAC), site interface requirements and COL action or information items, meet the acceptance criteria given in Subsection II. SRP Section 14.3 contains procedures for the review of certified design material for the standard design, including the site parameters, interface criteria, and ITAAC.		
13.4 Operational Programs	The SRP sections referenced in the attached sample FSAR Table 13.4-x include SRP acceptance criteria for operational programs. *(See referenced table for requirements.)*	Conformance with exceptions. The details are Combined License applicant responsibility.	13.4

Table 1.9.2-13 US-APWR Conformance with Standard Review Plan Chapter 13 Conduct of Operations (sheet 7 of 9)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
13.5.1.1 Administrative Procedures - General	For DC and COL reviews, the findings will also summarize the staff's evaluation of COL action items relevant to this SRP section.	Conformance with exceptions. The details are Combined License applicant responsibility.	13.5
13.5.1.2 Administrative Procedures - Initial Test Program (Content subsumed into SRP Section 14.2)	No SRP issued.	Not applicable. This SRP has not been issued.	N/A
13.5.2.1 Operating and Emergency Operating Procedures	It is recognized that development of detailed procedures and associated training materials may be beyond the scope of the application (e.g., for design certification) and are the responsibility of a combined license (COL) applicant referencing the certified design. The SAR should provide descriptions of the content and development process for procedures as detailed below, including preliminary schedules for preparation of procedures.	Conformance with exceptions. The details are Combined License applicant responsibility.	13.5
13.5.2.2 Maintenance and Other Operating Procedures (Content subsumed into SRP Section 17.5)	No SRP issued.	Not applicable. This SRP has not been issued.	N/A
13.6 Physical Security	Guidance for the review of design certification applications is located in SRP 13.6.2. (This SRP contains no acceptance criteria. See SRP 13.6.2.)	Not applicable. This SRP contains no acceptance criteria.	N/A
13.6.1 Physical Security - Combined License	(This SRP contains no acceptance criteria applicable to Design Certification of the US-APWR. See SRP 13.6.2.)	Not applicable. This SRP is provided for Combined License.	N/A

Table 1.9.2-13 US-APWR Conformance with Standard Review Plan Chapter 13 Conduct of Operations (sheet 8 of 9)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
13.6.2 Physical Security - Design Certification	1. Section (c) of 10CFR73.55 - Physical Barriers The licensee shall locate vital equipment only within a vital area, which in turn, shall be located within a protected area such that access to vital equipment requires passage through at least two physical barriers as defined in 10CFR73.2. The physical barriers at the perimeter shall be separated from any other barrier designated as physical barrier for a vital area within the protected area. Isolation zones in outdoor areas adjacent to the physical barrier at the perimeter of the protected area permit observation. Intrusion detection system detects penetration or attempted penetration of the protected area (PA) barrier. All exterior areas within the protected area are illuminated. The external walls, doors, ceiling and floors in the main control room are bullet resistant. Vehicle control measures which include vehicle barrier systems protect against the use of land vehicle. 2. Section (d) of 10CFR73.55 - Access Requirements The licensee shall control all points of personnel and vehicle access into a protected area, to include detection equipment capable of detecting firearms, explosives and incendiary devices. Unoccupied vital areas are locked and alarmed with activated intrusion detection systems that annunciate in both the central and secondary alarm stations upon intrusion into a vital area. The individual responsible for the last access control function (controlling admission to the protected area) must be isolated within a bullet-resisting structure.	Not applicable. These are to be addressed in Physical Security Element Review.	N/A

Table 1.9.2-13 US-APWR Conformance with Standard Review Plan Chapter 13 Conduct of Operations (sheet 9 of 9)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
13.6.2 Physical Security - Design Certification (continued)	3. Section (e) of 10CFR73.55 - Detection Aids All alarms required pursuant to this part shall annunciate in a continuously manned central alarm station located within the protected area and in at least one other continuously manned station, not necessarily onsite, such that a single act cannot remove the capabilities of calling for assistance or otherwise responding to an alarm. The central alarm station shall be considered a vital area, shall be bullet-resisting, the interior will not be visible from the protected area perimeter, and associated onsite secondary power supplies for alarm annunciators and non-portable communication equipment must be located within vital areas. Alarm devices and transmission lines must be tamper indicating and self checking. Alarm annunciation shall indicate type of alarm and location. All emergency exits from protected and vital areas shall be alarmed. 4. Section (f) of 10CFR73.55 - Communication Requirements Each security officer, watchman or armed response individual shall be capable of maintaining continuous communications with an individual in each continuously manned alarm stations. Conventional telephone and radio or microwave transmitted two-way voice communications shall be established with local law enforcement authorities. 5. Section (g) of 10CFR73.55 - Testing and Maintenance Each applicant shall develop test and maintenance provisions for intrusion alarms, emergency alarms, communication equipment, access control equipment, physical barriers, and other security-related devices or equipment.		
13.6.3 Physical Security - Early Site Permit	(This SRP contains no acceptance criteria applicable to Design Certification of the US-APWR. See SRP 13.6.2.)	Not applicable. This SRP is provided for Early Site Permit.	N/A

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 1 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.2 Initial Plant Test Program - Design Certification and New License Applicants	1. Summary of Test Program and Objectives This SRP section lists the general criteria of RG 1.68 that a DC, COL, or OL applicant or holder should address in its safety analysis report (SAR). DC/COL/OL Applicants A. The ITP should describe its objectives, including a description of the objectives for each of the major phases of the test program. B. The ITP should describe the criteria for selection of plant features to be tested by the applicant. C. Objectives and testing selection criteria should be consistent with the general guidelines and applicable regulatory positions in RG 1.68. Applicants should appropriately justify exceptions. 2. Test Program's Conformance with Regulatory Guides DC/COL/OL Applicants A. The applicant should commit to the revision of RG 1.68 and the RGs listed in RG 1.68, that are referenced in this SRP and are in effect six months prior to submittal. The applicant may propose exceptions or alternatives to the specific criteria in any of these RGs, and the staff may find them acceptable if the applicant provides adequate justification. The reviewer responsible for the RG evaluates any exceptions or alternatives. The safety evaluation report (SER) should also list such exceptions or alternatives. 3. Initial Test Program Administrative Procedures DC Applicant The applicant should provide a summary description of the following areas: A. The applicant should provide general guidance to control ITP activities, including administrative controls that will be used to develop, review, and approve individual test procedures, coordination with organizations involved in the test program, participation of plant operating and technical staff, and review, evaluation, and approval of test results.	Conformance with exceptions for the guidance of a COL applicant.	14.2

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 2 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.2 Initial Plant Test Program - Design Certification and New License Applicants (continued)	B. The applicant should include general guidance for the review of relevant operating and testing experiences at other facilities. This guidance should recognize reportable occurrences of repeatedly experienced safety concerns and other operating experiences that could potentially impact the performance of the test program. C. The applicant should include general guidance about how, and to what extent, the test program will use and/or test plant operating, emergency, and surveillance procedures. D. The applicant should provide test abstracts of SSCs and unique design features that will be tested to verify that system and component performance is in accordance with the design. These test abstracts should include the objectives, tests, and acceptance criteria that will be included in the test procedures. 4. Initial Startup Tests DC Applicant The applicant should provide a summary description of the following areas: A. Initial Fuel Loading/Initial Criticality/Low-Power/Power Ascension Testing i. The applicant should include in the ITP a description of the general provisions and precautions for fuel loading, initial fuel loading, initial criticality, low-power testing, and power ascension phases. Precautions, prerequisites, and measures should be consistent with the guidelines and regulatory positions in RG 1.68. This includes guidance for (1) the completion of all ITAAC associated with preoperational tests before fuel load, (2) measures to review and evaluate the results of the completed preoperational tests, (3) appropriate remedial actions to take if acceptance criteria are not satisfied, (4) applicable technical specification requirements, and (5) actions to take if unanticipated errors or malfunctions occur.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 3 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.2 Initial Plant Test Program - Design Certification and New License Applicants (continued)	5. Individual Test Descriptions/Abstracts DC/COL/OL Applicants A. The applicant should provide abstracts of planned tests to demonstrate and verify the performance capabilities of SSCs and design features that serve the following functions: i. Used for safe shutdown and cooldown of the reactor under normal plant conditions and for maintenance of the reactor in a safe condition for an extended shutdown period ii. Used for safe shutdown and cooldown of the reactor under transient conditions (infrequently or moderately frequent events) and postulated accident conditions and for maintenance of the reactor in a safe condition for an extended shutdown period following such condition iii. Used for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility technical specifications iv. Classified as engineered safety features or used to support or ensure the operations of engineered safety features within design limits v. Assumed to function, or for which credit is taken, in the accident analysis for the facility, as described in the DCD or SAR (as applicable) vi. Used to process, store, control, measure, or limit the release of radioactive materials vii. Used in a special low-power testing program to be conducted at power levels no greater than 5 percent for the purpose of providing meaningful technical information beyond that obtained in the normal startup test program, as required for the resolution of TMI Action Item I.G.1 viii. Identified as risk significant in the design-specific probabilistic risk assessment B. The abstracts should include test objectives, prerequisites, test methods, significant parameters and plant performance characteristics to be monitored, and acceptance criteria in sufficient detail to establish the functional adequacy of the SSCs and design features tested.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 4 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
	C. For new, unique, or first-of-a-kind design features used in the facility, the functional testing requirements and acceptance criteria necessary to verify their performance should be submitted for review and approval. D. If the testing method will not subject the SSC to representative design operating conditions, the test abstract should contain sufficient information to justify the proposed test method. 6. Initial Test Program Acceptance Criteria DC Applicants A. The applicant should provide in Tier 1 a general description of the preoperational and power ascension test programs and the major program documents that define how the ITP will be conducted and controlled (i.e., a site-specific startup administrative manual, test specifications, and test procedures). Tier 2, Chapter 14.2, should contain a complete description of the ITP. B. The applicant should describe the key elements of the ITP in Tier 1 to ensure that the COL Applicant cannot unilaterally initiate subsequent changes in the conduct of the ITP. C. The applicant should include provisions to ensure that test procedures and test specifications are made available to the NRC.		
14.2.1 Generic Guidelines for Extended Power Uprate Testing Programs	This SRP is for power uprates and is not applicable to the US-APWR design certification.	Not applicable. US-APWR does not intend to perform an extended power uprate power ascension test program.	N/A
14.3 Inspections, Tests, Analyses, and Acceptance Criteria	1. Acceptance on the scope of the ITAAC is based on the complete facility or for a DC application, limited to the SSCs covered by the DC. 2. Acceptance criteria on the sufficiency of the ITAAC for the areas of review are specified in SRP Section 14.3 subsections.	Conformance with no exceptions for the safety related features. ITAAC implementation is not a scope of this section.	14.3
14.3.1 [Reserved]	Reserved. No SRP issued	Not applicable. NRC has not issued an SRP under this number/title.	N/A

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 5 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.2 Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria	1. The reviewer should primarily utilize the NRC rules and regulations to review the top level commitments in Tier 1. Other sources of review guidelines include RGs, SRP guidelines, and PRA insights from the standard design safety and severe accident analyses and operating experience. If applicable, the staff also must adhere to policy decisions by the Commission. Examples of these are contained in the SRM related to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," as modified by the Commission guidance in the SRM related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs." The SRM related to SECY-93-087 is dated July 21, 1993. 2. Design descriptions, figures (including key dimensions) and ITAAC should be developed and grouped by systems and building structures. For building structures, the structural capability is typically verified by performing an analysis to reconcile the as-built data with the structural design bases for each safety-related building. System-specific performance tests are typically conducted to demonstrate that the system can perform its intended function. For major components, the verification of design, fabrication, testing, and performance requirements should be partially addressed in conjunction with the specific system ITAAC. The review checklists for fluid systems, electrical systems, and building structures in Appendix C of SRP Section 14.3 should be used as aids for establishing consistency and completeness for the Tier 1 information.	Conformance with no exceptions for the safety related features. ITAAC implementation is not a scope of this section.	14.3.4.2

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 6 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.2 Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	3. Review of the Standard Design Structural Integrity The scope of structural design covers the major structural systems in the standard design plant, including the RPV, ASME Code Class 1, 2, and 3 piping systems, and major building structures (primary containment, R/B, control building, TB, service building, and radwaste building). For PWRs, this includes the reactor vessel (RV), ASME Code Class 1, 2, and 3 piping systems, and major building structures (primary containment, nuclear island structures, TB, component cooling water (CCW) heat exchanger structures, diesel fuel storage structures (DFSSs), and radwaste building). The RPV, piping systems, and primary containment (For PWRS, RV, piping systems, and primary containment) are included because they provide the defense-in-depth principle for nuclear plants. The major building structures house those systems and components that are important to safety. In establishing the top level requirements for structural design, the staff used the General Design Criteria (GDC) of 10CFRPart 50, Appendix A, as its basis. *(Additional text follows on requirements.)*		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 7 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.2 Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	4. Pressure Boundary Integrity To ensure that the applicable requirements of GDC 14, 16, and 50 have been adequately addressed, ITAAC should be established to verify the pressure boundary integrity of the RPV, piping, and primary containment (For PWRs, RV, piping, and primary containment) for the standard design . GDC 16, GDC 50, and 10CFR50, Appendix J apply to the primary containment and GDC 14 applies to the RPV (RV for PWRs) and the reactor coolant pressure boundary piping systems. The pressure integrity for these major structural systems are needed to ensure the defense-in-depth principle. For the RPV and piping, hydrostatic tests performed in conjunction with the ASME Boiler and Pressure Vessel Code, Section III, should be required by ITAAC. See the standard ITAAC for hydrostatic tests in Appendix D to SRP Section 14.3. For the primary containment, a structural integrity test and containment integrated leakage rate test should be required by ITAAC to be performed on the pressure boundary components of the primary containment in accordance with the ASME Boiler and Pressure Vessel Code, Section III, and 10CFR50, Appendix J. Because the requirements of GDC 14, 16, and 50 do not apply to the reactor, control, turbine, service, and radwaste buildings (nuclear island structures, TB, CCW heat exchanger structures, DFSSs, and radwaste building for PWRs), ITAAC are not required to verify the pressure integrity for these other buildings.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 8 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.2 Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	5. Normal Loads To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAAC should be established to verify that the normal and accident loads have been appropriately combined with the effects of natural phenomena. For piping systems, ITAAC should require an analysis to reconcile the as-built piping design with the design-basis loads (which include the appropriate combination of normal and accident loads). See SRP Section 14.3.3 for additional information. For the RPV, the fabrication may be performed primarily in the vendor's shop where adherence to design drawings is tightly controlled. Therefore, ITAAC for the as-built reconciliation of normal loads with accident loads for the RPV are inappropriate. Instead, ITAAC should verify that the ASME Code-required reports exist to document that the RPV has been designed, fabricated, inspected, and tested to Code requirements to ensure adequate safety margin. Similarly, for safety-related buildings, ITAAC should require an analysis for reconciling the as-built plant with the structural design basis loads (which include the combination of normal and accident loads with the effects of natural phenomena). The analysis results should be documented in a structural analysis report, the scope and contents of which must be described in Tier 2. The staff may determine that the design of certain structures does not require verification by ITAAC, based on their safety significance. In particular, these ITAAC should apply only to safety-related structures and are not applicable to the service and TBs (radwaste and turbine building for PWRs). However, ITAAC for other design aspects of structures may be appropriate.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 9 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.2 Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	6. Seismic Loads To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAAC are established to verify that the safety-related systems and structures have been designed to seismic loadings. Component qualification for seismic loads should be addressed by ITAAC for verifying the basic configuration of systems. See the standard ITAAC for basic configuration in Appendix D to SRP Section 14.3 for additional information, and the discussion in SRP Section 14.3.3. As discussed above for normal loads on piping systems and the RPV, ITAAC should require an analysis to reconcile the as-built piping design with the design basis loads (which include seismic loads). See also the discussion in SRP Section 14.3.3. For the RPV, ITAAC for the as-built reconciliation of seismic loads for the RPV are deemed to be inappropriate as previously discussed. Instead, ITAAC verify that the ASME Code required reports exist for the RPV ensuring that the RPV has been designed, fabricated, inspected, and tested to ASME Code requirements. For safety-related buildings, ITAAC require an analysis for reconciling the as-built plant with the structural design-basis loads (which include seismic loads). The analysis results are to be documented in a structural analysis report, as discussed above. These ITAAC apply only to safety-related structures and are not applicable to the service and turbine buildings (radwaste and turbine building for PWRs). However, because the leakage path for fission products includes components within the turbine building, the turbine building should be able to withstand the effects of a safe-shutdown earthquake, if not, ITAAC should be established to verify that, under seismic loads, the collapse of the turbine building will not impair the safety-related functions of any safety-related SSCs located adjacent to or within the turbine building. For non-seismic Category I SSCs, the need for ITAAC to verify that their failure will not impair the ability of near-by safety-related SSCs to perform their safety-related functions should be assessed based on the specific design.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 10 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.2 Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	If the design detail and as-built and as-procured information for many non-safety-related systems (e.g., field-run piping and balance-of-plant systems) is not provided by the applicant for design certification and the spatial relationship between such systems and seismic Category I SSCs cannot be established until after the as-built design information is available, the non-seismic to seismic (II/I) interaction cannot be evaluated until the plant has been constructed. Accordingly, the design criteria for ensuring acceptable II/I interactions and a commitment for the COL Applicant to describe the process for completion of the design of balance-of-plant and non-safety related systems to minimize II/I interactions and proposed procedures for an inspection of the as-built plant for II/I interactions should be specified as a COL action item in Tier 2 7. Suppression Pool Hydrodynamic Loads (BWRs only). This requirement is for boiling water reactors (BWRs) and does not apply to the US-APWR.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 11 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.2 Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	8. Flood, Wind, Tornado, Rain, and Snow To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAAC should be established to verify that the safety-related systems and structures have been designed to withstand the effects of natural phenomena other than those associated with seismic loadings. The effects include those associated with flood, wind, tornado, rain, and snow. These loadings do not apply to the RPV, the ASME Code Class 1, 2, and 3 piping systems and components, nor the primary containment (except for the exposed portions of the concrete containments) because they are all housed within the safety-related buildings. For safety-related buildings, ITAAC should require an analysis for reconciling the as-built plant with the structural design basis loads (which include the flood, wind, tornado, rain, and snow loads). Based on their safety significance, these ITAAC need apply only to safety-related structures and need not be applicable to the service and TBs (radwaste and TB for PWRs). For flooding, site parameters are specified that require the maximum flood level and ground water level be below the finished plant grade level. ITAACs also require inspections to verify that divisional flood barriers and watertight doors exist, and penetrations (except for watertight doors) in the divisional walls are sealed up to the internal and external flood levels. In addition, for safety-related buildings, flood barriers are established up to the finished plant grade level to protect against water seepage, and flood doors and flood barrier penetrations are provided with flood protection features. ITAAC should also require inspections to verify that watertight doors exist, penetrations (except for watertight doors) in the divisional walls are at least 2.5 m above the floor, and safety-related electrical, instrumentation, and control equipment are located at least 20 cm above the floor surface. In addition, for safety-related buildings, ITAAC should require that external walls below flood level are equal to or greater than 0.6 m to protect against water seepage, and penetrations in the external walls below flood level are provided with flood protection features.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 12 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.2 Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	9. Pipe Break To ensure that the applicable requirements of GDC 4 have been adequately addressed, ITAAC should be established to verify that the safety-related SSCs have been designed to the dynamic effects of pipe breaks. Component qualification for the dynamic effects of pipe breaks should be addressed by ITAAC established for verifying the basic configuration of systems. For the RPV, ITAAC that verify the basic configuration of the RPV system require an inspection of the critical locations that establish the bounding loads in the LOCA analyses for the RPV to ensure that the as-built areas not exceed the postulated break areas assumed in the LOCA analyses. In addition, ITAAC should be established to verify by inspections of as-built, high-energy pipe break mitigation features and of the pipe break analysis report that safety-related SSCs be protected against the dynamic and environmental effects associated with postulated high-energy pipe breaks. ITAAC to verify pipe break loads are not required for the turbine, service, and radwaste buildings (turbine and radwaste buildings for PWRs) either because they are not safety-related structures or there are no high-energy lines located within the structure.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 13 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.2 Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	10. Codes and Standards To ensure that the applicable requirements of GDC 1 have been adequately addressed, ITAAC should be established to verify that appropriate codes and standards are used in the design and construction of safety-related systems and components. In general, the staff considers those codes and standards endorsed by the regulations under 10CFR50.55a in determining which codes and standards were appropriate for Tier 1 verification. The ASME Boiler and Pressure Vessel Code, Section III for Code Class 1, 2, and 3 systems and components is established as the code for the design and construction of standard design piping systems and the RPV. For safety-related building designs, the staff should base its safety findings on audits of standard design calculations, which relied on specific codes and standards. These codes and standards are contained in the appropriate sections of DCD Tier 2 Chapter 3. Inspections will be conducted as a part of ITAAC to verify that ASME Code-required documents exist that demonstrate that the RPV, piping systems and containment pressure boundaries have been designed and constructed to their appropriate Code requirements. For other ASME Code components and equipment, the verification of Code compliance will be performed in conjunction with the quality assurance programs and by the authorized inspection agency as required by the ASME Boiler and Pressure Vessel Code. This DCD Tier 2 material should be considered for designation as Tier 2* information. Tier 2* information is information that, if considered for a change by an applicant or licensee that references the certified standard design, would require NRC approval prior to implementation of the change. Tier 2* material is discussed further in SRP Section 14.3.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 14 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.2 Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	11. As-built Reconciliation As discussed in various sections above, to ensure that the final as-built plant structures are built in accordance with the certified design as required by 10CFRPart 52, structural analyses should be performed which reconcile the as-built configuration of the plant structures with the structural design bases of the certified design. The structural analyses should be documented in structural analysis reports. Structural analysis reports should be verified in conjunction with ITAAC for the primary containment and the reactor, control, radwaste, and TBs (nuclear island structures, radwaste building, CCW heat exchangers, DFSSs, and TB for PWRs). The detailed supporting information on what is required for an acceptable analysis report should be contained in DCD Tier 2 Chapter 3. Similarly, for piping systems, an as-built analysis should be performed using the as-designed and as-built information. ITAAC should verify the existence of acceptable final as-built piping stress reports that conclude the as-built piping systems are adequately designed. See SRP Section 14.3.3 for additional information. For the RPV, the key dimensions of the RPV system should be verified in conjunction with the basic configuration check of the system. The key dimensions of the RPV system and the acceptable variations of the key dimensions should be provided in the certified design description. Alternatively, acceptable variations and the bases for them should be provided in Tier 2. For component qualification, tests, analyses, or a combination of tests and analyses should be performed for seismic Category I mechanical and electrical equipment (including connected instrumentation and controls) to demonstrate that the as-built equipment and associated anchorages are qualified to withstand design basis dynamic loads without loss of safety function. These test and analyses should be performed as a part of ITAAC to verify the basic configuration of the system in which the equipment is located. See Section 14.3.3 for additional information.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 15 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.3 Piping Systems and Components - Inspections, Tests, Analyses, and Acceptance Criteria	1. Generic Piping Design DC applicants may provide less than the complete design information for piping design before DC because the design may depend upon as-built and as-procured information. Instead, applicants may provide the processes and design acceptance criteria (DAC) by which design details in this area would be developed and evaluated. Implementation of the processes is the responsibility of the COL Applicant or licensee. The DAC are discussed further in to SRP Section 14.3, Appendix A. The reviewer should use the SRP guidelines to evaluate the piping design information in Tiers 1 and 2 and audit the piping design criteria in detail, including sample calculations. The staff should evaluate the adequacy of the structural integrity and functional capability of safety-related piping systems. The review is not limited to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Classes 1, 2, and 3 piping and supports, but includes buried piping, instrumentation lines, the interaction of non-seismic Category I piping with seismic Category I piping, and any safety-related piping designed to industry standards other than the ASME Code. The staff's evaluation should include the analysis methods, design procedures, acceptance criteria, and related ITAAC (and DAC if applicable) that are to be used for the completion and verification of the standard design piping design. (*Additional text follows on requirements.)* 2. Verifications of Components and Systems In addition to the generic approach to piping design in Tier 1, the verification of piping and component classification, fabrication, dynamic and seismic qualification, and selected testing and performance requirements is also addressed by specific ITAAC in the individual Tier 1 systems.	Conformance with no exceptions for the safety related features. ITAAC implementation is not a scope of this section.	14.3.4.3

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 16 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.3 Piping Systems and Components - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	A. Piping and Component Safety Classification. 10CFRPart 50, Appendix A, General Design Criterion (GDC) 1, requires that safety-related SSCs be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions performed. Nuclear power plant components classified as Quality Groups A, B, and C are required by 10CFR50.55a to meet the requirements for ASME Code Class 1, 2, or 3, respectively; therefore, SSC safety classifications should be in each system's design description, and the functional drawings should identify the ASME Code classification boundaries applicable to the safety class. (*Additional text follows on requirements.)* B. Fabrication (Welding). 10CFRPart 50, Appendix A, GDC 14, requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage. In addition, GDC 30 requires that component parts of the reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest quality standards practical. (*Additional text follows on requirements.)*		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 17 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.3 Piping Systems and Components - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	C. Hydrostatic Test. The integrity of the pressure boundary is required to be maintained because it is directly involved in preventing or mitigating an accident or event under the defense-in-depth principle. The pressure boundary integrity is also ensured, in part, through a hydrostatic test verifying the leak-tightness of the ASME Code piping systems. A hydrostatic test is generally specified by the ASME Code, Section III, for ASME Code Class 1, 2, and 3 SSCs to verify whether pressure integrity is maintained in the process of fabricating the overall piping system, including any welding and bolting requirements. However, the ASME piping stress report in the generic piping ITAAC does not include the results of hydrostatic tests; therefore, the standard hydrostatic test ITAAC in SRP Section 14.3, Appendix D, should be specified in each system ITAAC with ASME Code Class 1, 2, or 3 SSCs. The hydrostatic test ITAAC also may be specified in other appropriate Tier 1 systems. D. Equipment Seismic and Dynamic Qualification. The basic configuration ITAAC listed in SRP Section 14.3, Appendix D, include verifications of the dynamic qualification (e.g., seismic, loss-of-coolant accident, and safety relief valve discharge loads) of seismic Category I mechanical and electrical equipment (including connected instrumentation and controls) in the design descriptions and figures. This inspection verifies the capability of mechanical and electrical equipment in as-built condition, including anchorages, to perform safety functions during and following a SSE. Detailed supporting information for dynamic qualification requirements, including seismic qualification records, is in DCD Tier 2, Chapter 3. The Tier 2 information describing dynamic qualification of equipment should be considered for designation as Tier 2*. Tier 2* information is addressed further in SRP Section 14.3, Appendix A.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 18 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.3 Piping Systems and Components - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	E. MOVs and Other Valves. The verification of the design qualification of valves is performed in conjunction with the basic configuration check for mechanical equipment as discussed above. For MOVs in particular, a special inspection is part of the basic configuration check to verify the records of vendor tests that demonstrate MOV ability to function under design conditions. The list of MOVs in Tier 1 should include, but not be limited to, those with active safety-related functions. These may be listed in Tier 2 in the inservice testing plan or other locations. The DCD Tier 2, Chapter 3 material should have detailed supporting information for the CDM for the methods of the COL Applicant or licensee for the design, qualification, and testing of MOVs to demonstrate their design-basis capability. This material should be considered for designation as Tier 2* information. Tier 2* information is addressed further in SRP Section 14.3, Appendix A. (*Additional text follows on requirements.)*		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 19 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.4 Reactor Systems - Inspections, Tests, Analyses, and Acceptance Criteria	1. Appendix A of SRP 14.3 describes and provides guidance relative to the content of the DCD for a design certification application and defines Tier 1 and Tier 2 design-related information that is to be ultimately incorporated by reference into the design certification rules. The basis for identifying Tier 1 information as derived from Tier 2 information, which is essentially the same information as is required for a design certification application, is that the top-level design features and performance standards (Tier 1) are those that are most important to safety, including safety-related and defense-in-depth features and functions, and non-safety-related systems that potentially impact safety. Tier 1 should be reviewed to verify that plant safety analyses, such as for core cooling, transients, overpressure protection, steam generator tube rupture, and anticipated transients without scram (ATWS), are adequately addressed. Applicants should provide tables in DCD Tier 2 Section 14.3 to show how the important input parameters used in the transient and accident analyses for the design are verified by the ITAAC. For intersystem LOCAs, the design pressure of the piping of the systems that interface with the reactor coolant pressure boundary should be specified in the design descriptions or figures. The specific fuel, control rod, and core designs presented in Tier 2 constitute an approved design that may be used for the COL first-cycle core loading without further NRC staff review. *(Additional text follows on requirements.)* If any other core design is requested for the first cycle, the COL Applicant or licensee will be required to submit for staff review those specific fuel, control rod, and core design analyses as described in DCD Tier 2 Chapters 4, 6, and 15. Much of the detailed supporting information in Tier 2 for the nuclear fuel, fuel channel, and control rods, if considered for a change by a COL Applicant or licensee that references the certified standard design, would require prior NRC approval. Therefore, for the evolutionary and passive designs, the staff concluded that this information should be designated as Tier 2* information (see Appendix A of SRP Section 14.3 for a definition).	Conformance with no exceptions for the safety related features. ITAAC implementation is not a scope of this section.	14.3.4.4

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 20 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.4 Reactor Systems - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	However, staff will allow some of the Tier 2* designations to expire after the first full-power operation of the facility when the detailed design has been completed and the core performance characteristics are known from the startup and power-ascension test programs. The NRC bears the final responsibility for designating which material in Tier 2 is Tier 2*. The following issues are identified to ensure comprehensive and consistent treatment of Tier 1 based on the safety significance of the system being reviewed: a. System purpose and functions b. Location/functional arrangement of system c. Key design features of the system d. System operation in various modes e. Seismic and ASME code classifications f. Materials—weld quality and pressure-boundary integrity g. Controls, alarms, and displays h. Logic i. Interlocks j. Class 1E electrical power sources and divisions k. Equipment to be qualified for harsh environments l. Valve qualification and operation m. Interface requirements with other systems n. Numeric performance values (flow rates, capacities, etc.) o. Accuracy and quality of figures p. Active systems that provide defense-in-depth functions designated as non-safety systems Appendix C to SRP 14.3 provides “checklists” for the fluid systems as an aid for establishing consistency and comprehensiveness in the review of the system.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 21 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.4 Reactor Systems - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	2. The source of information used to determine safety significance of SSCs for the design of reactor and core cooling systems include applicable rules and regulations, general design criteria, unresolved safety issues, and generic safety issues, NRC generic correspondence, PRA, insights from the standard design's safety and severe accident analyses, and operating experience. Inputs from the PRA review, including shutdown safety evaluations, and severe accident analyses ensure important insights and design features from these analyses are incorporated into Tier 1. For both PRA and severe accident analyses, although large uncertainties and unknowns may be associated with the event phenomena, design features important for severe accident prevention and mitigation resulting from these analyses should be selected for treatment in Tier 1. 3. The passive-designed reactors use safety systems that employ passive means (natural forces), such as gravity, natural circulation, condensation and evaporation, and stored energy, for accident mitigation. These designs also include active systems that provide defense-in-depth capabilities for reactor-coolant makeup and decay heat removal. These active systems are the first line of defense to reduce challenges to the passive systems in the event of transients or plant upsets. SECY-95-132, "Policy and Technical Issues Associated with Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084)" provides certain guidance and positions for ensuring consistent and complete treatment of those systems that might be classified as non-safety-related by the designer or applicant but are important to safety or otherwise provide defense-in-depth functions.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 22 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.4 Reactor Systems - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	4. Applicable regulatory guidance from the Commission for selected policy and technical issues related to particular design should be followed. For the severe accident analyses, the basis for the staff's review for the evolutionary and passive standard designs was the Commission guidance related to SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements." SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs" generically presents guidance and NRC positions on evolutionary and passive LWR design certification issues. For guidance, positions, and issues related to specific designs, guidance is available in such documents as SECY-97-044, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design" or SECY-92-137, "Reviews of Inspections, Test, Analyses, and Acceptance Criteria (ITAAC) Requirements for the General Electric (GE) Advanced Boiling Water Reactor (ABWR)." Regarding DAC, SECY-02-059, "Use of Design Acceptance Criteria for the AP1000 Standard Plant Design," presents staff conclusions on acceptable use of DAC for I&C, control room, and piping design areas, contingent upon Westinghouse's and the staff's agreeing on adequate DAC during the design certification review. In SECY-92-053, "Use of Design Acceptance Criteria During 10CFRPart 52 Design Certification Process," the staff noted that DAC is defined as "a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies, in a limited number of technical areas, in making a final safety determination to support a design certification." *(Additional text follows on requirements.)*		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 23 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.4 Reactor Systems - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	5. Appendix D of SRP 14.3 lists acceptable “Standard ITAAC Entries” in the standard three-column format for ITAAC entries for configuration of systems, hydrostatic tests, net positive suction head for pumps, divisional power supply, etc., that should be contained in the overall set of ITAAC entries, as appropriate. RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition),” contains guidance for developing ITAAC assuming that a COL Applicant does not reference a certified design and/or an early site permit. Guidance in Section III for COLs referencing a certified design notes that the ITAAC contained in the certified design must apply to those portions of the facility design that have been approved. Appendix C.II.2-A provides “general ITAAC development guidance” on fluid, I&C, and electrical systems.		
14.3.5 Instrumentation and Controls - Inspections, Tests, Analyses, and Acceptance Criteria	1. The methodology for selecting SSCs that will be subject to ITAAC as well as the criteria for establishing the necessary and sufficient ITAAC should be appropriate for and consistently applied to I&C systems.	Conformance with no exceptions identified. ITAAC implementation is not a scope of this section.	14.3.4.5

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 24 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.5 Instrumentation and Controls - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	2. Tier 1 Design Descriptions (for DC and for COL referencing DC) and ITAAC Design Descriptions or ITAAC references to the FSAR (for COL not referencing DC) should describe the top-level I&C design features and performance characteristics that are significant to safety. For safety systems, this should include a description of system purpose, safety functions, equipment quality (e.g., meet the functional requirements of IEEE Std. 603-1991 and the digital system life cycle design process), equipment qualification, automatic decision-making and trip logic functions, manual initiation functions, and design features (e.g., system architecture) provided to achieve high functional reliability. The functions and characteristics of other I&C systems important to safety should also be discussed to the extent that the functions and characteristics are necessary to support remote shutdown, support required operator actions or assessment of plant conditions and safety system performance, maintain safety systems in a state that assures their availability during an accident, minimize or mitigate control system failures that would interfere with or cause unnecessary challenges to safety systems, or provide diverse back-up to protection systems. SRP Section 14.3, Appendix A, Subsection B.1, provides additional guidance on the content of Tier 1 Design Descriptions, ITAAC Design Descriptions, or ITAAC references to the FSAR. 3. ITAAC should identify the significant features of the I&C systems on which the Staff is relying to assure compliance with each NRC requirement identified in SRP Appendix 7.1-A. Tests, analyses, and acceptance criteria associated with each design commitment should, when taken together, be sufficient to provide reasonable assurance that the final as-built I&C system fulfills NRC requirements. SRP Appendix 7.1-C provides an expanded discussion of SRP acceptance criteria for safety system compliance with 10CFR50.55a(h). SRP Appendix 7.1-D further discusses SRP acceptance criteria for safety and protection systems using digital computer-based technology. SRP Section 14.3, Appendix A, Subsection B.2, provides additional guidance on the expected scope, content, and format of ITAAC.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 25 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.5 Instrumentation and Controls - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	4. For DC or for COL applications referencing a DC, Tier 1 Design Descriptions and ITAAC design commitments should be based on and consistent with the Tier 2 material. For a COL application not referencing a DC, the ITAAC Design Descriptions (if provided) and ITAAC design commitments should be based on and consistent with the FSAR portion of the application. 5. The applicant may provide design acceptance criteria (DAC) in lieu of detailed system design information. In this case, the DAC should be sufficiently detailed to provide an adequate basis for the Staff to make a final safety determination regarding the design, subject only to satisfactory design implementation and verification of the DAC by the COL Applicant or licensee. Implementation of the DAC should be verified as part of the ITAAC performed to demonstrate that the as-built facility conforms to the certified design.		
14.3.6 Electrical Systems - Inspections, Tests, Analyses, and Acceptance Criteria	The staff's review of the standard plant is conducted to ensure, in part, that Tier 1 contains top level design, fabrication, testing, and performance requirements for SSCs important to safety. Design Descriptions and ITAAC should be established to verify that these top level requirements (or design commitments) are met when the plant is built. IEEE nuclear standards should be used, as appropriate, to further establish top level requirements. IEEE Std. 308, "IEEE Standard Criteria for Class 1E power Systems for Nuclear Power Generating Stations," in conjunction with other related IEEE standards, establish specific design criteria for nuclear power plant electrical systems and equipment. The standard design Class 1E electrical systems may include: (1) the Class 1E electrical power distribution system, (2) the emergency diesel generators (EDGs), (3) the Class 1E direct current power supply, and (4) the Class 1E vital ac and Class 1E instrument and control power supplies. Using the above regulations, IEEE standards, operating experience, and PRA as its bases, the applicant should establish top-level design commitments for the Class 1E electrical systems of the standard design to be included in the design descriptions and verified by ITAAC. The top-level design commitments for the Class 1E electrical systems include design aspects related to:	Conformance with no exceptions for the safety related features. ITAAC implementation is not a scope of this section.	14.3.4.6

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 26 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.6 Electrical Systems - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	1. Equipment qualification for seismic and harsh environment To ensure that the seismic design requirements of GDC 2 and the EQ requirements of 10CFR50.49 have been adequately addressed, a "basis configuration" standard ITAAC may be established for applicable systems to verify these design aspects of electrical equipment important to safety. The Design Description should identify that Class 1E equipment is seismic Category 1 and equipment located in a harsh environment is qualified. The basic configuration standard ITAAC may be used to verify these areas. EQ of safe-shutdown equipment may be verified as part of the basic configuration ITAAC for safety related systems. EQ treatment in the ITAAC would then be discussed in the General Provisions section of Tier 1. Verification may include type tests or a combination of type tests and analyses of Class IE electrical equipment identified in the Design Description or accompanying figures to show that the equipment can withstand the conditions associated with a design basis accident without loss of safety function for the time that the function is needed. Qualification of systems and components for seismic and harsh environments should be verified by ITAAC. Electrical equipment located in a "mild" environment should be discussed in the applicable sections of the COL application only. An exception is made for state-of-the-art digital instrumentation and control (I&C) equipment and digital control and protection systems located in an "other than harsh" environment. Operational experience has shown these state-of-the-art equipment and systems to be sensitive to temperature. ITAAC should be included to verify the qualification of equipment whose performance may be impacted by sensitivity to particular environmental conditions not considered by regulations to be harsh.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 27 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.6 Electrical Systems - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	2. Redundancy and independence To ensure that the Class 1E electric systems meet the single failure requirements of GDC 17 (and other GDC), ITAAC may be established to verify the redundancy and independence of the Class 1E portion of the electrical design. For the electrical systems, ITAAC should verify the Class 1E divisional assignments and independence of electric power by both inspections and tests. The independence may be established by both electrical isolation and physical separation. Identification of the Class 1E divisional equipment should be included to aid in demonstrating the separation. (The detailed requirements are specified in Tier 2. For example, separation distances and identification are outlined in Tier 2). These attributes should be verified all the way to the electrically powered loads by a combination of the electrical system ITAAC and the ITAAC of the individual fluid, I&C, and heating, ventilation and air conditioning (HVAC) systems which also cover the electrical independence and divisional power supply requirements. ITAAC should be included to verify adequate separation, required inter-ties (if any), required identification (e.g., color coding), proper routing/termination (i.e., location), separation of non-Class 1E loads from 1E buses. Post-fire safe shutdown separation of electrical circuits should be addressed in the fire protection system ITAAC.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 28 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.6 Electrical Systems - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	3. Capacity and Capability To ensure that the electrical systems have the capacity and capability to supply the safety-related electrical loads, ITAAC should be established to verify the adequate sizing of the electrical system equipment and its ability to respond (e.g., automatically in the times needed to support the accident analyses) to postulated events. This includes the Class 1E portion and the non-Class 1E portion to the extent that it is involved in supporting the Class 1E system. ITAAC should be included to analyze the as-built electrical system and installed equipment (diesel generators, transformers, switchgear, batteries, etc.) to verify its ability to power the loads. In addition, the ITAAC should also include tests to demonstrate the operation of the equipment. Testing should be included in ITAAC to verify EDG capacity and capability based on the Technical Specifications. In some cases regulatory guidance specifies the need for margin in capacity to allow for future load growth. If it is only for future load growth, ITAAC does not need to check for the additional margin. ITAAC should be developed to verify the initiation of the Class 1E equipment necessary to mitigate postulated events for which the equipment is credited (e.g., loss of coolant accident (LOCA), loss of offsite power (LOOP), and degraded voltage conditions). ITAAC should be included to analyze the as-built electrical power system for its response to a LOCA, LOOP, combinations of LOCA and LOOP (including LOCA with delayed LOOP and LOOP with delayed LOCA), and degraded voltage, including tests to demonstrate the actuation of the electrical equipment in response to postulated events. Analyses to demonstrate the acceptability of a voltage drop should be included in ITAAC to verify adequacy for supporting the accomplishment of a direct safety function. The applicable section of the COL application should include a discussion of how the voltage analyses will be performed, i.e., reference to industry standards. Testing should be included in ITAAC to verify the EDG voltage and frequency response is acceptable and is the same as that specified in the Technical Specifications.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 29 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.6 Electrical Systems - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	4. Electrical protection features To ensure that the electrical power system is protected against potential electrical faults, ITAAC should be established to verify the adequacy of the electrical circuit protection included in the design. Operating experience and NRC Electrical Distribution System Functional Inspections (EDSFIs) have indicated some problems with the short circuit rating of some electrical equipment and breaker and protective device coordination. Inclusion in ITAAC should be based on the potential for preventing safety functions and the operating experience. ITAAC should be included to analyze the as-built electrical system equipment for its ability to withstand and clear electrical faults. ITAAC should also be included to analyze the protection feature coordination to verify its ability to limit the loss of equipment due to postulated faults. Equipment short circuit capability and breaker coordination should be verified by specifying ITAAC for analyses. The description of the analyses should be included in the applicable section of the application. Similarly, diesel generator protective trips (and bypasses if applicable) should be considered.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 30 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.6 Electrical Systems - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	5. Displays/controls/alarms To help ensure that the electrical power system is available when required, ITAAC should be included to verify the existence of monitoring and controls for the electrical equipment. The minimum set of displays, alarms, and controls is based on the emergency procedure guidelines. In some cases, additional displays, alarms, and controls may be specified based on special considerations in the design and/or operating experience. ITAAC should be included to inspect for the ability to retrieve the information (displays and alarms), and to control the electrical power system in the main control room and/or at locations provided for remote shutdown. Detection of undervoltage conditions along with the starting and loading of EDG should be included in ITAAC. This is a direct safety function in response to design basis event of loss of power. Problems with relay settings should be considered in this requirement. Other Electrical Equipment Important to Safety In addition to the Class 1E systems addressed above, other aspects of the electrical design that are deemed to be important to safety and the top-level design commitments are included in Tier 1.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 31 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.6 Electrical Systems - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	1. Offsite Power To ensure that the requirements of GDC 17 for the adequacy and independence of the preferred offsite power sources within the standard design scope were met, ITAAC should verify the capacity and capability of the offsite sources to feed the Class 1E divisions, and the independence of those sources. ITAAC should be included to inspect the direct connection of the offsite sources to the Class 1E divisions and to inspect for the independence/separation of the offsite sources. ITAAC should be developed to inspect for appropriate lightning protection and grounding features. In addition, the Design Description includes "interface" requirements for the portions of the offsite power outside of the standard design scope; however, no ITAAC are included for the interfaces. The interfaces define the requirements that the offsite portion of the design (that is out-of-scope) must meet to support and not degrade the in-scope design (See also Appendix A to SRP Section 14.3). 2. Containment Electrical Penetrations To ensure the containment electrical penetrations (both those containing Class 1E circuits and those containing Non Class 1E circuits) do not fail due to electrical faults and potentially breach the containment, ITAAC should verify that all electrical containment penetrations are protected against postulated currents greater than their continuous current rating. 3. AAC Power Source (if applicable) To ensure the availability of the alternate AC (AAC) power source for station blackout events, ITAAC should be developed to verify, through inspection and testing, the AAC power source (combustion gas turbines, diesel generators, or hydro units) and its auxiliaries along with its independence from other ac/AC sources.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 32 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.6 Electrical Systems - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	4. Lighting ITAAC should be included to verify the continuity of power sources for plant lighting systems to ensure that portions of the plant lighting remain available during accident scenarios and power failures. The basis for inclusion may be more related to defense-in-depth, support function, operating experience, or PRA rather than "accomplishing a direct safety function." 5. Electrical Power For Non-Safety Plant Systems To ensure that electrical power is provided to support the non-safety plant systems, Design Descriptions cover portions of the non-Class 1E electrical systems. ITAAC should be included to verify the functional arrangement of electrical power systems provided to support non-safety plant systems to the extent that those systems perform a significant safety function.		
14.3.7 Plant Systems - Inspections, Tests, Analyses, and Acceptance Criteria	1. The reviewer should utilize the SRP in its review of Tier 1 to determine the safety significance of SSCs. Other sources include applicable rules and regulations, GDCs, RGs, USIs and GSIs, NRC generic correspondence, PRA, insights from the standard design's safety and severe accident analyses, and operating experience. Tier 1 should be reviewed for consistency with the initial test program described in DCD Tier 2 Chapter 14.2. The reviewer should also use the review checklists provided in Appendix C to SRP Section 14.3 as an aid for establishing consistency and comprehensiveness in his review of the systems. If applicable, the reviewer should utilize regulatory guidance from the Commission for selected policy and technical issues related to particular design. Examples of these are contained in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs." The SRM related to this is dated July 21, 1993. 2. Tier 1 should be reviewed for treatment of design information proportional to the safety significance of the SSC for that system. Many items may be judged to be important to safety, and thus should be included in Tier 1. The following issues are identified to ensure comprehensive and consistent treatment in Tier 1 based on the safety significance of the system being reviewed:	Conformance with no exceptions for the safety related features. ITAAC implementation is not a scope of this section.	14.3.4.7

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 33 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.7 Plant Systems - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	(1) System purpose and functions (2) Location of system (3) Key design features of the system (4) Seismic and ASME code classifications (5) System operation in various modes (6) Controls, alarms, and displays (7) Logic (8) Interlocks (9) Class 1E electrical power sources and divisions (10) Equipment to be qualified for harsh environments (11) Interface requirements (12) Numeric performance values (13) Accuracy and quality of figures 3. Standard ITAAC entries should be utilized to verify selected issues, where appropriate. The reviewer should ensure consistent application and treatment of the standard ITAAC entries for basic configuration ITAAC, net positive suction head, and physical separation for appropriate systems in Tier 1. In particular, the general provision for environmental qualification aspects of SSCs invoked by the basic configuration ITAAC should be reviewed to ensure appropriate treatment in Tier 1. 4. Environmental qualification (EQ) of safe-shutdown equipment may be verified as part of the basic configuration ITAAC for safety-related systems. EQ treatment in the ITAAC would then be discussed in the General Provisions section of Tier 1. Verification may include type tests or a combination of type tests and analyses of Class 1E electrical equipment identified in the Design Description or accompanying figures to show that the equipment can withstand the conditions associated with a design basis accident without loss of safety function for the time that the function is needed.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 34 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.7 Plant Systems - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	5. The design features in Tier 1 should be selected to ensure that the integrity of the analyses are preserved in an as-built facility. For example, 3-hour fire boundaries and divisional separation may be shown in the building figures. Also, flooding features such as structure elevations should be specified in the site parameters, flood doors may be shown on the building figures, and elevations are shown on the buildings to verify that the approximate physical location of components and relative elevations of buildings minimize the effects of flooding. As-built reconciliation reports for fires and floods to ensure consistency with Tier 2 analyses should be required by the appropriate system ITAAC (e.g., fire protection system) and selected building ITAAC, respectively. 6. Other specific issues that should be addressed include heat removal capabilities for design-basis accidents and tornado and missile protection. Heat removal capabilities may be verified through heat removal requirements for core cooling system heat exchangers and interface requirements for site-specific systems. Tornado and missile protection may be provided by inlet and outlet dampers in ventilation systems, and through the structural design of buildings.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 35 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.7 Plant Systems - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	7. The areas of review for radioactive waste systems include design objectives, design criteria, identification of all expected releases of radioactive effluents, methods of treatment, methods used in calculating effluent source terms and releases of radioactive materials in the environment, and operational programs in controlling and monitoring effluent releases and for assessing associated doses to members of the public. The radioactive waste systems include the liquid waste management system (LWMS), gaseous waste management system (GWMS), and the solid waste management system (SWMS). These systems deal with the management of radioactive wastes, as liquid, wet, and dry solids, produced during normal operation and anticipated operational occurrences. In addition, the review includes an evaluation of the process and effluent radiological monitoring instrumentation and sampling systems (PERMISS) which are used to monitor liquid and gaseous process streams and effluents and solid wastes generated by these systems. The PERMISS includes subsystems used to collect process and effluent samples during normal operation, anticipated operational occurrences, and under post-accident conditions. The lead branch responsible in implementing the review should coordinate the review of these systems and operational programs and receive input on the design and compliance with acceptance criteria listed in SRP Sections 11.2 to 11.5 from other branches, including, balance of plant, structural, instrumentation and controls, HVAC, quality assurance, technical specifications, and emergency planning.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 36 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.7 Plant Systems - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	8. The reviewer should receive inputs on the treatment of issues identified above from other branches such as the structural, electrical and I&C branches. In addition, the secondary review branches specified in SRP Section 14.3 should provide inputs on selected issues. These issues include key insights and assumptions from PRA and severe accident analyses, as well as inputs for issues such as treatment of alarms, displays and controls, and functionality of MOVs. Cross-references from Tier 2 to Tier 1 for key insights and assumptions from PRA and severe accidents should be provided by applicants in Tier 2 together with these analyses. 9. Tier 1 should address and verify at least the minimum inventory of alarms, controls, and indications as derived from the Emergency Procedure Guidelines, the requirements of RG 1.97, and probabilistic risk assessment insights. These may be specified in the MCR and the Remote Shutdown System (RSS) ITAAC, or addressed in the appropriate ITAAC, and verified to exist. Other controls, indications and alarms should be identified in the system ITAAC based on their safety significance. Locations for these should be shown on system figures if important to system design and function. The ability of these controls, indications, and alarms to function should be checked during operation of the system for the functional tests required by the system ITAAC. Because the intent of the ITAAC is to verify the final as-built condition of the plant, the operation of the system during the completion of the functional tests required in the system ITAAC should be conducted from the MCR. Therefore, the verification that the system can be operated from the MCR need not be a separate ITAAC. Also, because the operation of the equipment from the control room demonstrates the control function, continuity checks between the RSS and the equipment demonstrates that the control signal will be received by the component and provides adequate assurance that the equipment can be operated by the RSS. The results of the pre-operational test program may be utilized to demonstrate the ability to operate plant equipment by the RSS.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 37 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.8 Radiation Protection Inspections, Tests, Analyses, and Acceptance Criteria	1. The reviewer should primarily use the applicable rules and regulations, general design criteria, regulatory guides, unresolved safety issues, and generic safety issues in the review of Tier 1 to determine the safety significance of SSCs with respect to the radiation protection for occupational workers and the general public they provide. Other sources include the SRP and applicable U.S. Nuclear Regulatory Commission (NRC) generic correspondence. The reviewer should use the guidance in Appendix C to SRP Section 14.3 as an aid for ensuring the comprehensiveness and consistency of this review. 2. Radiation Protection: The reviewer should ensure that Tier 1 identifies and describes, commensurate with their safety significance, those SSCs that provide radiation shielding, confinement or containment of radioactivity, ventilation of airborne contamination, or radiation (or radioactivity concentration) monitoring for normal operations and during accidents. Tier 1 identifies and describes the measures that need to be employed during first-of-a-kind engineering to ensure that final design details (i.e., materials and component selection, equipment placement, and pipe routing) are consistent with the radiation protection commitments (including the commitment that radiation exposures will be ALARA) in the certified design. Tier 1 contains ITAAC that ensure that the identified SSCs will function in a manner consistent with the certified design.	Conformance with no exceptions for the safety related features. ITAAC implementation is not a scope of this section.	14.3.4.8

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 38 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.8 Radiation Protection Inspections, Tests, Analyses, and Acceptance Criteria (continued)	3. Design Processes and Design Acceptance Criteria: A DC applicant may not provide sufficient detail in selected aspects of the design, including sufficient information to stipulate the source terms needed to verify the design of the shielding, ventilation, and airborne radioactivity monitoring systems. The applicant may choose to provide design processes and DAC for this material, as discussed in Appendix A to SRP Section 14.3. The applicant should document in DCD Tier 2, Section 14.3, its rationale for determining which areas of the design should use design processes and acceptance criteria. Essentially, the applicant should extract the most important design processes and acceptance criteria from DCD Chapter 12 of Tier 2 and identify them in Tier 1. This may be done either in a separate section of Tier 1 or in the applicable systems of Tier 1. A COL Applicant or licensee must meet these criteria in the design of the plant, and the staff can audit the facility's design documentation to ensure that the criteria are met. The following discussion is specific to the review of design processes and acceptance criteria in this area. DC applicants may not provide the complete design information in this design area before the design is certified because the radiation shielding design and the calculated concentrations of airborne radioactive material depend on as-built and as-procured information about plant systems and components. Therefore, applicants may be unable to describe the standard design's radiation source terms (i.e., the quantity and concentration of radioactive materials contained in, or leaking from, plant systems) in sufficient detail to allow the staff to verify the adequacy of the shielding design, ventilation system designs, or the design and placement of the airborne radioactivity monitors. Instead, applicants may provide the processes and acceptance criteria by which the details of the design in this area are to be developed, designed, and evaluated. *(Additional text follows on requirements.)*		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 39 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.9 Human Factors Engineering - Inspections, Tests, Analyses, and Acceptance Criteria	1. SRP Chapter 18 provides guidance for the NRC staff to use in determining whether an applicant has proposed an acceptable HFE design. The applicant's HFE program will be evaluated in accordance with the review criteria of SRP Chapter 18 and NUREG-0711, "Human Factors Engineering Program Review Model." As indicated in Chapter 18, the HFE program technical information for the DC or COL review may be based on a design and implementation process plan. Therefore, the DC or COL ITAAC may be based on a design and implementation process plan. For example, acceptance criteria for the task analysis program element may be stated as "a report exists and concludes that function-based task analyses were conducted in conformance with the task analysis implementation plan and include the following functions . . ." 2. If an implementation plan, rather than a completed HFE element, was accepted as part of the design certification process, then ITAAC should address the completion of the HFE program element. 3. If an implementation plan was not reviewed and approved as part of the design certification, then the ITAAC should address both the development of the plan as well as item 2 above. 4. The reviewer will verify that HFE-related ITAAC information is provided based on accepted HFE principles and program elements as discussed in SRP Chapter 18 and incorporated into the plant's design. 5. HFE-related ITAAC should primarily address verification of products (e.g., the control room, the human-system interfaces, etc.) or results reports from implementing the HFE program element implementation plan.	Conformance with no exceptions identified. ITAAC implementation is not a scope of this section.	14.3.4.9

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 40 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.9 Human Factors Engineering - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	6. Minimum Inventory of Displays, Alarms and Controls: Tier 1 includes a minimum inventory of displays, controls, and alarms that are necessary to carry out the vendor's emergency procedure guidelines (i.e., Owners' Groups Generic Technical Guidelines) and critical actions identified from the applicant's PRA and task analysis of operator actions. The reviewers evaluation of the minimum inventory will encompass a multi-disciplinary effort consisting of human factors, I&C, PRA, and plant, reactor, and electrical system engineering. The minimum inventory list has been implemented through the rule-making process for four certified designs (10CFRPart 52 Appendixes A, B, C, and D). The criteria used to determine acceptability of the inventory includes assuring that: (1) the scope of these items in the Generic Technical Guidelines and PRA effort are adequately considered, (2) the task analysis is detailed and comprehensive, (3) RG 1.97, Revision 3, Category 1 variables or RG 1.97, Revision 4, Type A, B, and C variables for accident monitoring are included, and (4) important system displays and controls described in Tier 1 system design descriptions necessary for transient mitigation are included.		
14.3.10 Initial Test Program and D-RAP - Inspections, Tests, Analyses, and Acceptance Criteria	1. The reviewer should ensure that for a design certification where an applicant has chosen to address emergency response facilities that the information provided adequately discusses facilities for emergency response. These include a habitable technical support center (TSC) with space, data retrieval capabilities and dedicated communications equipment, and an operational support center (OSC) with adequate communications, consistent with the applicable criteria in Supplement 1 to NUREG-0737 and NUREG-0696.	Conformance with no exceptions identified. ITAAC implementation is not a scope of this section.	14.3.4.10

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 41 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.10 Initial Test Program and D-RAP - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	2. A generic set of acceptable emergency planning EP-ITAAC was developed through coordination efforts between the NRC and the Nuclear Energy Institute (NEI) and resulted in the development of generic EP-ITAAC that are provided in Table 14.3.10-1(Table C.II.2-B11 of RG 1.206). These EP-ITAAC were established on a generic basis; they are not associated with any particular site or design. As such, several of the generic EP-ITAAC require the COL Applicant to provide more specific acceptance criteria that reflect the plant-specific design and site-specific emergency response plans and facilities. This generic set is applicable to ESP applications that include ITAAC information. The reviewer should consider this set of EP-ITAAC in the review of application-specific EP-ITAAC that is tailored to the specific reactor design and emergency planning program requirements for the proposed plant and site. A smaller set of EP-ITAAC is acceptable if the application contains information that fully addresses emergency preparedness requirements associated with any of the generic ITAAC contained in Table 14.3.10-1 which is not all-inclusive, or exclusive of other ITAAC an applicant may propose. Additional plant-specific EP-ITAAC (i.e., beyond those listed in Table 14.3.10-1) may be proposed, and they will be examined to determine their acceptability on an applicant-specific basis. Table 14.3.10-1 also includes ITAAC associated with emergency response facilities that are within the scope of the design certification. COL applications referencing a certified design must include these design certification ITAAC on emergency response facilities. EP-ITAAC are proposed by the COL Applicant and, except for EP-ITAAC from the referenced design certification or ESP, are subject to NRC review and a hearing with respect to whether they satisfy the “necessary and sufficient” requirement of 10CFR52.80(a). The complete set of EP-ITAAC will be incorporated into the COL as a license condition to be satisfied prior to fuel load. A COL holder may request a change in one or more of the EP-ITAAC, except those provided in the referenced certified design, via the license amendment process applicable to 10CFRPart 52.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 42 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.11 Containment Systems and Severe Accidents - Inspections, Tests, Analyses, and Acceptance Criteria	1. The reviewer should primarily utilize the SRP sections related to containment systems in its review of Tier 1 to determine the safety significance of SSCs. Other sources include applicable rules and regulations, GDCs, RGs, USIs and GSIs, NRC generic correspondence, PRA, insights from the standard design's safety and severe accident analyses, and operating experience. The reviewer should also use the review checklists provided in Appendix C to SRP Section 14.3 as an aid for establishing consistency and comprehensiveness in the review of the systems. 2. Tier 1 should be reviewed to verify that key parameters and insights from containment safety analyses, such as loss of coolant accident, main steamline break, main feedline break, subcompartment analyses, and suppression pool bypass are adequately addressed. Applicants should provide cross references in DCD Tier 2 Section 14.3 to show how the important input parameters used in the transient and accident analyses for the design are verified by the ITAAC. The reviewer should ensure that appropriate treatment of severe accident design features and containment design features are included in Tier 1. The supporting information regarding the detailed design and analyses should remain in Tier 2. For many of the design features, it may be impractical to test their functionality because of the absence of simulated severe accident conditions. Consequently, the existence of the feature on a figure, subject to a basic configuration walkdown, may be considered sufficient Tier 1 treatment. Applicants should provide cross references in the appropriate sections of Tier 2 to show how the important parameters from PRA, including shutdown risk, and severe accident analyses are verified by the ITAAC. For both PRA and severe accident analyses, although large uncertainties and unknowns may be associated with the event phenomena, design features important for severe accident prevention and mitigation resulting from these analyses should be selected for treatment in Tier 1.	Conformance with no exceptions for the safety related features. ITAAC implementation is not a scope of this section.	14.3.4.11

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 43 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.11 Containment Systems and Severe Accidents - Inspections, Tests, Analyses, and Acceptance Criteria (continued)	3. If applicable, the reviewer should utilize regulatory guidance from the Commission for selected policy and technical issues related to the particular design. Examples of these are contained in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs." The SRM related to this is dated July 21, 1993. 4. Containment isolation may be addressed by a combination of the system ITAACs or in a single system ITAAC. The containment isolation valves (CIVs) should be specified in Tier 1, and are most clearly shown on the system figures. The verification of the design qualification of the motor operated CIVs may be verified by the basic configuration check in each system ITAAC. In addition, in-situ tests should be required for containment isolation motor operated valves (MOV) and check valves in each system ITAAC. The ITAAC should verify that the CIVs close on receipt of an isolation signal. Actual closure of the containment isolation valves may be checked using the manual isolation switches in the main control room (MCR). Other ITAAC may verify that a containment isolation signal is generated for each of the process variables that will cause a containment isolation; the intent is to preclude multiple cycling of the containment isolation valves during the testing. 5. Tier 1 should address and verify at least the minimum inventory of alarms, displays, and controls in Design Control Document (DCD) Tier 2 Chapter 18. These are derived from Generic Technical Guidelines (e.g., Emergency Procedure Guidelines, Emergency Response Guidelines), the guidance of RG 1.97, and severe accident and PRA insights. They may be specified in the MCR and the Remote Shutdown System (RSS) ITAAC, or addressed in the appropriate ITAAC, and are verified to exist. Other controls, displays, and alarms should be identified in the system ITAAC based on their safety significance. Locations for these should be shown on system figures if important to system design and function.		

Table 1.9.2-14 US-APWR Conformance with Standard Review Plan Chapter 14 Verification Programs (sheet 44 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
14.3.12 Physical Security Hardware - Inspections, Tests, Analyses, and Acceptance Criteria	1. Appendix A to this SRP section provides an acceptable set of generic PS-ITAAC that an applicant may use to develop application-specific PS-ITAAC, tailored to specific physical security hardware requirements. Appendix A is not all-inclusive, or exclusive of other PS-ITAAC an applicant may propose. Additional plant-specific PS-ITAAC (i.e., beyond those listed in Appendix A) may be proposed and will be examined to determine their acceptability on a case-by-case basis. (Appendix A is a table that presents 7 pages of requirements.)	Conformance with no exceptions identified. ITAAC implementation is not a scope of this section.	14.3.4.12

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 1 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.0 Introduction - Transient and Accident Analyses	Subsection I.2 of this SRP section discusses general acceptance criteria, and SRP Chapter 15 subsections discuss specific acceptance criteria for transients or accidents. Note: Section I.2 provides general acceptance criteria on the following topics: A. Analysis Acceptance Criteria for Anticipated Operational Occurrences (AOOs). B. Analysis Acceptance Criteria for Postulated Accidents. C. Core, System, and Barrier Performance	Conformance with no exceptions identified.	15.0.0.1
15.0.1 Radiological Consequence Analyses Using Alternative Source Terms	This SRP applies to operating plants that are changing their accident analysis source term inputs, and is not applicable to the US-APWR design certification.	Not applicable. SRP applies to operating plants adopting alternative source term inputs.	N/A
15.0.2 Review of Transient and Accident Analysis Methods	The acceptance criteria are based on meeting the requirements of the regulations in 10CFRPart 50 that govern the evaluation models for the specific accident under consideration (e.g., 10CFR50.46 for a LOCA). The following sections discuss the specific criteria. 1. Documentation The submittal must identify the specific accident scenarios and plant configurations for which the codes will be used. The evaluation model documentation must be scrutable, complete, unambiguous, accurate, and reasonably self-contained. Consistent nomenclature must be used throughout the entire model documentation. Any referenced material must be readily available from a technical library. Copies of any referenced documents that are not readily obtainable from a technical library or the NRC Public Document Room, including proprietary reports, must be included with the documentation or provided upon request.	Conformance with no exceptions identified.	15.0.2

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 2 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.0.2 Review of Transient and Accident Analysis Methods (continued)	The code documentation must be sufficiently detailed that a qualified engineer can understand the documentation without recourse to the originator as required of any design calculation that meets the design control requirements of Appendix B to 10CFRPart 50, and the documentation requirement in Appendix K to 10CFRPart 50. It is desirable that the documentation include the responses to requests for additional information, sorted according to the review issue so that it is easy to follow the entire review history for a single issue. The reviewer can help obtain this goal by issuing RAI's organized by review issue. The documentation must include the following components: A. An overview of the evaluation model, which provides a clear roadmap describing all parts of the evaluation model, the relationships between them, and where they are located in the documentation. B. A complete description of the accident scenario including plant initial conditions, the initiating event and all subsequent events and phases of the accident, and the important physical phenomena and systems and/or component interactions that influence the outcome of the accident. C. A complete description of the code assessment comprising a description of each assessment test, why it was chosen, success criteria, diagrams of the test facility that show the location of instrumentation that is used in the assessment, a code model nodalization diagram, and all code options used in the calculation. D. A determination of the code uncertainty for a sample plant accident calculation. (Appendix K models do not require a determination of the code uncertainty.)		

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 3 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.0.2 Review of Transient and Accident Analysis Methods (continued)	E. A theory manual that is a self-contained document and that describes (a) field equations, (b) closure relationships, (c) numerical solution techniques, (d) simplifications and approximations (including limitations) inherent in the chosen field equations and numerical methods, (e) pedigree or origin of closure relationships used in the code, and (f) limits of applicability for all models in the code. F. A user manual that provides (a) detailed instructions about how the computer code is used, (b) a description of how to choose model input parameters and appropriate code options, (c) guidance about code limitations and options that should be avoided for particular accidents, components, or reactor types, and (d) if multiple computer codes are used, documented procedures for ensuring complete and accurate transfer of information between different elements of the evaluation model .G. A quality assurance plan that describes the procedures and controls under which the code was developed and assessed, and the corrective action procedures that are followed when an error is discovered. (Additional sections of text follow on requirements for Evaluation Model, Accident Scenario Identification Process, Code Assessment, Uncertainty Analysis, and Quality Assurance Plan.)		

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 4 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.0.3 Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors	Excerpt from the "Acceptance Criteria" section of this SRP: 1. Offsite Radiological Consequences of Postulated Design Basis Accidents The acceptance criteria are based on the requirements of 10CFR50.34(a)(1) as related to mitigating the radiological consequences of an accident in accordance with 10CFR52.17(a)(1) [early site permits], 10CFR52.47(a)(1) [standard design certifications] and 10CFR52.79(b) [combined licenses]. The plant design features intended to mitigate the radiological consequences of accidents, site atmospheric dispersion characteristics and the distances to the exclusion area boundary (EAB) and to the low population zone (LPZ) outer boundary are acceptable if the total calculated radiological consequences for the postulated fission product release fall within the following exposure acceptance criteria specified in 10CFR50.34(a)(1)(ii)(D): A. An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE), and B. An individual located at any point on the outer boundary of the LPZ, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 25 rem TEDE... (Followed by additional information on requirements)	Conformance with exceptions. Full conformance by COL applicant with site specific meteorological information.	15.0.3, 15.1.5.5, 15.3.3.5, 15.4.8.5, 15.6.2, 15.6.3.5, 15.6.5.5, 15.7.4

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 5 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.0.3 Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors (continued)	2. Control Room Radiological Habitability The acceptance criterion is based on the requirements of GDC 19 that mandate a control room design providing adequate radiation protection to permit access and occupancy of the control room under accident conditions for the duration of the accident, without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. These requirements are incorporated by reference in 10CFR52.47(a)(1) [standard design certifications] and 10CFR52.79(b) [combined licenses]. The radiation protection design of the control room is acceptable if the total calculated radiological consequences for the postulated fission product release fall within the exposure acceptance criteria specified in GDC 19 of 5 rem TEDE for the duration of the accident. 3. Technical Support Center Radiological Habitability This acceptance criterion is based on the requirement of Paragraph IV.E.8 of Appendix E to 10CFRPart 50 to provide an onsite TSC from which effective direction can be given and effective control can be exercised during an emergency. The radiation protection design of the TSC is acceptable if the total calculated radiological consequences for the postulated fission product release fall within the exposure acceptance criteria specified for the control room of 5 rem TEDE for the duration of the accident.		

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 6 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
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	1. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values. 2. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs and the CPR remains above the MCPR safety limit for BWRs based on acceptable correlations (see SRP Section 4.4). 3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently. 4. To meet the requirements of General Design Criteria 10, 13, 15, 20, and 26 the positions of Regulatory Guide 1.105, "Instrument Spans and Setpoints," are used with regard to their impact on the plant response to the type of transient addressed in this SRP section. 5. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10CFRPart 50, shall be identified and assumed in the analysis and shall satisfy the positions of Regulatory Guide 1.53.	Conformance with no exceptions identified.	15.1.1, 15.1.2, 15.1.3, 15.1.4

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 7 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
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 (continued)	The applicant's analysis of transients caused by excessive heat removal should be performed using an acceptable analytical model, the NRC approved methodologies and the computer codes. If the applicant proposes to use analytical methods which have not been approved, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer performs an evaluation based on SRP section 15.0.2, “Transient and Accident Analysis methods.” The values of the parameters used in the analytical model should be suitably conservative. The following values are considered acceptable for use in the model: 1. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event. 2. Conservative scram characteristics are assumed, i.e., for a PWR - maximum time delay with the most reactive rod held out of the core, and for a BWR - a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate, unless (a) a different conservatism factor can be justified through the uncertainty methodology and evaluation, or (b) the uncertainty has otherwise been accounted for (see SAR or DCD) Section 4.4. 3. The core burn-up is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution. 4. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with Regulatory Guide 1.105. Compliance with Regulatory Guide 1.105 is determined by the Instrumentation and Control Systems.		

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 8 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.1.5 Steam System Piping Failures Inside and Outside of Containment (PWR)	1. Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, considering potential brittle as well as ductile failures. 2. The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit for PWRs based on acceptable correlations (see SRP Section 4.4). If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability. 3. The radiological criteria used in the evaluation of steam system pipe break accidents (PWRs only) appear in SRP section 15.0.3. 4. The integrity of the reactor coolant pumps should be maintained such that loss of ac power and containment isolation will not result in pump seal damage. 5. The auxiliary feedwater system or other means of decay heat removal must be safety related and, when required, automatically initiated. In the case of AP1000 the PRHR provides the safety related means of decay heat removal. 6. Tripping of the reactor coolant pumps should be consistent with the resolution to Task Action Plan item II.K.3.5. There are certain assumptions regarding important parameters used to describe the initial plant conditions and postulated system failures which should be used. These are listed below:	Conformance with no exceptions identified.	15.1.5

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 9 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.1.5 Steam System Piping Failures Inside and Outside of Containment (PWR) (continued)	1. The reactor power level and number of operating loops assumed at the initiation of the transient should correspond to the operating condition which maximizes the consequences of the accident. These assumed initial conditions will vary with the particular NSSS design, and sensitivity studies will be required to determine the most conservative combination of power level and plant operating mode. These sensitivity studies may be presented in a generic report and referenced in the SAR. 2. Assumptions as to the loss of offsite power and the time of loss should be made to study their effects on the consequences of the accident. A loss of offsite power may occur simultaneously with the pipe break or during the accident, or offsite power may not be lost. Analyses should be made to determine the most conservative assumption appropriate to the particular plant design. The reviewer should note that the assumption that offsite power is not lost may maximize heat removal from the core and reactor core. The analyses should take account of the effect that loss of offsite power has on reactor coolant pump and main feedwater pump trips and on the initiation of auxiliary feedwater flow, and the effects on the sequence of events for these accidents. For new applications, loss of offsite power should be considered in addition to any limiting single active failure. (This position is based upon interpretation of GDC 17, as documented in the Final Safety Analysis Report for the ABB-CE System 80+ design certification.) 3. The effects (pipe whip, jet impingement, reaction forces, temperature, humidity, etc.) of postulated steam line breaks on other systems should be considered in a manner consistent with the intent of Branch Technical Position (BTP) 3-3 and BTP 3-4. 4. The worst single active component failure should be assumed to occur. For new applications, loss of offsite power should not be considered as a single failure, (see assumption b above). The assumed single failure may cause more than one steam generator to blow down, failure of main feedwater to isolate, or may be in any of the systems required to control the transient.		

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 10 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.1.5 Steam System Piping Failures Inside and Outside of Containment (PWR) (continued)	5. The maximum-worth rod should be assumed to be held in the fully withdrawn position. An appropriate rod reactivity worth versus rod position curve should be used. Local power peaking at the location of the stuck out control rod should be considered. Local power peaking will affect the DNBR analysis in the initial period as the safety rods are entering the core and during any subsequent return to power resulting from reactivity addition to the core from the cooldown. 6. The core burnup (time in core life) should be selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution. 7. The initial core flow assumed for the analysis of the steam line rupture accident should be chosen conservatively. If the minimum core flow allowed by the technical specifications is assumed, the minimum DNBR margin results; however, for the analysis of steam line break accidents, this may not be the most conservative assumption. For example, maximum initial core flow results in increased reactor coolant system cooldown and depressurization, decreased shutdown margin, and an increased possibility that the core will become critical and return to power. Since it is not clear what initial core flow is most conservative, the assumed value should be justified. 8. Failure of a steam line at a plant with multiple coolant loops will cause asymmetric temperatures within the reactor core. Asymmetric core temperatures will affect the local power distribution and the DNBR analysis. Assumptions for mixing in the downcomer and the reactor vessel lower plenum will affect the predicted core temperature distributions, reactivity feedback and local power. Assumptions for mixing should be chosen so as to be conservative for predicting maximum local core power and DNBR.		

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 11 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.1.5 Steam System Piping Failures Inside and Outside of Containment (PWR) (continued)	9. For postulated pipe failure in nonseismically qualified portions of the main steam line (outside containment and downstream of the main steam isolation valves, (MSIVs) due to a seismically initiated event, only safety related equipment should be assumed operative to mitigate the consequences of the break. 10. For postulated instantaneous pipe failures in seismically qualified portions of the main steam line (inside containment and upstream of the MSIVs), only safety related equipment should be assumed operative. If, in addition, a single malfunction or failure of an active component is postulated, credit may be taken for the use of a backup nonsafety-related component to mitigate the consequences of the break. 11. During the initial 10 minutes of the transient, should credit for operator action be required (e.g., reactor coolant pump trip), an assessment for the limiting consequence must be performed in order to account for operator delay and/or error.		
15.1.5.A Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR	The acceptance criteria are based on the relevant requirements of 10CFRPart 100 as related to the radiological consequences of a postulated accident. The plant site and the dose mitigating engineered safety features are acceptable with respect to the radiological consequences of a postulated MSLB outside containment of a PWR facility if the calculated whole-body and thyroid doses at the exclusion area and the low population zone outer boundaries do not exceed the following exposure guidelines: 1. for an MSLB with an assumed pre-accident iodine spike and for an MSLB with the highest worth control rod stuck out of the core, the calculated doses should not exceed the guideline values of 10CFRPart 100, Section 11 (Ref. 1), and	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.	N/A

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 12 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.1.5.A Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR (continued)	2. for an MSLB with the equilibrium iodine concentration for continued full power operation in combination with an assumed accident initiated iodine spike, the calculated doses should not exceed a small fraction of the above guideline values, i.e., 10 percent or 2.5 rem and 30 rem respectively, for the whole-body and thyroid doses. The methodology and assumptions for calculating the radiological consequences should reflect the regulatory positions of Regulatory Guide 1.4 (Ref. 8) except for the atmospheric dispersion factors which are reviewed under SRP Section 2.3.4. Plant technical specifications are required for the iodine activity in the primary and secondary coolant system and for the leak rate from the primary to the secondary coolant system in the steam generator(s). These specifications are acceptable if the calculated potential radiological consequences from the MSLB accident are within the exposure guidelines for the above two cases.		
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)	1. The basic objectives of the review of the initiating events listed in subsection I of this SRP section: A. To identify which moderate-frequency event that results in an unplanned decrease in secondary system heat removal is the most limiting, in particular as to primary pressure, secondary pressure, and long-term decay heat removal. B. To verify whether the predicted plant response for the most limiting event satisfies the specific criteria for fuel damage and system pressure. C. To verify whether the plant protection systems setpoints assumed in the transients analyses are selected with adequate allowance for measurement inaccuracies as delineated in RG 1.105. D. To verify whether the event evaluation considers single failures, operator errors, and performance of nonsafety-related systems consistent with the RG 1.206 regulatory guidelines.	Conformance with no exceptions identified.	15.2.1, 15.2.2, 15.2.3, 15.2.4, 15.2.5

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 13 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
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) (continued)	2. With the ANS standards as guidance, specific criteria meet the relevant requirements of GDCs 10, 13, 15, 17, and 26 for events of moderate frequency. A. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values. B. Fuel cladding integrity must be maintained by the minimum departure from nucleate boiling ratio (DNBR) remaining above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) remaining above the minimum CPR safety limit for BWRs based on acceptable correlations (see SAR (or DCD) Section 4.4) and by satisfaction of any other SAFDL applicable to the particular reactor design. C. An incident of moderate frequency should not generate an aggravated plant condition without other faults occurring independently. D. The requirements in RG 1.105, "Instrument Spans and Setpoints," are used for their impact on the plant response to the type of AOOs addressed in this SRP section. E. The most limiting plant system single failure, as defined in "Definitions and Explanations," 10CFRPart 50, Appendix A, must be assumed in the analysis according to the guidance of RG 1.53 and GDC 17. F. Performance of nonsafety-related systems during transients and accidents and single failures of active and passive systems (especially as to the performance of check valves in passive systems) must be evaluated and verified according to the guidance of SECY 77-439, SECY 94-084, and RG 1.206 3. The applicant should analyze these events using an acceptable analytical model. Any other analytical method proposed by the applicant is evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by the appropriate organization for reactor systems. The values of the parameters in the analytical model should be suitably conservative. The following values are acceptable:		

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 14 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
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) (continued)	A. The reactor is initially at 102 percent of the rated (licensed) core thermal power (to account for a 2 percent power measurement uncertainty unless a lower number can be justified through measurement uncertainty methodology and evaluation or unless the uncertainty otherwise is accounted for (see SAR (or DCD) Section 4.4)), and primary loop flow is at the nominal design flow less the flow measurement uncertainty. B. Conservative scram characteristics are assumed (*i.e.*, for a PWR maximum time delay with the most reactive rod held out of the core, for a BWR a 0.8 design conservatism multiplier on the predicted reactivity insertion rate) unless (i) a different conservatism factor can be justified through the uncertainty methodology and evaluation or (ii) the uncertainty is otherwise accounted for (see SAR (or DCD) Section 4.4). C. The core burn-up is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution. D. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with Regulatory Guide 1.105.		
15.2.6 Loss of Non-Emergency ac/AC Power to the Station Auxiliaries	Specific criteria necessary to meet the relevant requirements of GDCs 10, 13, 15, and 26 for events of moderate frequency (see definitions of design and plant process conditions in are as follow: 1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values. 2. Fuel cladding integrity should be maintained by keeping the minimum departure from nucleate boiling ratio (DNBR) above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) above the minimum critical power ratio safety limit for BWRs based on acceptable correlations (see SRP Section 4.4). 3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.	Conformance with no exceptions identified.	15.2.6

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 15 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.2.6 Loss of Non-Emergency ac/AC Power to the Station Auxiliaries (continued)	4. For the requirements of GDCs 10 and 15, the positions of Regulatory Guide (RG) 1.105, "Instrument Setpoints for Safety Related Systems," have impact on the plant response to the type of transient addressed in this SRP section. 5. The most limiting plant system single failure, as defined in the "Definitions and Explanations" of 10CFRPart 50, Appendix A, must be assumed in the analysis and must satisfy the positions of RG 1.53. The applicant's analysis of the loss of ac power transient should be based on an acceptable and NRC-approved model. If the applicant proposes analytical methods not approved, these are evaluated by the staff for acceptability and approval. For new generic methods, the reviewer requests an appropriate evaluation. The parameter values in the analytical model should be suitably conservative. The following values are acceptable: A. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2 percent to account for power measurement uncertainties unless the applicant can justify a lower power level. The number of loops (RCS loop requirements as applicable for BWR design) operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event. B. Conservative scram characteristics are assumed (i.e., for a PWR maximum time delay with the most reactive rod held out of the core and for a BWR a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate). C. The core burn-up is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, power profile, and radial power distribution. D. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with RG 1.105. Compliance with RG 1.105 is determined.		

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 16 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.2.7 Loss of Normal Feedwater Flow	1. The basic objective in the review of the loss of normal feedwater transient is to confirm that the following criteria are met: A. The plant responds to the loss of feedwater transient in such a way that the criteria regarding fuel damage and system pressure are met. B. There is sufficient capacity for long term decay heat removal for the plant to reach a stabilized condition. C. The plant protection systems setpoints assumed in the transient analyses are selected with adequate allowance for measurement uncertainties as delineated in Regulatory Guide 1.105. D. The event evaluation takes into consideration single failures, operator errors, and performance of non-safety related systems that are consistent with regulatory guidelines set forth in RG 1.206. 2. Using the ANS standards as guidance, specific criteria have been developed to meet the relevant requirements of GDCs 10, 13, 15, 17, and 26 for events of moderate frequency and they are as follows: A. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values. B. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs, and the CPR remains above the MCPR safety limit for BWRs based on acceptable correlations (see SAR (or DCD) Section 4.4), as well as by satisfaction of any other SAFDL that may be applicable to the particular reactor design. C. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently. D. To meet the requirements of GDCs 10 and 15, the positions of Regulatory Guide 1.105, "Instrument Spans and Setpoints," are used with regard to their impact on the plant response to the type of transient addressed in this SRP section.	Conformance with no exceptions identified.	15.2.7

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 17 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.2.7 Loss of Normal Feedwater Flow (continued)	E. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10CFRPart 50, shall be identified and assumed in the analysis and shall satisfy the positions of Regulatory Guide 1.53 and GDC 17. F. The guidance provided in SECY 77-439, SECY 94-084 and RG 1.206 with respect to the consideration of the performance of non-safety related systems during transients and accidents, as well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems) must be evaluated and verified. 3. The applicant's analysis of the loss of normal feedwater transient should be performed using an acceptable analytical model. If the applicant proposes to use analytical methods which have not been approved, these methods are evaluated by the staff for acceptability. For new generic methods the reviewer requests an evaluation by the appropriate organization for reactor systems. The value of parameters used in the analytical model should be suitably conservative. The following values are considered acceptable for use in the model. A. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event. B. Conservative scram characteristics are assumed, i.e., for a PWR – maximum time delay with the most reactive rod held out of the core and for a BWR – a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate, unless (a) a different conservatism factor can be justified through the uncertainty methodology and evaluation, or (b) the uncertainty has otherwise been accounted for (see SAR (or DCD) Section 4.4).		

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 18 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.2.7 Loss of Normal Feedwater Flow (continued)	C. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, power profile and radial power distribution. D. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with Regulatory Guide 1.105.		
15.2.8 Feedwater System Pipe Breaks Inside and Outside Containment (PWR)	1. Requirements for maintenance of adequate decay heat removal by the AFWS are in 10CFR50.34(f)(1)(ii), (TMI issue II E 1.1) and 10CFR50.34(f)(2)(xii), (TMI issue II E 1.2). Requirements for reactor coolant pump (RCP) operation are in 10CFR50.34(f)(1)(iii), (TMI issue 2 K 2). The reviewer should see Chapter 20 of the NRC FSAR for AP1000 to see how these post TMI requirements are met by the PRHR, the non-safety related start-up feedwater system (SUFWS) and the canned-motor RCPs of AP1000. 2. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressures (American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III) for low-probability events and below 120 percent for very low-probability events like double-ended guillotine breaks. 3. The potential for core damage is evaluated for an acceptable minimum DNBR remaining above the 95/95 DNBR limit for pressurized-water reactors (PWRs) based on acceptable correlations (see SRP Section 4.4). If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods not meeting these criteria unless, from an acceptable fuel damage model (see SRP Section 4.2) including the potential adverse effects of hydraulic instabilities, fewer failures can be shown to occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core remains in place and intact with no loss of core cooling capability. 4. Calculated doses at the site boundary from any activity release must be a small fraction of the 10CFRPart 100 guidelines.	Conformance with no exceptions identified.	15.2.8

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 19 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.2.8 Feedwater System Pipe Breaks Inside and Outside Containment (PWR) (continued)	5. The integrity of the RCPs should be maintained so loss of ac/AC power and containment isolation do not result in seal damage. 6. The AFWS must be safety grade and automatically initiated when required. 7. Certain assumptions should be in the analysis of important parameters that describe initial plant conditions and postulated system failures: A. The power level assumed and number of loops operating at the initiation of the transient should correspond to the operating condition which maximizes accident consequences. These assumed initial conditions vary with the particular nuclear steam supply system and sensitivity studies are required to determine the most conservative combination of power level and plant operating mode. These sensitivity studies may be presented in a generic report as references if applicable. B. The assumptions as to whether offsite power is lost and the time of loss should be conservative. Offsite power may be lost simultaneously with the pipe break, the loss may occur during the accident, or offsite power may not be lost. A study should determine the most conservative assumption appropriate to the plant design reviewed. The study should take account of the effects that loss of offsite power (LOOP) has on reactor coolant and main feedwater pump trips and on the initiation of auxiliary feedwater and the consequent modification of the sequence of events. C. The effects (pipe whip, jet impingement, reaction forces, temperature, humidity, etc.) of the postulated feedwater line breaks on other systems should be considered consistently with the intent of Branch Technical Positions (BTP) 3-3 and BTP 3-4.		

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 20 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.2.8 Feedwater System Pipe Breaks Inside and Outside Containment (PWR) (continued)	D. The worst single active component failure should be assumed to occur in the systems required to control the transient. For new applications, LOOP should not be considered a single failure; feedwater pipe breaks should be analyzed with and without LOOP, as in assumption B, in combination with a single, active failure. (This position is based upon interpretation of GDC 17 as documented in the FSER for the ABB-CE System 80+ DC.) E. The maximum rod worth should be assumed to be held in the fully withdrawn position per GDC 25. An appropriate rod reactivity worth versus rod position curve should be assumed. F. The core burn-up (time in core life) should be selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution. G. The initial core flow assumed for the analysis of the feedwater line rupture accident should be chosen conservatively. If the minimum core flow allowed by the technical specifications is assumed, the minimum DNBR margin is the result for a feedwater line rupture inside containment; however, this assumption may not be the most conservative. For example, maximum initial core flow increases RCS cool-down and depressurization, decreases shutdown margin, and increases the possibility that the core will become critical and return to power. As it is not clear which initial core flow is most conservative, the applicant's assumption should be justified by appropriate sensitivity studies. H. During the initial 10 minutes of the transient, if credit for operator action is required (i.e., RCP trip), an assessment for the limiting consequence must account for operator delay and/or error.		

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 21 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.3.1-15.3.2 Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	The basic objectives of the review of loss of forced reactor coolant flow transients are to identify the most limiting transients and to verify whether, for the most limiting transients, the plant response to the loss of flow transients satisfies fuel damage and system pressure criteria. The following specific criteria are necessary to meet the regulatory requirements for incidents of moderate frequency: 1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values. 2. Fuel-cladding integrity must be maintained by the minimum DNBR remaining above the 95 percent probability/95 percent confidence DNBR limit for PWRs and the critical power ratio (CPR) remaining above the minimum critical power ratio (MCPR) safety limit for BWRs based on acceptable correlations (see SRP Section 4.4). 3. An incident of moderate frequency should not generate an aggravated plant condition without other faults occurring independently. 4. The requirements stated in RG 1.105, "Instrument Spans and Setpoints," are evaluated for their impact on the plant response to AOOs addressed in this SRP section. 5. Onsite and offsite electric power systems must be maintained so safety-related SSCs function during normal operation and AOOs. 6. The most limiting plant system single failure, as defined in the "Definitions and Explanations" of 10CFR50, Appendix A, must be assumed in the analysis and should follow the guidance of RG 1.53. 7. The performance of nonsafety-related systems during transients and accidents and of single failures of active and passive systems (especially the performance of check valves in passive systems), must be evaluated and verified by the guidance of SECY 77-439, SECY 94-084 and RG 1.206. 8. The applicant's analysis of the most limiting AOOs should use an acceptable model. Unapproved analytical methods proposed by the applicant are evaluated by the staff for acceptability.	Conformance with no exceptions identified (15.3.1) 15.3.2 is applicable only to BWRs	15.3.1, 15.3.2

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 22 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.3.1-15.3.2 Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions (continued)	9. Parameter values in the analytical model should be suitably conservative. The following values are acceptable: A. Initial power level is rated output (licensed core thermal power) for the number of loops initially assumed operating plus an allowance of 2 percent to account for power measurement uncertainty unless (i) a lower number can be justified through the measurement uncertainty methodology and evaluation or (ii) the uncertainty is accounted for otherwise (see SRP 4.4). The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event. B. Conservative scram characteristics are assumed (e.g., maximum time delay with the most reactive rod held out of the core for a PWR, a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate for a BWR), unless (i) a different conservatism factor can be justified through the uncertainty methodology and evaluation or (ii) the uncertainty is accounted for otherwise (see SRP Section 4.4). C. The core burn-up is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution. D. Mitigating systems should be assumed as actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with RG 1.105 and as determined by the organization responsible for instrumentation and controls.		

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 23 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.3.3-15.3.4 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	The specific criteria necessary to meet the relevant requirements of General Design Criteria 27, 28, and 31 and 10CFRPart 100 for the rotor seizure and shaft break event are: 1. Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, considering potential brittle as well as ductile failures. 2. The potential for core damage is evaluated on the basis that it is acceptable if the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) remains above the minimum critical power ratio (MCPR) safety limit for BWRs based on acceptable correlations (see SRP Section 4.4). If the DNBR or CPR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. If rod internal pressure exceeds system pressure, then fuel rods may balloon shortly after entering DNB. The effect of ballooning fuel rods must be evaluated with respect to flow blockage and DNB propagation. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability. 3. Any release of radioactive material must be such that the calculated doses at the site boundary are a small fraction of the 10CFRPart 100 guidelines. 4. The integrity of the reactor coolant pumps should be maintained such that loss of ac power and containment isolation will not result in pump seal damage. 5. The auxiliary feedwater system must be safety grade and, when required, automatically initiated. 6. A rotor seizure or shaft break in a reactor coolant pump should not, by itself, generate a more serious condition or result in a loss of function of the reactor coolant system or containment barriers.	Conformance with no exceptions identified.	15.3.3, 15.3.4

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 24 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.3.3-15.3.4 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break (continued)	7. Only safety-grade equipment should be used to mitigate the consequences of the event. Safety functions should be accomplished assuming the worst single failure of a safety system active component. For new applications, loss of offsite power should not be considered a single failure; reactor coolant pump rotor seizures and shaft breaks should be analyzed with a loss of off-site power (see item 9, below) in combination with a single active failure. (This position is based upon interpretation of GDC 17, as documented in the Final Safety Evaluation Report for the ABB-CE System 80+ design certification.) 8. The ability to achieve and maintain long-term core cooling should be verified. 9. This event should be analyzed assuming turbine trip and coincident loss of offsite power and coastdown of undamaged pumps. The applicant's analysis should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 8 through 12 are acceptable. The NRC staff found References 13 and 14 to be acceptable transient analysis computer codes for design analysis of the Advanced Boiling Water Reactor (ABWR). References 15 through 19 were found to be acceptable computer codes for transient analyses (i.e., except for loss-of-coolant accidents, or LOCAs) for the Combustion Engineering System 80+ final safety evaluation report staff review. In addition, NUREG-1465 contains guidance on accident source terms for light-water nuclear power plants. When conducting transient analyses, the NUREG-1465 guidance is particularly important for reviewing fractions of relevant isotopes (noble gases, iodine, cesium, and rubidium) and chemical species of iodine assumed to exist within the gap between fuel pellets and cladding. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability.		

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 25 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.3.3-15.3.4 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break (continued)	For new generic methods, the reviewer requests an evaluation. There are certain assumptions regarding important parameters used to describe the initial plant conditions and postulated system failures which should be used. These are listed below: 1. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating, plus an allowance to account for power measurement uncertainties. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event. 2. The local flow conditions used in the core thermal-hydraulics model should be calculated based upon an inlet flow distribution corresponding to N–1 reactor coolant pumps (initial minus faulted pump) and a conservative time-dependent flow coastdown. Note that the inlet flow distribution will change as more pumps begin to coastdown following turbine trip and coincident loss of offsite power. 3. Conservative scram characteristics are assumed, i.e., for a PWR maximum time delay with the most reactive rod held out of the core, and for a BWR a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate. 4. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.		
15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	1. The requirements of GDC 10, 20, and 25 concerning the specified acceptable fuel design limits are assumed to be met for this event when: A. The thermal margin limits (DNBR for PWRs and MCPR for BWRs) as specified in SRP Section 4.4 are met. B. Fuel centerline temperatures (for PWRs) as specified in SRP Section 4.2 do not exceed the melting point. C. Uniform cladding strain (for BWRs) as specified in SRP Section 4.2 does not exceed 1%.	Conformance with no exceptions identified.	15.4.1

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 26 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power	1. The requirements of General Design Criteria 10, 17, 20, and 25 concerning the specified acceptable fuel design limits are assumed to be met for this event when: A. The thermal margin limits departure from nucleate boiling ratio for PWRs and maximum critical power ratio for BWRs as specified in SRP Section 4.4, subsection II.1, are met. B. Fuel centerline temperatures (for PWRs) as specified in SRP Section 4.2, subsection II.A.2(a) and (b), do not exceed the melting point. C. Uniform cladding strain (for BWRs) as specified in SRP Section 4.2, subsection II.A.2(b), does not exceed 1%.	Conformance with no exceptions identified.	15.4.2
15.4.3 Control Rod Misoperation (System Malfunction or Operator Error)	The requirements of General Design Criteria 10, 20, and 25 concerning the specified acceptable fuel design limits are assumed to be met for this event when: 1. The thermal margin limits (departure from nucleate boiling ratio for PWRs) as specified in SRP Section 4.4, subsection II.1, are met. 2. Fuel centerline temperatures as specified in SRP Section 4.2, subsection II.A.2(a) and (b), do not exceed the melting point. 3. Uniform cladding strain as specified in SRP Section 4.2, subsection II.A.2(b), does not exceed 1%.	Conformance with no exceptions identified.	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	Using the ANS standards as guidance, the specific criteria necessary to meet the relevant requirements of the regulations identified above for incidents of moderate frequency are as follows: A. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values. B. Fuel-cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95% probability/95% confidence DNBR limit for PWRs and the critical power ratio (CPR) remains above the minimum critical power ratio (MCPR) safety limit for BWRs, based on acceptable correlations (see SRP Section 4.4). C. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.	Conformance with exception. MHI's position is that the withdrawal of a single RCCA cannot occur due to any single equipment failure and therefore, this event is not an AOO. As a result, a very limited amount of consequential fuel damage may occur , mitigated by its low expected frequency. The other two events (one or more dropped RCCAs in a bank or group and one or more misaligned RCCAs relative to their bank) meet the AOO acceptance criteria in the SRP.	15.4.3

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 27 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
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 (continued)	D. The requirements stated in Regulatory Guide 1.105, "Instrument Spans and Setpoints," are used with regard to their impact on the plant response to the type of AOOs addressed in this SRP section. E. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10CFR50, shall be identified and assumed in the analysis and should satisfy the guidance stated in Regulatory Guide 1.53. F. The guidance provided in SECY 77-439, SECY 94-084 and RG 1.206 with respect to the consideration of the performance of non-safety related systems during transients and accidents, as well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems), must be evaluated and verified. The applicant's analysis of the most limiting AOOs should be performed using an acceptable model. If analytical methods that have not been approved are proposed by the applicant, they are evaluated by the staff for acceptability. The values of parameters used in the analytical model are to be suitably conservative. The following values are considered acceptable: 1. Initial power level is rated output (licensed core thermal power) for the number of loops initially assumed to be operating, plus an allowance of 2% to account for power measurement uncertainty, unless (a) a lower number can be justified through the measurement uncertainty methodology and evaluation, or (b) unless the uncertainty has otherwise been accounted for (see SRP Section 4.4). An analysis to determine the effects of a flow increase must be made for each allowed mode of operation (i.e., one, two or three loops initially operating) or the effects referenced to a limiting case. 2. Conservative scram characteristics are assumed, e.g., maximum time delay with the most reactive rod held out of the core for a PWR and a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate for a BWR, unless (a) a different conservatism factor can be justified through the uncertainty methodology and evaluation, or (b) unless the uncertainty has otherwise been accounted for (see SRP Section 4.4).	Not applicable. Events do not apply to US-APWR because: a) Tech Specs do not allow power operations with an inactive loop, and b) flow controller event is applicable to BWRs only.	N/A (discussed in 15.4.4 and 15.4.5

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 28 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
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 (continued)	3. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile and radial power distribution. 4. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with Regulatory Guide 1.105 as determined by the organization responsible for instrumentation and controls. The reviewer shall verify that the protection system (1) automatically initiates the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded for this event, and (2) senses the plant conditions and initiates the operation of SSCs important to safety.		
15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant (PWR)	1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values. 2. Fuel cladding integrity must be maintained so the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for pressurized-water reactors (PWRs) based on acceptable correlations with SRP Section 4.4. 3. An incident of moderate frequency should not generate a more serious than moderate plant condition without other faults occurring independently. 4. If operator action is required to terminate the transient, the following minimum time intervals must be available between the time an alarm announces an unplanned moderator dilution and the time shutdown margin is lost: A. During refueling: 30 minutes. B. During startup, cold shutdown, hot shutdown, hot standby, and power operation: 15 minutes.	Conformance with no exceptions identified.	15.4.6

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 29 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant (PWR) (continued)	5. The applicant's analysis of moderator dilution events should use an acceptable analytical model. Staff must evaluate any proposed unreviewed analytical methods. The reviewer initiates an evaluation of new generic methods. The following plant initial conditions should be considered in the analysis: refueling, startup, power operation (automatic control and manual modes), hot standby, hot shutdown and cold shutdown. Parameters and assumptions in the analytical model should be suitably conservative. The following values and assumptions are acceptable: A. For analyses during power operation, the initial power level is rated output (licensed core thermal power) plus an allowance of 2 percent to account for power-measurement uncertainty. The analysis may use a smaller power-measurement uncertainty if justified adequately. B. The boron dilution is assumed to occur at the maximum possible rate. C. Core burnup and corresponding boron concentration must yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution. The core burnup must be justified by either analysis or evaluation. D. All fuel assemblies are installed in the core. E. A conservatively low value is assumed for the reactor coolant volume. F. For analyses during refueling, all control rods are withdrawn from the core. An alternate assumption requires adequate justification and delineation of necessary controls so the alternate assumption remains valid. G. For analyses during power operation, the minimum shutdown margin allowed by the technical specifications (usually 1 percent) is assumed prior to boron dilution.		

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 30 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant (PWR) (continued)	H. A conservatively high reactivity addition rate is assumed for each analyzed event to take into account the effect of increasing boron worth with dilution. I. Conservative scram characteristics are assumed (*i.e.*, maximum time delay with the most reactive rod out of the core).		
15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	The primary safeguards against fuel-loading errors are procedures and design features to minimize the likelihood of the event. Additional safeguards include incore instrumentation systems which would detect errors. However, should an error be made and go undetected, it is possible in some reactor designs for fuel rod failure limits to be exceeded. Therefore, the following acceptance criteria cover the event of operation with misloaded fuel caused by loading errors: 1. To meet the requirements of GDC 13, plant operating procedures should include a provision requiring that reactor instrumentation be used to search for potential fuel-loading errors after fueling operations. 2. In the event the error is not detectable by the instrumentation system and fuel rod failure limits could be exceeded during normal operation, the offsite consequences should be a small fraction of the 10CFRPart 100 criteria. A small fraction is interpreted to be less than 10% of the 10CFRPart 100 reference values. For the purpose of this review, the radiological consequences of any fuel-loading error should include consideration of the containment, confinement, and filtering systems. The applicant's source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing should reflect NRC-approved source terms and methodologies.	Conformance with no exceptions identified.	15.4.7

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 31 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.4.8 Spectrum of Rod Ejection Accidents (PWR)	Other SRP sections interface with this section as follows: 1. General information on transient and accident analyses is provided in SRP Section 15.0. 2. Design basis radiological consequence analyses associated with design basis accidents are reviewed under SRP Section 15.0.3. 3. Reactivity coefficients and control rod worths are reviewed under SRP Section 4.3. 4. Relevant thermal-hydraulic analyses are reviewed under SRP Section 4.4. 5. The applicant's determination of the reactor trip delay time (i.e., the time elapsed between when the sensed parameter reaches the level for which protective action is required and the onset of negative reactivity insertion) is reviewed under SRP Sections 7.2 and 7.3. The specific acceptance criteria and review procedures are contained in the referenced SRP sections. Note: As indicated above, there are no unique acceptance criteria established in this SRP.	Conformance with no exceptions identified. (based on interfacing SRPs)	15.4.8
15.4.8.A Radiological Consequences of a Control Rod Ejection Accident (PWR)	The acceptance criteria are based on requirements of 10CFRPart 100 as to mitigating the radiological consequences of an accident. The plant site and dose mitigating engineered safety features are acceptable with respect to the radiological consequences of a postulated control rod ejection accident if the calculated whole-body and thyroid doses at the exclusion area (EAB) and the low population zone (LPZ)-boundaries are well within the exposure guideline values specified in 10CFRPart 100, paragraph 11 (Ref. 1). Well within is defined as 25% of-the 10CFRPart 100 exposure guideline values or 75 rem for the thyroid and 6 rem for whole-body doses. A technical specification is required for the leak rate from the primary to secondary coolant system in the steam generators. This specification is acceptable if the calculated potential radiological consequences from the control rod ejection accident are within the exposure guidelines above.	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.	N/A

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 32 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.4.8.A Radiological Consequences of a Control Rod Ejection Accident (PWR) (continued)	The models for calculating the whole-body and thyroid doses are acceptable if they incorporate the appropriate conservative design basis assumptions outlined in Appendix B to Regulatory Guide 1.77 (Ref. 2) with the exception of the guidelines for the atmospheric dispersion factors (X/Q values). The acceptability of the X/Q values is determined under SRP Section 2.3.4.		
15.4.9 Spectrum of Rod Drop Accidents (BWR)	This SRP is written for boiling water reactors (BWRs) and is not applicable to the US-APWR.	Not applicable. SRP applies to BWRs only.	N/A
15.4.9.A Radiological Consequences of Control Rod Drop Accident (BWR)	This SRP is written for boiling water reactors (BWRs) and is not applicable to the US-APWR.	Not applicable. SRP applies to BWRs only.	N/A
15.5.1-15.5.2 Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	This event is an AOO, as defined in 10CFR50, Appendix A. Acceptance criteria for AOOs are specified in SRP 15.0. The specific acceptance criteria derived from GDC 10, 13, 15, and 26, and from the aforementioned ANS standards, are: 1. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values in accordance with the ASME Boiler and Pressure Vessel Code. 2. Fuel cladding integrity should be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) remains above the minimum critical power ratio (MCPR) safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).	Conformance with no exceptions identified.	15.5.1, 15.5.2

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 33 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.5.1-15.5.2 Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory (continued)	3. An AOO should not generate a more serious plant condition without other faults occurring independently. The applicant's analysis of events leading to an increase of reactor coolant inventory should be performed using an acceptable analytical model. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer performs an evaluation of the new method as part of its review under this SRP section. The values of parameters used in the analytical model should be suitably conservative. The following values are considered acceptable for use in the model: 1. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event. 2. Conservative scram characteristics are assumed, i.e., for a PWR maximum time delay with the most reactive rod held out of the core and for a BWR a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate. 3. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.		
15.6.1 Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve	1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values. 2. Fuel cladding integrity is maintained if the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) above the minimum critical power ratio safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).	Conformance with no exceptions identified.	15.6.1

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 34 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.6.1 Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve (continued)	3. An AOO should not develop into a more serious plant condition without other faults occurring independently. Satisfaction of this criterion precludes the possibility of a more serious event during the lifetime of the plant. To meet the requirements of GDCs 10, 13, 15, and 26, the positions of Regulatory Guide (RG) 1.105, "Instrument Setpoints for Safety-Related Systems," are useful as to their impact on the plant response to the type of transient addressed in this SRP section. The most limiting plant system single failure, as defined in the "Definitions and Explanations" of 10CFRPart 50, Appendix A, should be assumed in the analysis and should satisfy the positions of RG 1.53. The applicant's analysis of this transient should use an acceptable analytical model. If the applicant proposes to use analytical methods not previously reviewed and approved by the staff, the staff evaluates them for acceptability. For new generic methods, the reviewer initiates an evaluation of the new analytical model. The values of the parameters in the analytical model should be suitably conservative. The following values are acceptable. A. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to operate plus an allowance of 2 percent to account for power measurement uncertainties unless the applicant can justify a lower power level. The number of loops operating at the initiation of the event should correspond to the operating condition that maximizes the consequences of the event. B. Conservative scram characteristics are assumed (i.e., for a PWR maximum time delay with the most reactive rod held out of the core and for a BWR a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate). C. The core burn-up is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution. D. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with RG 1.105.		

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 35 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.6.2 Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	The acceptance criteria for this SRP section are based on the relevant requirements of the following regulations: 1. General Design Criterion 55 (Ref. 1) as it relates to the identification of small diameter lines connected to the primary system that are exempted from the isolation requirements of GDC 55 and that are acceptable on the basis of meeting item (2) below, 2. 10CFRPart 100, §100.11 (Ref. 3) as it relates to the radiological consequences of a small line break carrying primary coolant outside containment. The plant site and the dose mitigating engineered safety feature (ESF) systems are acceptable with respect to the radiological consequences of a postulated failure outside the containment of a small line carrying reactor coolant if the calculated whole-body and thyroid doses at the exclusion area and the low population zone outer boundaries do not exceed a small fraction of the exposure guideline values of 10CFRPart 100, §100.11 (Ref. 3) as stated in position C.1.b of Regulatory Guide 1.11 (Ref. 2). A "small fraction" of 10CFRPart 100 means 10 percent of these exposure guideline values, that is, 2.5 rem and 30 rem for the whole-body and thyroid doses, respectively. A plant-specific technical specification is required for the iodine activity in the primary coolant system. The specification is acceptable with respect to the postulated failure if the calculated doses resulting from the failure are within the above exposure guidelines.	Conformance with no exceptions identified.	15.6.2
15.6.3 Radiological Consequences of Steam Generator Tube Failure (PWR)	The acceptance criteria are based on the relevant requirements of 10CFRPart 100 as it relates to mitigating the radiological consequences of an accident. The plant site and the dose mitigating engineered safety features are acceptable with respect to the radiological consequences of a postulated steam generator tube failure accident at a PWR facility if the calculated whole-body and thyroid doses at the exclusion area and the low population zone outer boundaries do not exceed the following exposure guidelines: (1) for the postulated accident with an assumed preaccident iodine spike in the reactor coolant and for the postulated accident with the highest worth control rod stuck out of the core the calculated doses should not exceed the guideline values of 10CFRPart 100, Section 11 (Ref. 1), and	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.	N/A

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 36 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.6.3 Radiological Consequences of Steam Generator Tube Failure (PWR) (continued)	(2) for the postulated accident with the equilibrium iodine concentration for continued full power operation in combination with an assumed accident initiated iodine spike, the calculated doses should not exceed a small fraction of the above guideline values, i.e., 10 percent or 2.5 rem and 30 rem, respectively, for the whole-body and thyroid doses. The methodology and assumptions for calculating the radiological consequences should reflect the regulatory positions of Regulatory Guide 1.4 (Ref. 2) except for the atmospheric dispersion factors which are reviewed under SRP Section 2.3.4. Plant technical specifications are required for iodine activity in the primary and secondary coolant systems. These specifications are acceptable if the calculated potential radiological consequences from the steam generator tube failure accident are within the exposure guidelines for the above two cases.		
15.6.4 Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)	This SRP is written for boiling water reactors (BWRs) and is not applicable to the US-APWR.	Not applicable.	N/A
15.6.5 Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	Specific criteria necessary to meet the relevant requirements of the regulations identified above and necessary to meet the TMI Action Plan requirements are as follows: 1. An evaluation of ECCS performance has been performed by the applicant in accordance with an evaluation model that satisfies the requirements of 10CFR50.46. Regulatory Guide 1.157 and Section I of Appendix K to 10CFRPart 50 provide guidance on acceptable evaluation models. For the full spectrum of reactor coolant pipe breaks, and taking into consideration requirements for reactor coolant pump operation during a small break loss-of-coolant accident, the results of the evaluation must show that the specific requirements of the acceptance criteria for ECCS are satisfied as given below. This also includes analyses of a spectrum of large break and small break LOCAs to assure boric acid precipitation is precluded for all break sizes and locations. The analyses should be performed in accordance with 10CFR50.46, including methods referred to in 10CFR50.46(a)(1) or (2).	Conformance with no exceptions identified.	15.6.5

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 37 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.6.5 Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary (continued)	The analyses must demonstrate sufficient redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities such that the safety functions could be accomplished assuming a single failure in conjunction with the availability of onsite power (assuming offsite electric power is not available, with onsite electric power available; or assuming onsite electric power is not available with offsite electric power available). Additionally the LOCA methodology used and the LOCA analyses should be shown to apply to the individual plant by satisfying 10CFR50.46(c)(2), and the analysis results should meet the performance criteria in 10CFR50.46(b). A. The calculated maximum fuel element cladding temperature does not exceed 1200 oC (2200 oF). B. The calculated total local oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation. Total local oxidation includes pre-accident oxidation as well as oxidation that occurs during the course of the accident. C. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react. D. Calculated changes in core geometry are such that the core remains amenable to cooling. E. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity. 2. The radiological consequences of the most severe LOCA are within the guidelines of and 10CFR100 or 10CFR50.67. For applications under 10CFRPart 52, reviewers should use SRP Section 15.0.3, “Radiological Consequences of Design Basis Accidents - for ESP, DC and COL Applications.” 3. The TMI Action Plan requirements for II.E.2.3, II.K.2.8, II.K.3.5, II.K.3.25, II.K.3.30, II.K.3.31, and II.K.3.40 have been met.		

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 38 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.6.5.A Radiological Consequences of a Design Basis Loss-of-Coolant Accident Including Containment Leakage Contribution	The acceptance criteria are based on the requirements of 10CFRPart 100 as related to mitigating the radiological consequences of an accident. Specific acceptance criteria for the total calculated. doses and for the containment leakage contribution are as follows: 1. The distances to the exclusion area boundary and to the low population zone outer boundary are acceptable if the total calculated radiological consequences (i.e.,.thyroid and whole body doses) for the hypothetical LOCA fall within the appropriate exposure guideline values specified In 10CFRPart 100, §100.11 (Ref. 1). The total dose is the combined dose from all release paths from the containment to the atmosphere. At the construction permit (CP) review stage, the staff applies exposure guideline values of 150 rem to the thyroid and 20 rem to the whole body in accordance with Regulatory Guides 1.3 and 1.4. This is to allow for uncertainties in meteorology and other site-related data and to allow for system design changes that might influence the final design of engineered safety features or the dose reduction factors of these features. These lower values are applied at the CP stage to provide reasonable assurance that the 10CFRPart 100 guideline values can be met at the operating license (OL) review stage. 2. The model for and the calculation of the post-LOCA leakage contribution to the total whole body and thyroid doses of a hypothetical LOCA are acceptable if they incorporate the appropriate conservative design basis assumptions outlined in the regulatory positions of Regulatory Guide 1.3 (Ref. 2) for a BWk facility and of Regulatory Guide 1.4 (Ref. 3) for a PWR facility with the exception of the guidelines for the atmospheric dispersion fusion factors (X/Q values). The acceptability of the X/Q values is determined under SRP Section 2.3.4.	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.	N/A

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 39 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.6.5.B Radiological Consequences of a Design Basis Loss-of-Coolant Accident Leakage From Engineered Safety Feature Components Outside Containment	The acceptance criteria are based on the requirements of 10CFRPart 100 (Ref. 2) as related to mitigating the radiological consequences of an accident. Specific criteria necessary to meet this requirement are as follows: (1) ESF systems that circulate water outside the containment are assumed to leak during their intended operation (e.g., valve stem leakage) and as a result of a failure of a passive component; Both types of leakage are included in the review. ESF atmosphere filtration systems should be provided in those areas where such leakage is postulated to occur in order to mitigate the radiological consequences from the fission product release. (2) The radiological consequences from the postulated leakage should be calculated using conservative assumptions. 50% of the core iodine inventory, based upon the maximum reactor power level, should be assumed to be mixed in the sump water being circulated through the containment external piping systems, in accordance with the values listed in Table 1 of Regulatory Guide 1.7 (Ref. 1). The atmospheric dispersion factors (X/Q values) as determined under SRP Section 2.3.4 should be used in the analysis. (3) The radiological consequences from ESF component leakage, as calculated by the staff, should be combined, under SRP Section 15.6.5 Appendix A, with the consequences from other fission product release paths to determine the total calculated radiological consequences from the hypothetical LOCA. The acceptability of the site, with respect to the total radiological consequences, is determined by the adequacy of the exclusion area and low population zone outer boundary distances in conjunction with the operation of dose-mitigating ESF systems. For operating license applications, the total doses should be within the exposure guidelines of 10CFRPart, 100, § 100.11 (Ref. 2) and for a construction permit application, the total doses should be within the guideline value of Regulatory Guides 1.3 (Ref. 3) and 1.4 (Ref. 4), as appropriate. This acceptability is determined under SRP Section 15.6.5, Appendix A.	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.	N/A

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 40 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.6.5.D Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Main Steam Isolation Valve Leakage Control System (BWR)	This SRP is written for boiling water reactors (BWRs) and is not applicable to the US-APWR.	Not applicable. SRP applies to BWRs only.	N/A
15.7.3 Postulated Radioactive Releases Due to Liquid-Containing Tank Failures (content of this section has been relocated to BTP 11-6)	ETSB acceptance criteria are based on meeting the relevant requirements of the following regulations: 1. General Design Criterion 60 as it relates to the radioactive waste management systems being designed to control releases of radioactive materials to the environment. 2. 10CFRPart 20 as it relates to radioactivity in effluents to unrestricted areas. Tanks and associated components containing radioactive liquids outside containment are acceptable if failure does not result in radionuclide concentrations in excess of the limits in 10CFRPart 20, Appendix B, Table II, Column 2, at the nearest potable water supply,* in an unrestricted area, or if special design features are provided to mitigate the effects of postulated failures for systems not meeting these limits.	Not applicable. BTP 11-6, "Postulated Radioactive Release Due to Liquid-containing Tank Failures" is applied instead of SRP 15.7.3.	N/A
15.7.4 Radiological Consequences of Fuel Handling Accidents	The AEB acceptance criteria for this SRP section are based on requirements of 10CFRPart 100 (Ref. 1) with respect to the calculated radiological consequences of a fuel handling accident and General Design Criterion 61 (Ref. 2) with respect to appropriate containment, confinement, and filtering systems. Specific criteria necessary to meet the requirements are: 1. The plant site and dose mitigating ESF systems are acceptable with respect to the radiological consequences of a postulated fuel handling accident if the calculated whole-body and thyroid doses at the exclusion area and low population zone boundaries are well within the exposure guideline values of 10CFRPart 100, paragraph 11. "Well within" means 25 percent or less of the 10CFRPart 100 exposure guideline values, i.e., 75 rem for the thyroid and 6 rem for the whole-body doses.	Not applicable. SRP 15.0.3, "Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors" is applied instead of SRP 15.7.4.	N/A

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 41 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.7.4 Radiological Consequences of Fuel Handling Accidents (continued)	2. The radioactivity control features of the fuel storage and handling systems inside containment and in the fuel building are acceptable if they meet the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," (Ref. 2) with respect to appropriate containment, confinement and filtering systems. 3. The model for calculating the whole-body and thyroid doses is acceptable if it incorporates the appropriate conservative assumptions in Regulatory Guide 1.25 (Ref. 3) with the exception of the guidelines for the atmospheric dispersion factors (X/Q values). The acceptability of the X/Q values is determined under SRP Section 2.3.4. 4. An ESF grade atmosphere clean-up system is required for the spent fuel storage area to reduce the potential radiological consequences. 5. The containment design is acceptable with respect to a postulated fuel handling accident if it possesses the capability for prompt radiation detection by use of redundant radiation monitors and automatic isolation if fuel handling operations inside containment occur when the containment is open to the environment (i.e., with a containment purge exhaust system). An acceptable alternative approach is containment venting through an ESF atmosphere cleanup system or containment isolation during fuel handling operations.		
15.7.5 Spent Fuel Cask Drop Accidents	The AEB acceptance criteria for this SRP section are based on the requirements of 10CFRPart 100 (Ref. 1) with respect to the calculated radiological consequences of a spent fuel cask drop accident and General Design Criterion 61 (Ref. 2) with respect to appropriate containment, confinement and filtering systems. 1. The plant site and dose mitigating ESF systems are acceptable with respect to the radiological consequences of a postulated spent fuel cask drop accident if the calculated whole-body and thyroid doses at the exclusion area and low population zone boundaries are well within the exposure guideline values of 10CFRPart 100, paragraph 11. "Well within" means 25 percent or less of the 10CFRPart 100 exposure guideline values, i.e., 75 rem for the thyroid and 6 rem for the whole-body doses.	Not applicable. Design configuration of the spent fuel cask handling crane limits its travel so that: a) it cannot pass over the spent fuel pool, and b) potential cask drop distances are less than 30 feet.	N/A

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 42 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.7.5 Spent Fuel Cask Drop Accidents (continued)	2. The radioactivity control features of the fuel storage and spent fuel cask handling system in the fuel building are acceptable if they meet the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," (Ref. 2) with respect to appropriate containment, confinement 'and filtering systems. 3. The model for calculating the whole-body and thyroid doses is acceptable if it incorporates the appropriate conservative assumptions in Regulatory Guide 1.25 (Ref. 3) with respect to gap. inventory as stated in positions C.1.d,e, and f of the guide. The acceptability of the atmospheric dispersion factors, X/Q values, is determined under SRP Section 2.3.4. 4. An ESF grade atmospheric cleanup system is required for the fuel handling building to reduce the potential radiological consequences of the fuel cask drop accident. 5. The plant design with regard to spent fuel cask drop accidents is acceptable without calculation of radiological consequences if potential cask drop distances are less than 30 feet and appropriate impact limiting devices are employed during cask movements, as determined by ASB.		
15.8 Anticipated Transients Without Scram	The rule specifies that light water reactors must have a number of prescribed systems, and equipment that are design-dependent, and have been proven to reduce the risk attributable to the ATWS events, to an acceptable level. In addition, the applicants must submit information sufficient to demonstrate the adequacy of the implemented ATWS features. Design and quality assurance criteria for the required systems and equipment should meet or exceed the criteria established in conjunction with ATWS rulemaking, as described in SRP Section 7.1, Appendix A, to ensure adequate independence, diversity, and reliability where required by the ATWS rule. 1. Acceptance criteria for Boiling Water Reactors (BWRs): This portion of this SRP does not apply to the US-APWR. 2. For Pressurized Water Reactors (PWRs): A. Provide measures to automatically initiate the auxiliary (or emergency) feedwater system and a turbine trip under conditions indicative of an ATWS. This equipment shall be independent and diverse from the reactor trip system from sensor output to the final actuation device.	Conformance with no exceptions identified. Diverse Actuation System provided.	15.8

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 43 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.8 Anticipated Transients Without Scram (continued)	B. Combustion Engineering or Babcock and Wilcox reactors applicants shall have provision for a scram system that is independent and diverse from the reactor trip system, from sensor output to the points of interruption of power to the control rods. C. These system and equipment shall be demonstrated to provide reasonable assurance that unacceptable plant conditions do not occur in the event of an anticipated transients D. The reactor coolant system (RCS) pressure shall not exceed ASME Service Level C limits (approximately 22 MPa or 3200 psig) containment safety parameters (e.g., temperature or pressure) should not exceed design limits 3. For Evolutionary Plants A. For evolutionary plants where the ATWS rule does not explicitly require a diverse scram system, the applicant may provide either of the following: i. A diverse scram system satisfying the design and quality assurance criteria specified in SRP Section 7.2 ii. Demonstrate that the consequences of an ATWS event are within acceptable values. B. For evolutionary plants, some of the equipment required to satisfy the rule may not be apply. For example, passive BWRs do not have recirculation pumps; therefore, these designs cannot provide equipment to trip them as required by the rule. For these designs provision of an equivalent action such as reducing the vessel water level may be acceptable. C. Applicants must demonstrate that the failure probability of failing the ATWS success criteria is sufficiently small because either: (1) the criteria are met, or (2) a diverse scram system is installed that reduces significantly the probability of a failure to scram. The analysis leading to the ATWS rule in NUREG-0460 used the following ATWS success criteria, which have their bases in the Commission regulations and GDC listed above. Applicant's design shall maintain :		

Table 1.9.2-15 US-APWR Conformance with Standard Review Plan Chapter 15 Transient and Accident Analyses (sheet 44 of 44)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
15.8 Anticipated Transients Without Scram (continued)	i. Coolable geometry for the reactor core. If fuel and clad damage were to occur following a failure to scram, GDC 35 requires that this condition should not interfere with continued effective core cooling. 10CFR50.46 defines three specific core-coolability criteria: (1) Peak clad temperature shall not to exceed 1221"C (2200"F), (2) Maximum cladding oxidation shall not to exceed 17% the total cladding thickness before oxidation, and (3) Maximum hydrogen generation shall not to exceed 1% of the maximum hypothetical amount if all the fuel clad had reacted to produce hydrogen. ii. Maintain reactor coolant pressure boundary integrity. Appendix A to WASH-1270 states that in evaluating the reactor coolant system boundary for ATWS events, "the calculated reactor coolant system transient pressure should be limited such that the maximum primary stress anywhere in the system boundary is less than that of the ‘emergency conditions' as defined in the ASME Nuclear Power Plant Components Code, Section III." The acceptance criteria for reactor coolant pressure, based upon the ASME Service Level C limits, are approximately 10.3 MPa (1500 psig) for BWRs and approximately 22MPa (3200 psig) for PWRs. iii. Maintain containment Integrity. Following a failure to scram, the containment pressure and temperature must be maintained at acceptably low levels based on GDC 16 and 38. The containment pressure and temperature limits are design dependent; but to satisfy GDC 50, those limits must ensure that containment design leakage rates are not exceeded when subjected to the calculated pressure and temperature conditions resulting from any ATWS event.		
15.9 Boiling Water Reactor Stability	This SRP is written for boiling water reactors (BWRs) and is not applicable to the US-APWR.	Not applicable. SRP applies only to BWRs.	N/A

Table 1.9.2-16 US-APWR Conformance with Standard Review Plan Chapter 16 Technical Specifications (sheet 1 of 1)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
16.0 Technical Specifications	The proposed plant-specific TS satisfy 10CFR50.34, 10CFR50.36, and 10CFR50.36a and are therefore acceptable if consistent with the regulatory guidance of the following STS documents and present plant-specific values for parameters at the indicated level of detail: • NUREG-1430, STS, Babcock and Wilcox Plants • NUREG-1431, STS, Westinghouse Plants • NUREG-1432, STS, Combustion Engineering Plants • NUREG-1433, STS, General Electric Plants, BWR/4 • NUREG-1434, STS, General Electric Plants, BWR/6 In TS change requests for facilities with TS based on previous STS, licensees should comply with comparable provisions in these STS NUREGs to the extent possible or justify deviations from the STS. Acceptable justifications for deviation would include retention of existing TS requirements, nonadoption of STS requirements not represented in existing TS (e.g., an LCO in STS but not in existing TS), editorial preference, facility design, and a technically justified alternative presentation equivalent to the STS intent. In some cases, comparison to the previous STS may help evaluate the proposed changes by clarifying the TS intent. The previous STS NUREGs are as follows: • NUREG-0103, STS, Babcock and Wilcox Plants • NUREG-0452, STS, Westinghouse Plants • NUREG-0212, STS, Combustion Engineering Plants • NUREG-0123, STS, General Electric Plants For applicants referencing a certified design, the certified generic TS of the referenced design provide the guidelines for the evaluation of proposed plant-specific TS.	Conformance with no exceptions identified. Note: per expectation expressed in the SRP, site specific characterization data will be contained in the COLA. Chapter 16.0 of the DCD contains specific site parameter requirements necessary to meet the engineering and design needs for safe construction and operation of the US-APWR.	16.3.4.16.3.5,16.3.6, 16.3.7,16.3.9
16.1 Risk-Informed Decision Making: Technical Specifications	This SRP describes a voluntary, risk-informed, non-traditional method for establishing Technical Specification values. It is not applicable to the US-APWR.	Not applicable. The US-APWR adopts the Risk-Managed Technical Specifications (RMTS) for its safety systems. Most of the requirement to implement RMTS specified by NEI 06-09 cannot be satisfied at the application stage of the design certification due to lack of plant-specific or station-specific information.	N/A

Table 1.9.2-17 US-APWR Conformance with Standard Review Plan Chapter 17 Quality Assurance and Reliability Assurance (sheet 1 of 4)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
17.1 Quality Assurance During the Design and Construction Phases	(17.1.1) The Organization (17.1.2) Quality Assurance Program (17.1.3) Design Control (17.1.4) Procurement Document Control (17.1.5) Instructions, Procedures, and Drawings (17.1.6) Document Control (17.1.7) Control of Purchased material, Equipment, and Services (17.1.8) Identification and Control of Materials, Parts, and Components (17.1.9) Control of Special Processes (17.1.10) Inspection (17.1.11) Test Control (17.1.12) Control of Measuring and Test Equipment (17.1.13) Handling, Storage, and Shipping (17.1.14) Inspection, Test, and Operating Status (17.1.15) Nonconforming materials, Parts, or Components (17.1.16) Corrective Action (17.1.17) Quality Assurance Records (17.1.18) Audits	Conformance with exceptions. (17.1.8), (17.1.9), (17.1.13), (17.1.14), are N/A in DC phase	17.1, 17.5
17.2 Quality Assurance During the Operations Phase	(17.2.1) The Organization (17.2.2) Quality Assurance Program (17.2.3) Design Control (17.2.4) Procurement Document Control (17.2.5) Instructions, Procedures, and Drawings (17.2.6) Document Control (17.2.7) Control of Purchased material, Equipment, and Services (17.2.8) Identification and Control of Materials, Parts, and Components (17.2.9) Control of Special Processes (17.2.10) Inspection (17.2.11) Test Control (17.2.12) Control of Measuring and Test Equipment (17.2.13) Handling, Storage, and Shipping (17.2.14) Inspection, Test, and Operating Status (17.2.15) Nonconforming materials, Parts, or Components (17.2.16) Corrective Action (17.2.17) Quality Assurance Records (17.2.18) Audits	Not applicable. COL applicant is responsible for.	N/A

Table 1.9.2-17 US-APWR Conformance with Standard Review Plan Chapter 17 Quality Assurance and Reliability Assurance (sheet 2 of 4)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
17.3 Quality Assurance Program Description	A. MANAGEMENT B. PERFORMANCE/VERIFICATION 1. Methodology 2. Design Control 3. Design Verification 4. Procurement Control 5. Procurement Verification 6. Identification and Control of Items 7. Handling, storage, and Shipping 8. Test Control 9. Measuring and Test Equipment Control 10.Inspection, Test, and Operating Status 11. Special Process Control 12. Inspection 13. Corrective Action 14. Document Control 15. Records C. SELF-ASSESSMENT	Conformance with exception. B-6, 7, 10, 11 are N/A in DC phase.	17.3
17.4 Reliability Assurance Program (RAP)	Section A below applies to a DC applicant and Section B below applies to a COL Applicant referencing a certified design. A. DESIGN CERTIFICATION The application describes the following RAP information: 1. The scope and purpose. The scope and purpose of the RAP are described in Subsections I and II of this SRP section. 2. The application of the quality elements associated with organization, design control, procedures and instructions, records, corrective action, and audit plans as follows: *(Additional text follows on requirements)* 3. The expert panel qualifications in the areas of personnel knowledgeable in the design, operation and maintenance of a plant, and experience necessary to perform the SSC selections if an expert panel is utilized. 4. Deterministic or other methods of analysis used to identify SSCs included in the RAP and the SSCs affected.	Conformance with no exceptions identified.	17.4

Table 1.9.2-17 US-APWR Conformance with Standard Review Plan Chapter 17 Quality Assurance and Reliability Assurance (sheet 3 of 4)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
17.4 Reliability Assurance Program (RAP) (continued)	5. A non-system-based ITAAC for the RAP that provides reasonable assurance that the design of SSCs within the scope of the RAP is consistent with their assumed design reliability. The ITAAC acceptance criteria should ensure that the estimated reliability of each as-built SSC is at least equal to the assumed design reliability and that industry experience including operations, maintenance, and monitoring activities were assessed in estimating the reliability of these SSCs. 6. A COL action item that a COL Applicant referencing a certified design will identify the site-specific SSCs within the scope of the RAP.		
17.5 Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants	(Note: All sections below contain additional text on requirements.) A. Organization B. Quality Assurance program C. Design control and verification D. Procurement document control E. Instructions, procedures, and drawings F. Document control G. Control of purchased material, equipment, and services H. Identification and control of materials, parts, and components I. Control of special processes J. Inspection K. Test control L. Control of measuring and test equipment M. Handling, storage, and shipping N. Inspection, test, and operating status O. Nonconforming materials, parts, or components P. Corrective action Q. Records R. Audits S. Training and qualification criteria - quality assurance T. Training and qualification - inspection and test U. QA program commitments V. Nonsafety-related SSC Quality Controls W. Independent review X. COL Action Items and Certification Requirements and Restrictions	Conformance with exceptions. H, I, M, N, W ,Y are identified in the SRP as a COL Applicant responsibility.	17.5

Table 1.9.2-17 US-APWR Conformance with Standard Review Plan Chapter 17 Quality Assurance and Reliability Assurance (sheet 4 of 4)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
17.5 Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants (continued)	For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters). For a COL application referencing a DC, a COL Applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL Applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC. Y. Operational Program Description and Implementation For a COL application, the staff reviews the Quality Assurance Program - Operation program description and the proposed implementation milestones. The staff also reviews final safety analysis report (FSAR) Table 13.x to ensure that the Quality Assurance Program - Operation and associated milestones are included.		
17.6 Maintenance Rule	The Maintenance Rule program is an operational program addressed in a COL Application, and does not apply to US-APWR design certification.	Not applicable. Maintenance rule is a COLA requirement.	N/A

Table 1.9.2-18 US-APWR Conformance with Standard Review Plan Chapter 18 Human Factors Engineering (sheet 1 of 1)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
18.0 Human Factors Engineering	A. Review of the HFE Aspects of a New Plant (Followed by additional text on requirements) A.1 HFE Program Management (Followed by additional text on requirements) A.2 Operating Experience Review (Followed by additional text on requirements) A.3 Functional Requirements Analysis and Function Allocation (Followed by additional text on requirements) A.4 Task Analysis (Followed by additional text on requirements) A.5 Staffing and Qualifications (Followed by additional text on requirements) A.6 Human Reliability Analysis (Followed by additional text on requirements) A.7 Human-System Interface Design (Followed by additional text on requirements) A.8 Procedure Development (Followed by additional text on requirements) A.9 Training Program Development (Followed by additional text on requirements) A.10 Verification and Validation (Followed by additional text on requirements) A.11 Design Implementation (Followed by additional text on requirements) A.12 Human Performance Monitoring (Followed by additional text on requirements) B. Review of the HFE Aspects of Control Room Modifications C. Review of the HFE Aspects of Modifications Affecting Risk-Important Human Actions	Conformance with exceptions. Criteria B and C are for modifications and are not applicable to the US-APWR design certification.	18.1, 18.2, 18.3, 18.4, 18.5, 18.6, 18.7, 18.8, 18.9, 18.10, 18.11, 18.12

Table 1.9.2-19 US-APWR Conformance with Standard Review Plan Chapter 19 Probabilistic Risk Assessment and Severe Accident Evaluation (sheet 1 of 2)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
19.0 Probabilistic Risk Assessment and Severe Accident Evaluation	***From the "Requirements" section of the SRP***: "For a DC 1. 10CFR52.47(8) - The information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10CFR50.34(f), specifically 10CFR50.34(f)(1)(i). 2. 10CFR52.47(a)(23) - For light-water reactor designs, a description and analysis of design features for the prevention and mitigation of severe accidents, e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass. 3. 10CFR52.47(a)(27) - A description of the design-specific probabilistic risk assessment (PRA) and its results." ***From the "SRP Acceptance Criteria" section of the SRP***: "1. NRC Policy Statement, "Severe Reactor Accidents Regarding Future Designs and Existing Plants," 50 FR 32138, August 8, 1985. 2. NRC Policy Statement, "Safety Goals for the Operations of Nuclear Power Plants," 51 FR 28044, August 4, 1986. 3. NRC Policy Statement, "Nuclear Power Plant Standardization," 52 FR 34884, September 15, 1987. 4. NRC Policy Statement, "Regulation of Advanced Nuclear Power Plants," 59 FR 35461, July 12, 1994. 5. NRC Policy Statement, "The Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," 60 FR 42622, August 16, 1995. 6. SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," ADAMS Accession No. ML003707849, January 12, 1990, and the related staff requirements memorandum (SRM), ADAMS Accession No. ML003707885, June 26, 1990. 7. SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," ADAMS Accession No. ML003708021, April 2, 1993, and the related SRM, ADAMS Accession No. ML003708056, July 21, 1993.	Conformance with exceptions. Note: The "8" and "9" of SRP Acceptance Criteria are for AP600 and out of the US-APWR scope. The other SRP requirements will be satisfied by chapter 19 of the DCD and a separate PRA report that will be provided as a supporting reference.	19.0, 19.1, 19.2, 19.3

Table 1.9.2-19 US-APWR Conformance with Standard Review Plan Chapter 19 Probabilistic Risk Assessment and Severe Accident Evaluation (sheet 2 of 2)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
19.0 Probabilistic Risk Assessment and Severe Accident Evaluation (continued)	8. SECY-96-128, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," ADAMS Accession No. ML003708224, June 12, 1996, and the related SRM, ADAMS Accession No. ML003708192, January 15, 1997. 9. SECY-97-044, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," ADAMS Accession No. ML003708316, February 18, 1997, and the related SRM, ADAMS Accession No. ML003708232, June 30, 1997. The first five NRC policy statements provide guidance regarding the appropriate course of action to address severe accidents and the use of PRA. The Commission SRMs relating to SECY-90-016, SECY-93-087, SECY-96-128, and SECY-97-044 provide Commission-approved guidance for implementing features in new designs to prevent severe accidents and to mitigate their effects, should they occur." *(Additional text follows on requirements)*		
19.1 Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed	"In order for the NRC staff to conclude that a PRA is of sufficient technical adequacy to support an application, the staff needs to be assured that (1) the parts of the PRA needed to support the application have been appropriately identified and (2) those parts have been performed in a manner consistent with current good PRA practice. The former needs to be addressed as part of the assessment of the application. The latter can be met by determining that the necessary parts of the PRA have been performed in accordance with the staff position on consensus PRA standards or industry programs as documented in the appendices to Regulatory Guide 1.200. Where there are differences in approach to performing a specific part, the staff must determine that the approach used by the applicant is either equivalent to, or better than, that supported by the staff position."	Conformance with no exceptions identified.	19.0, 19.1, 19.2, 19.3,
19.2 Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance	Note: This section was written to address PRAs performed in support of changes proposed for existing, already-licensed plants.	Not applicable. This SRP section was written to address PRAs performed in support of changes proposed for existing, already-licensed plants.	N/A

1.9.3 Generic Issues

Language cited from Reg Guide 1.206 section C.I.1.9.3 and from Standard Review Plan 1.0, "Introduction and Interfaces", section I.9, states that an applicant must include an evaluation of the proposed technical resolutions for those unresolved safety issues and medium- and high-priority generic safety issues that are identified in the version of NUREG-0933, "A Prioritization of Generic Safety Issues," current on the date up to 6 months before the docket date of the application and are technically relevant to the design. Section C.IV.8 of Reg Guide 1.206 provides additional guidance for addressing the issues identified in NUREG-0933.

Table 1.9.3-1 summarizes the Generic Issues as specified above that apply to the US-APWR, for the revision of NUREG-0933 bearing the publication date of September 2007. Language extracted from NRC's document is shown in the "Summary" column of the table. An explanation of the NRC statuses indicated for the generic issues in Table 1.9.3-1 is as follows:

- "Note 6" indicates new requirements recommended for future plants
- "HIGH" indicates a high safety priority
- "CONTINUE" indicates continuing work on the issue by NRC in accordance with Management Directive 6.4.

In Table 1.9.3-1, each generic issue is referenced by number and title, a summary for each issue is paraphrased from the NUREG-0933 , US-APWR status and discussion are provided, and references to the appropriate US-APWR DCD sections are also provided.

According to 10CFR52.47(a)(8), the information with respect to compliance with technically relevant positions of the Three Mile Island requirements in 10CFR50.34(f) must be contained in the DCD. The locations of the corresponding description in the DCD are provided in Table 1.9.3-2

Table 1.9.3-1 Conformance with Generic Issues (sheet 1 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #89 Stiff Pipe Clamps NRC priority: Note 6	This issue was identified following a staff evaluation of allegations that improper consideration of "stiff" pipe clamps in Class 1 piping systems could result in unsafe plant operation…In the staff's evaluation, it was found that piping designers often assumed that the clamp effects on piping systems were negligible and did not warrant any explicit consideration. This assumption was acceptable for most clamp applications. However, for some applications, certain piping system conditions coupled with specific stiff pipe clamp design requirements could result in interaction effects that should be evaluated in order to determine the significance of pipe stresses induced…Stiff pipe clamps were installed because of requirements for piping systems to withstand dynamic loads such as SRV discharges to suppression pools, LOCA-induced loads, and seismic loadings. A preloading of pipe clamp U-bolts or straps (which imposes a constant compressive load on the piping) is necessary to prevent stiff pipe clamps from lifting off piping under dynamic loading conditions. Since clamp-induced stresses are generally not significant with conventional pipe clamps, the pipe stresses induced by stiff pipe clamps generally were also not considered. Therefore, it was believed that further analyses of these stresses on piping systems were necessary before determining whether the stresses were significant. In addition to the large preloading of the clamps, four other new design features were identified by the staff as requiring additional analyses because of their difference from conventional pipe clamps. These were: (1) use of high-strength or non-ASME approved materials; (2) local surface contact on the pipe; (3) uncommonly thick and/or wide design of clamp; and (4) clamp applications to piping components other than straight pipe, such as pipe elbows. If neglect of the additional stress from stiff pipe clamps results in overestimating the pressure-retaining capabilities of piping systems, the probability of pipe breaks caused by dynamic loads may be higher than previously estimated. This increased probability could potentially result in an increased CDF that could lead to PRAs understating the public risk. This issue affected those operating and future plants that installed stiff pipe clamps. A possible solution could have the following elements: (1) Evaluation of the local pipe stresses induced by stiff pipe clamps under all loading conditions; (2) If the evaluation in (1) above indicated that clamp-induced pipe stresses were unacceptable, hardware modifications should be considered.	The conditions described in this generic issue are outside the boundaries of good practice for design of piping and pipe supports, and stiff pipe clamps, which are preloaded to prevent themselves from lifting off the piping under dynamic loading conditions , are not used for ASME Code, Section III, Class 1 piping in the US-APWR design .	3.12

Table 1.9.3-1 Conformance with Generic Issues (sheet 2 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #156.6.1 Pipe Break Effects on Systems and Components NRC priority: HIGH	This is a Systematic Evaluation Program (SEP) Issue. GDC 4 is the primary regulatory requirement of concern. It requires, in part, that structures, systems and components important to safety be appropriately protected against the environmental and dynamic effects that may result from equipment failures, including the effects of pipe whipping and discharging fluids. Several possible scenarios for plants that do not have adequate protection against pipe whip were identified as a result of the research performed in support of the enhanced prioritization. Related regulatory criteria include common-cause failures, protection system independence, and the single failure criterion. Possible Solution is to Issue generic letters to the affected plants requesting that they perform plant-specific reviews and walkdowns, identify vulnerable pipe break locations, and inform the NRC of proposed corrective actions. Three cases identified for PWRs: Case 1: Failure of Non-Leak-Before-Break Reactor Coolant System, Feedwater, or Main Steam Piping Resulting in Pipe Whip or Jet Impingement on Reactor Protection or Instrumentation & Control Electrical, Hydraulic or Pneumatic Lines or Components and Eventually Resulting in Failure of Mitigation Systems and Core Damage Case 2: Failure of Main Steam or Feedwater Piping Resulting in Pipe Whip and Containment Impact/Failure, with Resultant Failure of All Emergency Core Cooling Systems Case 3: Failure of Main Steam or Feedwater Piping Resulting in Pipe Whip Impact on CCW System to the Extent That the CCW Pressure Boundary is Broken, Potentially Opening a Path to Outside Containment if Containment Isolation Fails to Occur; Also Possible Loss of CCW Outside Containment for Mitigation	The US-APWR design criteria used to evaluate pipe failure protection are generally consistent with the NRC guidelines including those in NUREG-0800, SRP 3.6.1, 3.6.2, and 3.6.3, and applicable Branch Technical Position (BTP) 3-4 and BTP 3-3.	3.6

Table 1.9.3-1 Conformance with Generic Issues (sheet 3 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #156.6.1 Pipe Break Effects on Systems and Components NRC priority: HIGH (continued)		Safety-related systems and components are defined as those required to shutdown the reactor and mitigate the consequences of the postulated piping failure. Section 3.6.1 of the DCD provides the design bases and criteria for the analysis required to demonstrate that the essential systems are protected. The high– and moderate-energy systems representing the potential source of dynamic effects are listed and criteria for separation and effects of adverse consequences are defined. Section 3.6.2 defines the criteria for postulated break location and configuration. High-energy pipes are evaluated for the effects of circumferential and longitudinal pipe breaks and through–wall cracks. Moderate-energy pipes are evaluated for the effects of through-wall cracks. Analysis methods and criteria for evaluating pipe whip and evaluating consequences of jet impingement, motion of the pipe, and system depressurization on the integrity and operability are provided. The evaluation of containment penetrations, pipe whip restraints, guard pipes, and other protective devices are also described. Section 3.6.3 describes the application of leak-before-break (LBB) criteria. In accordance with NUREG-0800, SRP 3.6.1, the US-APWR is designed for protection against piping failure inside or outside the containment to assure that such a failures would not compromise cause the functional capability loss of needed functions of safety-related systems and to restore ensure that the plant in the could be safely shutdown condition and maintain it in that condition in the event of such failures.	

Table 1.9.3-1 Conformance with Generic Issues (sheet 4 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #156.6.1 Pipe Break Effects on Systems and Components NRC priority: HIGH (continued)		The design includes consideration of high-energy and moderate-energy fluid system piping located inside and outside of the containment. The habitability of the main control room habitability system is also protected. In addition, containment penetrations and isolation valves (including non-safety systems) are also protected. Evaluations are made based upon circumferential or longitudinal pipe breaks, through-wall cracks, or leakage cracks as determined by the appropriate criteria. At locations determined to be subject to a circumferential or longitudinal pipe break, dynamic effects such as jet impingement and pipe whip are evaluated. At locations subject to through-wall cracks or leakage cracks, effects such as spray wetting and flooding are evaluated. Through-wall cracks, which are postulated in high-energy piping and in moderate-energy lines, are larger and have a larger flow rate of water or steam than the leakage cracks postulated for high-energy piping, which satisfies the LBB requirements. The pressurization loads on structures and components are evaluated for postulated circumferential breaks and longitudinal breaks in piping that cannot be qualified for does not meet LBB application requirements and for postulated leakage cracks in piping that meet the LBB requirements. The basis for this approach is that NUREG-0800, BTP 3-4 describes an acceptable method for selecting the design locations and orientations for potential breaks and cracks in fluid systems piping. SRPs 3.6.1 and 3.6.2 describe acceptable measures that could be taken for protection against the breaks and cracks and for restraint against pipe whip that may result from breaks.	

Table 1.9.3-1 Conformance with Generic Issues (sheet 5 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #156.6.1 Pipe Break Effects on Systems and Components NRC priority: HIGH (continued)		In order to maintain the safety of the plant when a pipe break is postulated, the following are considered in the design of the plant: • Maintain the functions of engineering safety features facilities and related features/facilities required in cooling the reactor core • Maintain the functions of reactor shutdown systems • Ensure that containment integrity is maintained • Ensure that radiological doses of a postulated piping failure remain below the limits of 10CFR100 • Ensure that the control room function and habitability is maintained To maintain and ensure the functionality and performance described above, design considerations include building arrangement, equipment arrangement, and arrangement and design of piping and pipe supports.	

Table 1.9.3-1 Conformance with Generic Issues (sheet 6 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #163 Multiple Steam Generator Tube Leakage NRC priority: HIGH	This issue was identified to address the safety concern associated with potential multiple steam generator tube leaks during a main steam line break that cannot be isolated. This sequence could lead to core damage that could result from the loss of all primary system coolant and safety injection fluid in the refueling water storage tank. The issue was based on a DPO filed in June 1992. The safety concern of this issue is being addressed in the staff's work on steam generator tube integrity. The staff originally planned to develop a proposed rule on steam generator tube integrity to implement a more flexible regulatory framework for steam generator surveillance and maintenance activities that would allow a degradation-specific management approach. However, the results of a regulatory analysis suggested that a more optimal approach was to utilize a generic letter. The staff suggested, and the Commission subsequently approved, a revision to the regulatory approach to utilize a generic letter. The existing regulatory framework provides reasonable assurance that operating plants are safe; however, this framework has numerous shortcomings. In order to resolve these shortcomings, the staff will revise the regulatory framework to utilize a risk-informed performance-based approach that will ensure compliance with existing regulations. Thus, this issue was given a HIGH priority ranking.	The event involved in this event is an un-isolable main steam line break outside containment with multiple leaking steam generator tubes, resulting in so much primary-to-secondary crossover flow for such a long period that the reactor coolant and refueling water inventories (i.e., normal, emergency core and long term recirculation cooling) become depleted, resulting in damage to the reactor core. This would appear to be a severe accident concern, specifically over the possibility of a main steam line break in the break exclusion zone between the outboard side of the containment wall and the main steam isolation valves. In the case of the US-APWR, there are 4 main steam lines, each with its own MSIV, and the space is referred to as the main steam piping room, which is outside containment but in the R/B. ***Comparable DBAs*** - While the beyond design basis accident described in this generic issue has not been specifically analyzed for US-APWR, several DBAs are summarized below, with the intent of showing that each is an event of comparable magnitude and phenomena, and each has proven the US-APWR to be a safe design. DCD section 15.6.2 examines the radiological consequences of the failure of small lines carrying primary coolant outside containment. The bounding case was determined to be a leak in the sample line due to manufacturing defect, corrosion, or maintenance activities.	3.6 (line breaks and break exclusion zone), 10.3.5 (water chemistry), 10.3 (main steam supply system), 10.4.9 (EFW system), 15.6.2 (failure of small lines carrying primary coolant outside containment), 15.6.3 (SGTR analysis), 15.6.5 (LOCAs in containment)

Table 1.9.3-1 Conformance with Generic Issues (sheet 7 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #163 Multiple Steam Generator Tube Leakage NRC priority: HIGH (continued)		The sample lines extending outside containment are provided with isolation valves on both sides of the containment wall and are designed in accordance with the requirements of General Design Criterion (GDC) 55. For small lines that meet GDC 55, the failure is assumed to occur downstream of the outboard containment isolation valve in conjunction with a single failure of one of the two containment isolation valves. The amount of primary coolant released outside the containment is determined by the time required to detect such a failure and the time required to isolate the failure (i.e., time to close the operable isolation valve). The amount of primary coolant released is conservatively estimated by assuming critical flow at the small line break location with the reactor coolant fluid enthalpy corresponding to normal reactor operating conditions. For the small line break outside containment transient, the loss of coolant reduces the volume control tank level and creates a demand for automatic makeup from the CVCS. Frequent operation of the automatic makeup system will provide indication of the loss of primary coolant.	

Table 1.9.3-1 Conformance with Generic Issues (sheet 8 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #163 Multiple Steam Generator Tube Leakage NRC priority: HIGH (continued)		Upon indication of a small line break, the operator would take action to isolate the break by closing the operable isolation valve for the damaged line. The operator is assumed to detect and isolate the break within 45 minutes. DCD Section 15.6.3 examines the steam generator tube rupture (SGTR) accident event, which involves the complete severance of a single steam generator tube. The accident is assumed to take place at full power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited number of defective fuel rods. The accident leads to leakage of radioactive coolant from the reactor coolant system (RCS) to the secondary system. In the event of a coincident loss of offsite power, or failure of the turbine bypass system, atmospheric discharge of radioactivity can take place via the steam generator main steam relief valves (MSRVs) or main steam safety valves (MSSVs). Primary-to-secondary leak flow continues after the SI flow is stopped until RCS pressure and the ruptured steam generator pressure equalize. Following termination of SI, the plant is basically stabilized. Charging flow, letdown, and pressurizer heaters have to be controlled to prevent re-pressurization of the RCS and re-initiation of leakage into the ruptured steam generator.	

Table 1.9.3-1 Conformance with Generic Issues (sheet 9 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #163 Multiple Steam Generator Tube Leakage NRC priority: HIGH (continued)		Mitigating factors for the SGTR DBA include: • The assumption of a compete tube severance is considered conservative because the tube material Alloy 690 is a corrosion resistant and ductile material. The more likely mode tube failure would be one or more minor leaks. • The radioactivity in the secondary system is under continual surveillance. An accumulation of activity (that would result from leaks) that exceeds the limits established in Technical Specifications is not permitted during operation.	

Table 1.9.3-1 Conformance with Generic Issues (sheet 10 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #163 Multiple Steam Generator Tube Leakage NRC priority: HIGH (continued)		• The operator is reasonably expected to recognize the occurrence of a steam generator tube rupture event, to identify and isolate the ruptured steam generator, and to take appropriate actions to stabilize the plant. These operator actions are expected to be performed in a timely manner to minimize contamination of the secondary system and the release of radioactivity to the atmosphere. Recovery procedures are expected to be carried out on a time scale that ensures the break flow to the secondary system is terminated before water level in the ruptured steam generator can rise to the main steam line. • The emergency feedwater and the safety injection flow from the in-containment refueling water storage pit provides the heat sink for decay heat from the reactor. This reduces the amount of steam dumped to the condenser, or in the case of loss of offsite power, steam discharged to the atmosphere through the main steam relief valve. • Makeup water from the safety injection flow increases the RCS water inventory and stabilizes the RCS pressure and pressurizer water level. After safety injection is terminated, the break flow stops when the RCS pressure equalizes with the ruptured steam generator pressure. At this point, the plant is stabilized and RHR is initiated to perform long term cooling.	

Table 1.9.3-1 Conformance with Generic Issues (sheet 11 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #163 Multiple Steam Generator Tube Leakage NRC priority: HIGH (continued)		Operator actions for SGTR recovery are provided in the Emergency Operating Procedures and include: a) identifying the steam generator with the ruptured tube, based on several reliable plant indications that are monitored routinely, b) isolating the affected steam generator, including closing the MSRV block valve if the MSRV fails open, c) terminating SI flow (after adequate RCS sub-cooling and sufficient reactor coolant inventory has been established, SI flow is stopped to terminate the primary-to-secondary leakage). The EOP actions are condition-based and can therefore accommodate variations in parameters such as break flow between an actual event and the DCD analysis. Also, large and small break loss of coolant accidents LOCAs, examined in 15.6.5, are significantly larger in magnitude than SGTR events. Both have been analyzed with results that demonstrate the reactor core remains amenable to cooling, despite these catastrophic challenges. ***Main Steam System*** - DCD section 10.3 describes that the safety-related portion of the main steam system (MSS) are designed such that a single failure in the MSS will not result in initiation of a Loss-of-coolant accident, loss of integrity of other steam lines, loss of the capability of the engineered safety features system to effect a safe reactor shutdown, or transmission of excessive loading to the containment pressure boundary. The main steam supply system sections constructed in accordance with ASME Section III, Class 2 requirements are provided with access to welds and removable insulation from areas required for in service inspection in accordance with ASME Section XI. The main steam lines between the steam generators and the containment penetration are designed to meet the leak-before-break criteria.	

Table 1.9.3-1 Conformance with Generic Issues (sheet 12 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #163 Multiple Steam Generator Tube Leakage NRC priority: HIGH (continued)		The portion of the main steam lines between the containment penetration and the anchor downstream of the main steam isolation valve is part of the break exclusion zone, described in DCD section 3.6. Section 3.6 addresses the applicability of leak-before-break and break exclusion zone to the main steam lines, and also describes special ASME code requirements for materials, welding, inspection and surveillance for lines in the break exclusion zone. Radioactive contamination of the MSS can occur by a primary side to secondary side leak in the steam generator. Under normal operating conditions there is no significant amount of radioactivity in the MSS. In-line radiation monitors on each steam line, condenser air removal system radiation monitor and steam generator blowdown line facilitate leak detection. Additionally, the main steam isolation valves are designed to provide controls for reducing releases by isolating the affected main steam line following a steam generator tube rupture. The safety-related portions of the main steam supply system are located in the containment and the main steam piping room. These buildings are designed to withstand the effects of earthquakes, tornadoes, hurricanes, floods, external missiles and other appropriate natural phenomena. DCD Sections 3.3, 3.4, 3.5, 3.7 and 3.8 describe the bases of the structural design of these buildings. The safety-related portion of the MSS system is designed to remain functional after a safe shutdown earthquake.	

Table 1.9.3-1 Conformance with Generic Issues (sheet 13 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #163 Multiple Steam Generator Tube Leakage NRC priority: HIGH (continued)		***Water Chemistry*** is described in DCD section 10.3.1. The objectives of the secondary side water chemistry controls are specifically designed to: a) minimize general corrosion in the steam generators, turbine, and feedwater system by maintaining pH control and minimizing oxygen ingress, and b) Minimize localized corrosion in the steam generators, turbine and feedwater system by minimizing chemical contaminant ingress and controlling contaminant levels by polishing condensate and steam generator blowdown. ***DCD section 10.3.1 describes the US-APWR Emergency Feedwater System*** - The emergency feedwater system (EFWS) is designed to supply feedwater to the steam generators whenever the reactor coolant temperature is above 350°F and the FWS is not in operation; i.e., during startup, cooldown, or emergency conditions resulting in a loss of main feedwater. . The EFWS is designed to remove reactor decay heat and RCS residual heat through the steam generators following transient conditions or postulated accidents such as reactor trip, loss of main feedwater, steam or feedwater line breaks, steam generator tube rupture, and unavailability of the FWS.	

Table 1.9.3-1 Conformance with Generic Issues (sheet 14 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #163 Multiple Steam Generator Tube Leakage NRC priority: HIGH (continued)		***Conclusions***: This issue is a severe accident consideration that US-APWR can address on a risk basis if required as NRC continues to develop an approach. For design basis events, the ECCS and other ESFs are robustly designed to handle main steam line break, SGTR, and LOCA events. Steam generator tube integrity is established by design with a selection of materials determined to prevent corrosion, and is maintained in plant operation by water chemistry controls and other periodic surveillance. Similarly, the integrity of the main steam piping in the break exclusion zone in the main steam piping room is heavily fortified by design and by administrative controls. Comparison to similar events in the DBA category show that these types of events do not proceed indefinitely, and that the US-APWR is designed to accommodate very severe challenges without core damage.	

Table 1.9.3-1 Conformance with Generic Issues (sheet 15 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #186 Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants NRC priority: CONTINUE	This issue was identified when the concern was raised that licensees operating within the regulatory guidelines of GL 85-11 may not have taken adequate measures to assess and mitigate the consequences of dropped heavy loads. n April 1996, NRC Bulletin 96-02 was issued to alert licensees of potential high consequences that could result from a cask drop and to remind them of complying with existing regulatory guidelines on the control and handling of heavy loads. In nuclear plant operation, maintenance, and refueling activities, heavy loads may be handled in several plant areas. If these loads were to drop because of human error or crane failure, they could impact on stored spent fuel, fuel in the core, or on equipment that may be required to achieve safe shutdown or permit continued decay heat removal. In some instances, load drops at specific times, locations, and weights could potentially lead to offsite doses that exceed 10CFRPart 100 limits. If a licensee elected to use long-term dry storage casks to store excess spent fuel, the large, heavy casks would have to be hoisted and transported to and from the spent fuel pool while the plant is at full power operation. A comprehensive analysis of U.S. nuclear industry crane operating experience from 1968 through 2002 was conducted by the NRC and documented in NUREG-1774. Some of the NRC's findings and observations were: 1) The human error rate for crane operating events increased significantly; 2) Load drop events between the period 1993-2002 increased over the period 1981-1992; 3) The number of below-the-hook crane events (mainly rigging deficiencies or failures) increased greatly; 4) Calculational methodologies, assumptions, and predicted consequences varied greatly from licensee to licensee for very similar accident scenarios; 5) The number of mobile crane events declined slightly; and 6) There were few load slips or drops involving very heavy loads. 7) Criteria for declaring a crane as single-failure-proof were applied inconsistently	The concern of this generic issue is regarding load drops that have occurred in recent years and the possibility that such an event could someday result in the load being dropped onto a source of radioactive inventory, such as stored spent fuel, fuel in the core, equipment that is performing a decay heat removal function, or equipment that would be required for safe shutdown. Per the language of the generic issue, some of the events that have occurred could have been prevented by single failure proof crane design (i.e., load drops or hook and block assembly drops). Many of these “below the hook” events, however, were rigging errors that were strictly the result of manual operator faults, and would not have been prevented by single failure proof crane design. NRC adopted 4 recommendations for development of follow-up guidance. Two of the recommendations involve evaluation and endorsement of cranes and rigging equipment that would result in fewer mishaps. The other two recommendations involve NRC developing guidance on good practices for crane operations, load movements, and load drop calculations. For the US-APWR, design (and later, by the COL Applicant, operational procedures) for the containment polar and refueling cranes, spent fuel pit crane, and auxiliary building crane preclude the dropping of heavy loads. A critical load is defined in ASME NOG-1-2004 and referred to in this DCD 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 10CFRlimits.	3.5 (design of SSCs - cranes), 9.1.1 through 9.1.5 (descriptions of new and spent fuel handling and storage), chapter 13 (conduct of operations), 15.7.4 (fuel handling accident) 15.7.5 (spent fuel cask drop accident) 18.2 (human factors - operations organization), 18.4 (human factors - task analysis and hazards evaluation), 18.9 (human factors - procedural development)

Table 1.9.3-1 Conformance with Generic Issues (sheet 16 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #186 Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants NRC priority: CONTINUE (continued)	8) Among events occurring during the period 1968 through 2002 involving cranes suitable for an upgrade to a single-failure-proof design, most load drop events were the result of poor program implementation or human performance errors that led to hoist wire rope or below-the-hook failures. All three very heavy load drops were the result of rigging failures, not crane failures. Consequently, there were no very heavy load drop events that could have been prevented had only a single-failure-proof crane been employed in the lift. However, there were load or hook and block assembly drops that could have been prevented with the use of single-failure-proof cranes and lifting devices. The screening and technical assessments of the issue were documented in NUREG-1774. At the completion of the technical assessment, four recommendations were made for follow-up guidance development by the NRC staff: 1) Evaluate the capability of various rigging components and materials to withstand rigging errors and issue necessary guidelines for rigging applications. 2) Endorse ASME NOG-1 for Type I cranes as an acceptable method of qualifying new or upgraded cranes as single-failure-proof and issue guidance endorsing the standard, as appropriate. 3) Reemphasize the need to follow Phase I guidelines involving good practices for crane operations and load movements and continue to assess licensee implementation of heavy load controls in safety-significant applications. 4) Request the appropriate industry Code Committees to evaluate the need to standardize load drop calculational methodologies for nuclear power plants.	Cranes that may be used to handle critical loads over SC-I SSCs, are classified as Type I cranes as defined per ASME NOG-1-2004, and will conform to 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. • Polar Crane - The Reactor Containment is designed to have a reactor cavity of reinforced concrete construction with stainless steel lining, and is equipped with a refueling crane and a polar crane that will enable refueling operation to be carried out on the main operating Floor. The polar crane girder is directly fixed to the cylindrical portion of the containment vessel. When in use, the polar crane is under administrative controls. During hot standby and hot shutdown, it is anticipated that the polar crane will be used to minimize critical path outage times for cold shutdowns and refueling, and to assist with maintenance that can be performed in a hot plant condition. Planned usage includes activities such as crane inspections, operability checks, and movement of tools and equipment required for the cold shutdown/refueling outage. The anticipated loads would not be required to be lifted in the vicinity of the reactor vessel.	

Table 1.9.3-1 Conformance with Generic Issues (sheet 17 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #186 Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants NRC priority: CONTINUE (continued)		• Refueling Crane - The refueling crane is a bridge crane consisting of a frame and transfer carriage that move horizontally on the rail installed on the canals inside the reactor cavity and inside the reactor containment vessel. On the transfer carriage are a control platform and a mast tube assembly, including the gripper tube to grip the fuel assemblies. Contained in the mast tube, a fuel assembly can be moved to an appropriate position in the canals inside the reactor cavity and inside the reactor containment vessel. The gripper located in the lower part of the gripper tube is pneumatically operated, and is provided with a device that prevents the fuel assembly from being dropped by the gripper. If there is no air pressure, the fuel is held and cannot be removed from the gripper. Furthermore, the crane is provided with a load indicator and interlocks that prevent a lifting operation if the preset load is exceeded, thereby preventing an assembly from being dropped due to excessive load. Interlocks are also provided to assure safe and secure operation of the frame and transfer carriage, as well as safe and secure ascending and descending of the gripper tube. The refueling crane is designed with a device that secures the traveling portion to the rail so that it never falls, including during the earthquake event. • Spent Fuel Pit Crane - The spent fuel pit crane is a bridge crane running above the spent fuel pit and moves the spent fuel by a hoist, to which are attached a special frame and handling tools. The spent fuel pit crane is designed to “fail as is” with a loss of driving power, and a mechanical interlock for the handling tools is provided so that	

Table 1.9.3-1 Conformance with Generic Issues (sheet 18 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #186 Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants NRC priority: CONTINUE (continued)		the fuel assembly will not drop during fuel handling. The spent fuel pit crane is designed with a device that secures the traveling portion to the rail so that it never falls, including during the earthquake event. • Spent Fuel Cask Handling Crane - The spent fuel cask handling crane is an overhead traveling crane designed to safely and securely transfer the fresh fuel transport container, spent fuel transport packaging, and fresh fuel. The spent fuel cask handling crane is also designed so that it never falls, including during the earthquake event. US-APWR cranes that will 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 will be designed to remain in place during a seismic event. Cranes that handle critical loads as well as non-critical loads will conform to 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. Also, 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 will be	

Table 1.9.3-1 Conformance with Generic Issues (sheet 19 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #186 Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants NRC priority: CONTINUE (continued)		stressed to those involved with heavy load lifts. Anticipated heavy load movements will be 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 will be 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 will be 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. As a minimum, procedures will be used for handling loads with spent fuel cask bridge crane and polar crane, and for those loads listed in table 3-1 of NUREG 0612. It is anticipated that each procedure will address: • Specific equipment required to handle load (e.g., special lifting device, slings, shackles, turnbuckles, clevises, load cell, etc.). • Requirements for crane operator and riggers 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	

Table 1.9.3-1 Conformance with Generic Issues (sheet 20 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #186 Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants NRC priority: CONTINUE (continued)		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 will conform to NUREG 0612 • Equipment layout drawings showing the safe load path will be used to define safe load paths in load handling procedures, and deviation from defined safe load paths will require a written alternative procedure approved by the COL Applicant's plant review board In DCD chapter 18, human factors considerations and commitments appropriate to the DCD stage are described, applying to both design features and plant personnel. Of interest to this generic issue are sections 18.2 (human factors - operations organization), 18.4 (human factors - task analysis and hazards evaluation) and 18.9 (human factors - procedural development). 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 US-APWR will, of course, also be attentive to any new operational guidance that comes from NRC's efforts related to this issue.	

Table 1.9.3-1 Conformance with Generic Issues (sheet 21 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #189 Susceptibility of Ice Condenser Containments to Early Failure from Hydrogen Combustion During a Severe Accident NRC priority:	For the majority of PWRs with large dry or sub-atmospheric containments, direct containment heating (DCH) is the dominant mode of containment failure (a separate issue that was resolved by plant-specific comparison of DCH loads versus containment strengths), and the containment loads associated with hydrogen combustion are non-threatening. However, it was discovered in the study associated with NUREG/CR-6427 that, for ice condenser containments, the early containment failure probability is dominated by non-DCH hydrogen combustion events. This is not a surprising result, given the relatively low containment free volume and low containment strength in these designs. These containments rely on the pressure-suppression capability of their ice beds, and, for a design-basis accident, where the pressure is a result of the release of steam from blowdown of the primary (or secondary) system, an ability to withstand high internal pressures is not needed. In a beyond-design-basis accident, where the core is severely damaged, significant quantities of hydrogen gas can be released. This hydrogen is generated by the exothermic chemical reaction of water and steam with metal (especially the Zircaloy cladding), and (to some extent) by radiolysis of water, where gamma rays actually split water molecules into hydrogen and oxygen. To deal with large quantities of hydrogen, these containments are equipped with ac/AC-powered igniters, which are intended to control hydrogen concentrations in the containment atmosphere by initiating limited “burns” before a large quantity accumulates. In essence, the igniters prevent the hydrogen (or any other combustible gas) from accumulating in large quantities and then suddenly burning (or detonating) all at once, which would pose a threat to containment integrity. For most accident sequences, the hydrogen igniters can deal with the potential threat from combustible gas buildup. The situation of interest for this generic issue only occurs during accident sequences associated with station blackouts, where the igniter systems are not available because they are ac/AC-powered. The issue also applies to BWR MARK III containments, because they also have a relatively low free volume and low strength, comparable to those of the PWR ice condenser designs.	Not applicable to US-APWR design certification. (This issue is specifically applied to PWR ice condenser and BWR Mark III containments on the basis that they are relatively small in terms of containment volume, and have relatively low resistance to over-pressurization by hydrogen accumulation due to their thin steel shells. The US-APWR has a massive concrete containment structure that is both high strength and has a large volume.)	NA

Table 1.9.3-1 Conformance with Generic Issues (sheet 22 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #189 Susceptibility of Ice Condenser Containments to Early Failure from Hydrogen Combustion During a Severe Accident NRC priority: (continued)	The MARK III designs are equipped with hydrogen igniters just as are the PWR ice condenser designs, and are similarly potentially vulnerable in an accident sequence associated with station blackout. The solution is to provide an independent power supply for the igniter systems for the subject containments. The igniters are, essentially, diesel engine glow plugs. If necessary, they could be powered by storage batteries or by a portable generator. Based on the change in large early containment failure frequency (LERF) for both PWR ice condenser and BWR Mark III containment designs and on the change in risk (as measured by man-rem/ year) for the ice condenser designs, this issue passes the screening criteria and should go on to the technical assessment stage.		

Table 1.9.3-1 Conformance with Generic Issues (sheet 23 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #191 Assessment of Debris Accumulation on PWR Sump Performance NRC priority: HIGH	Results of research on BWR ECCS suction strainer blockage identified new phenomena and failure modes that were not considered in the resolution of Issue A-43. In addition, operating experience identified new contributors to debris and possible blockage of PWR sumps, such as degraded or failed containment paint coatings. Thus, this issue was identified by NRR and called for an expanded research effort to address these new safety concerns. A study was deemed to be required to determine whether PWR ECCS sumps are adequate to ensure proper ECCS operation. Based on the existence of an action plan to address the safety concerns, the issue was considered nearly-resolved in September 1996. It was later given a HIGH priority ranking in SECY-98-166.	US-APWR is following up to date methodology for sump design and performance, as summarized below. • ***Sump Description*** - Each quadrant of the RWSP contains paired suction piping and the suction pit arrangement for the CSS/RHR pumps and SI pumps. The open end of each suction pipe is equipped with a debris strainer that satisfies NEI 04-07 PWR Sump Performance Evaluation Methodology and conforms to RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant-Accident”. The suction strainers are designed to Seismic Category I and Quality Class B standards. The debris strainers are a passive disc-type design with a large “footprint” that is sufficient to preclude debris clogging. The debris strainers are made of stainless steel and could use perforated plates in a layered disc design to limit the maximum “pass through” debris size to accommodate with downstream design.	3.8.4.1 (robust Cat I design & construction requirements), 6.2.2.2.5 (refueling water storage pit), 6.2.2.2.6 and table 6.2.2-1 (ECCS/CS strainers and conformance with RG 1.8.2), Chapter 16 and related Tech Spec document (LCOs relevant to ensuring sump availability and performance)

Table 1.9.3-1 Conformance with Generic Issues (sheet 24 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #191 Assessment of Debris Accumulation on PWR Sump Performance NRC priority: HIGH (continued)		***Detailed conformance with RG 1.8.2*** - DCD Table 6.2-1 presents a comparison of the RWSP recirculation intake debris strainer design to the guidance of RG 1.82., addressing the topics of: • General materials and geometry • Minimizing Debris • Instrumentation • In-Service inspection • Evaluation of Alternative Water Sources • Evaluation of Long-Term Recirculation Capability • Debris Sources and Generation • Debris Transport • Debris Accumulation and Head Loss Some highlights from the US-APWR-specific response to the RG 1.82 requirements: • Four separate, independent and redundant 50% capacity trains each of CSS and SI are provided. Each quadrant of the RWSP contains paired CSS and SI suction pipes and each pair of CSS and SI suction pipes ends in a suction sump, with each suction sump protected by an associated suction strainer. The RWSP is the common suction source to the ECCS and CSS and contains approximately 607,500 gallons of 4500 ppm boric acid at pH 4.2. Crystalline NaTB is added to raise pH to at least 7 for iodine removal and long term LOCA cooling and recovery. LOCA spillage and spray return flow paths to RWSP promote full mixing.	

Table 1.9.3-1 Conformance with Generic Issues (sheet 25 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #191 Assessment of Debris Accumulation on PWR Sump Performance NRC priority: HIGH (continued)		• Containment drains (transfer pipes) into RWSP are protected from large debris by vertical debris bars capped by a ceiling plate. The sump openings (suction strainers) are located at approx. elevation 3’-7” of containment, with CSS and SI suction at approx –1’-5”. Disk -type suction strainer bases are mounted above the RWSP floor. • Suction strainers are to be base mounted above level RWSP floor. Design analysis inputs for debris transport are conservative. • The transfer pipe openings are equipped with vertical debris bars capped by a ceiling plate. The transfer pipes are located in areas of containment where drains will not directly impinge on them. • Vertical debris bars and ceiling plate protecting transfer pipe openings are of robust design and provide adequate protection from missiles and other large debris. Suction strainers are designed to Seismic Category I and Quality Class B standards. Design loads are properly combined and differential pressure caused by potential debris clogging is taken into account as part of mechanical analysis. • Corrosion resistant (stainless steel) material is used for suction strainers and all inner surfaces of the RWSP. • RWSP hatches are provided and suction strainers are designed for inspections.	

Table 1.9.3-1 Conformance with Generic Issues (sheet 26 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #191 Assessment of Debris Accumulation on PWR Sump Performance NRC priority: HIGH (continued)		• Strainers are sized appropriately to withstand debris. Because the RWSP has a large floor area, strainers are free from space restrictions and associated debris blockage. • The debris strainers are to be made of stainless steel and will use perforated plates in a layered disc design to limit the maximum “pass through” debris size to accommodate with downstream design. • RWSP suction strainers are submerged under a minimum of approximately 4 ft of water during a LOCA. The RWSP recirculation supply is sufficient to preclude adverse hydraulic effects such as vortex formation and high suction head loss. A low approach velocity at the strainer surface also mitigates the risk of vortexing. • The US-APWR design of ESF structures, systems or components does not include a CSS or SIS suction flow path that bypasses the RWSP suction strainers. • For purposes of minimizing debris, cleanliness, housekeeping and FMEA (foreign material exclusion areas) are administrative controls to be developed by the COL Applicant referencing the certified US-APWR design for construction and operation. Particulate (e.g., calcium silicate-based) insulation is excluded from containment by design.	

Table 1.9.3-1 Conformance with Generic Issues (sheet 27 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #191 Assessment of Debris Accumulation on PWR Sump Performance NRC priority: HIGH (continued)		• Information on debris produced by chemical reaction between ECCS water sources and containment materials (chemical effects) currently is being developed. Principle measures taken by the US-APWR design to preclude adverse chemical effects include use of buffering agent NaTB, and excluding particulate-producing material (e.g., calcium silicate-based insulation) from containment. • CS and SI pump operating information is available in the control room to assist in anomalous NPSH evaluation, including flow, suction and discharge pressure, and pump motor current. • In-Service Inspections of the sumps and strainers are the responsibility of any licensee who references the US-APWR certified design for construction and operation. • Calculated cooling performance (10CFR50.46(a)(1)(i)) of the US-APWR ECCS and CSS, including Criteria 5, long-term cooling, are addressed in DCD Chapter 15, Transient and Accident Analyses. • Performance of long-term recirculation is evaluated by adopting NEI 04-07 methodology.	

Table 1.9.3-1 Conformance with Generic Issues (sheet 28 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #191 Assessment of Debris Accumulation on PWR Sump Performance NRC priority: HIGH (continued)		• The break properties (e.g., sizes, locations) used in the NEI 04-07 methodology are considered for debris generation, and multiple potential debris sources, types and characteristics are considered. • US-APWR analysis conservatively assumes 100% of LOCA-related debris produced reaches the RWSP. Debris quantity calculations consider appropriate transport modes and mechanisms for LOCA phases and conditions, and are consistent with NEI 04-07 guidance and recommendations. RWSP transport and suction strainer performance computations consider appropriate bulk flow velocities and other LOCA-related hydrodynamic phenomena and forces. ***Technical Specification LCOs relevant to sump availability and performance*** • LCO 3.5.2 and 3.6.6 – Require operability of containment spray system, including containment sumps and screens as part of the flow path, and correct positioning of containment spray isolation valves.	

Table 1.9.3-1 Conformance with Generic Issues (sheet 29 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
		• LCO 3.5.4 – Refueling Water Storage Pit – Ensures sufficient water volume exists in the containment sump to support continued operation of the SI and CS/RHR, and that the sump pH is maintained in an acceptable range. • LCO 5.0 – Administrative Controls – 5.5.8, Inservice Testing Program, requires periodic surveillance of safety equipment to ensure functionality, which will include the containment sumps and screens. Conclusions: The US-APWR ECCS will be designed in accordance with Regulatory Guide 1.82, revision 3. Four redundant strainer systems will be installed within the RWSP, which has a broad footprint to obtain sufficient surface area. Particulate insulations such as calcium silicate are excluded from the US-APWR containment, to reduce potential sources of debris that would significantly increase head loss through the sumps. Coatings debris is estimated by the NEI 04-07 methodology, and 200 pounds of latent debris is assumed to reach each strainer location. NaTB is selected as the agent for pH control in the recirculation water inside containment, to mitigate the chemical effect that might be caused during long term cooling. The US-APWR design selects disk-type strainer systems that are currently available, avoiding the application of conventional flat-screen strainer design.	

Table 1.9.3-1 Conformance with Generic Issues (sheet 30 of 30)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/Sec.
New Generic Issue #193 BWR ECCS Suction Concerns NRC priority: CONTINUE	Does not apply to US-APWR.	Not applicable to US-APWR design certification. (Applicable only to boiling water reactors (BWRs))	NA

Table 1.9.3-2 Location of Description for Additional TMI-Related Requirements (sheet 1 of 14)

50.34(f) Item	Action Plan Item	Requirement	Location in DCD
(1)(i)	II.B.8	Perform a plant/site-specific PRA, 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.	19.1, 19.2.6
(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	10.4.9, 15.2.7, 15.2.8, 19.1.4.1.2
(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 LOOP. If damage cannot be precluded, provide an analysis of the limiting small-break LOCA with subsequent reactor coolant pump seal damage.	9.2.2.2.2.4, 15.0.0.9, 19.1.4.1
(1)(iv)	II.K.3.2	Perform an analysis of the probability of a small-break 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 RCS pressure falls after the PORV has opened. (PWRs only)	US-APWR does not have a pressurizer PORV. However, description of SB-LOCA with SDV open is in 19.1.4.1.

Table 1.9.3-2 Location of Description for Additional TMI-Related Requirements (sheet 2 of 14)

50.34(f) Item	Action Plan Item	Requirement	Location in DCD
(1)(v)	II.K.3.13	Perform an evaluation of the safety effectiveness of providing for separation of high-pressure coolant injection (HPCI) and RCIC system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and of providing that both systems restart on low water level. (For plants with high-pressure core spray [HPCS] systems in lieu of HPCI systems, substitute the words, “high-pressure core spray” for “high-pressure coolant injection” and “HPCS” for “HPCI”.) (BWRs only)	N/A (BWRs only)
(1)(vi)	II.K.3.16	Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. (BWRs only)	N/A (BWRs only)
(1)(vii)	II.K.3.18	Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system (ADS) design modifications that would eliminate the need for manual activation to ensure adequate core cooling. (BWRs only)	N/A (BWRs only)
(1)(viii)	II.K.3.21	Perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present. (BWRs only)	N/A (BWRs only)

Table 1.9.3-2 Location of Description for Additional TMI-Related Requirements (sheet 3 of 14)

50.34(f) Item	Action Plan Item	Requirement	Location in DCD
(1)(ix)	II.K.3.24	Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the RCIC and HPCI systems, following a complete LOOP to the plant for at least 2 hours. (For plants with high-pressure core spray [HPCS] systems in lieu of high-pressure coolant injection systems, substitute the words, “high-pressure core spray” for “high-pressure coolant injection” and “HPCS” for “HPCI”.) (BWRs only)	N/A (BWRs Only)
(1)(x)	II.K.3.28	Perform a study to ensure that the automatic depressurization system, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves. (BWRs only)	N/A (BWRs only)
(1)(xi)	II.K.3.45	Provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown. (BWRs only)	N/A (BWRs only)
(2)(i)	I.A.4.2	Provide a simulator capability that correctly models the control room and includes the capability to simulate small break LOCAs. (Applicable to construction permit applicants only)	N/A (COLA only)

Table 1.9.3-2 Location of Description for Additional TMI-Related Requirements (sheet 4 of 14)

50.34(f) Item	Action Plan Item	Requirement	Location in DCD
(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 [the Institute of Nuclear Power Operations (INPO)] and other industry efforts. (Applicable to construction permit applicants only)	N/A (COLA only)
(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.	7.5.1, 18.7.2., 18.7.2.1, 18.7.2.2, 18.7.2.3, 18.7.2.4, 18.7.2.5, 18.7.2.6, 18.7.3.1, 18.7.3.2
(2)(iv)	I.D.2	Provide a plant safety parameter display console that will display 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.	7.1.1.5.4, 7.5.1.4, 13.3, 18.7.2.5

Table 1.9.3-2 Location of Description for Additional TMI-Related Requirements (sheet 5 of 14)

50.34(f) Item	Action Plan Item	Requirement	Location in DCD
(2)(v)	I.D.3	Provide for automatic indication of the bypassed and operable status of safety systems.	7.1.1.5.2, 8.1.5.3, Table 8.1-1, 18.7.3.2, Table 18.7-1
(2)(vi)	II.B.1	Provide the capability of high-point venting of noncondensible 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.	5.4.12, 19.2.3.3.9

Table 1.9.3-2 Location of Description for Additional TMI-Related Requirements (sheet 6 of 14)

50.34(f) Item	Action Plan Item	Requirement	Location in DCD
(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 term[11] radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. 11 Footnote 11 in 10 CFR 50.34(f) reads as follows: "The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products."	12.2.1.3, 12.4.1.8
(2)(viii)	II.B.3	Provide a capability to promptly obtain and analyze samples from the RCS and containment that may contain accident source term[11] 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, and nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations.	9.3.2.2.3, 12.4.1.8

Table 1.9.3-2 Location of Description for Additional TMI-Related Requirements (sheet 7 of 14)

50.34(f) Item	Action Plan Item	Requirement	Location in DCD
(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.	6.2.5.1
(2)(x)	II.D.1	Provide a test program and associated model development, and conduct tests to qualify RCS relief and safety valves and, for PWRs, PORV block valves, for all fluid conditions expected under operating conditions, transients, and accidents. Consideration of 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.	14.2.12.1.4

Table 1.9.3-2 Location of Description for Additional TMI-Related Requirements (sheet 8 of 14)

50.34(f) Item	Action Plan Item	Requirement	Location in DCD
(2)(xi)	II.D.3	Provide direct indication of relief and safety valve position (open or closed) in the control room.	5.2.2.8
(2)(xii)	II.E.1.2	Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide AFW system flow indication in the control room. (PWRs only)	Table 7.5-3, 10.4.9, 15.0.0.3, 15.2.7.2
(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)	5.4.10.3.1, Table 8.3.1-4

Table 1.9.3-2 Location of Description for Additional TMI-Related Requirements (sheet 9 of 14)

50.34(f) Item	Action Plan Item	Requirement	Location in DCD
(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 lines) have two isolation barriers in series (C) Do not result in reopening of the containment isolation valves on resetting of the isolation signal (D) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation (E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs	6.2.4.1, Table 6.2.4-3, Figure 6.2.4-1
(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.	6.2.4, Table 6.2.4-3,
(2)(xvi)	II.E.5.1	Establish a design criterion for the allowable number of actuation cycles of the ECCS and reactor protection system consistent with the expected occurrence rates of severe overcooling events (considering both anticipated transients and accidents). (B&W designs only)	N/A B& W Design Only

Table 1.9.3-2 Location of Description for Additional TMI-Related Requirements (sheet 10 of 14)

50.34(f) Item	Action Plan Item	Requirement	Location in DCD
(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.	6.2.5, Table 7.5-3, 11.5.4.1, 12.4.1.8, 19.2.3.3.7
(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.	4.4.6.4, 7.5.1.1.3,
(2)(xix)	II.F.3	Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage.	7.5.1.1, 19.2.3.3.7
(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)	7.1.1.10, 7.4.1.6

Table 1.9.3-2 Location of Description for Additional TMI-Related Requirements (sheet 11 of 14)

50.34(f) Item	Action Plan Item	Requirement	Location in DCD
(2)(xxi)	II.K.1.22	Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable. (BWRs only)	N/A (BWRs Only)
(2)(xxii)	II.K.2.9	Perform a failure modes and effects analysis of the integrated control system (ICS) to include consideration of failures and effects of input and output signals to the ICS. (B&W designs only)	N/A (B& W design only)
(2)(xxiii)	II.K.2.10	Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip. (B&W designs only)	N/A (B& W design only)
(2)(xxiv)	II.K.3.23	Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. (BWRs only)	N/A (BWRs only)
(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.	7.5.1.6, 13.3, 18.7.3.1 (OSC is addressed as COL item.)

Table 1.9.3-2 Location of Description for Additional TMI-Related Requirements (sheet 12 of 14)

50.34(f) Item	Action Plan Item	Requirement	Location in DCD
(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[11] radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and the public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency.	Table 6.3-1(2/2)
(2)(xxvii)	III.D.3.3	Provide for monitoring of in-plant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions.	7.3.1.5, 7.7.1.11, Table 7.5-3, 11.5, 12.3
(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[11] release, and make necessary design provisions to preclude such problems.	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.	1.9.4, 13.5.1 (partly addressed in COLA)
(3)(ii)	I.F.1	Ensure that the QA list required by Criterion II in Appendix B to 10 CFR Part 50 includes all SSC important to safety.	17.5

Table 1.9.3-2 Location of Description for Additional TMI-Related Requirements (sheet 13 of 14)

50.34(f) Item	Action Plan Item	Requirement	Location in DCD
(3)(iii)	I.F.2	Establish a 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 QA/quality control (QC) 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.	17.5
(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.	19.2.3.3.9
(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.	N/A (No hydrogen recombiners. Igniters only.)

Table 1.9.3-2 Location of Description for Additional TMI-Related Requirements (sheet 14 of 14)

50.34(f) Item	Action Plan Item	Requirement	Location in DCD
(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.	N/A (Should be addressed in COLA)

1.9.4 Operational Experience (Generic Communications)

Language from Reg Guide 1.206 section C.I.1.9.4, and from Standard Review Plan (SRP) 1.0, “Introduction and Interfaces”, section I.9 states that the applicant “must include information to demonstrate how operating experience insights from generic letters and bulletins issued after the most recent revision of the applicable SRP and 6 months before the docket date of the application, or comparable international operating experience, have been incorporated into the plant design”. As of this writing, there have been no bulletins or generic letters issued in the time frame following the March 2007 revision of the SRP.

The US-APWR is an evolutionary nuclear power plant based on earlier Westinghouse PWR technology and improved by factoring in additional PWR operating experience in Japan. Since the US-APWR is most closely related to other Mitsubishi-designed PWRs, this DCD section primarily will demonstrate how relevant Japanese operating experience has been incorporated into the US-APWR design.

1.9.4.1 MHI Progression of Experience with PWRs

The US-APWR design features incorporation of lessons learned by:

- 20 years of joint development and technology transfer between MHI and Westinghouse Electric Company.
- More than 15 years supplying large components and meeting quality assurance requirements for reactor vessels, steam generators, pressurizers and turbines to nuclear plants in North America, Europe, and Asia
- Dominance of the pressurized water reactor market in Japan since inception in 1970
- More than 400 cycles of core PWR design and 15,000 fuel bundles fabricated

Of Japan’s 55 operating commercial nuclear plants, 23 are PWRs. Table 1.9.4-1 displays this group of plants including eight 2-loop plants, eight 3-loop plants, and seven 4-loop plants. MHI has been involved in the design and construction of all 23 PWRs, with four of the early designs featuring Westinghouse primary sides with secondary side by MHI, and the remaining nineteen featuring primary and secondary sides by MHI. This high level of involvement in the PWR market in Japan has afforded MHI with the opportunity to adopt U.S. technology and make enhancements, as each generation of PWR has evolved along with operational experience and regulatory changes.

Table 1.9.4-1 Summary of Japanese PWR Plants

Type Plant	Plant Name	Owner	Electric Output (MWe)	Commercial Operation Date
2-loop PWR	Mihama No. 1*	The Kansai Electric Power Co., Inc.	340	November 1970
	Mihama No. 2	The Kansai Electric Power Co., Inc.	500	July 1972
	Genkai No. 1	Kyushu Electric Power Co., Inc.	559	October 1975
	Ikata No. 1	Shikoku Electric Power Co., Inc.	566	September 1977
	Genkai No. 2	Kyushu Electric Power Co., Inc.	559	March 1981
	Ikata No. 2	Shikoku Electric Power Co., Inc.	566	March 1982
	Tomari No. 1	Hokkaido Electric Power Co., Inc.	579	June 1989
	Tomari No. 2	Hokkaido Electric Power Co., Inc.	579	April 1991
3-Loop PWR	Takahama No. 1*	The Kansai Electric Power Co., Inc.	826	November 1974
	Takahama No. 2	The Kansai Electric Power Co., Inc.	826	November 1975
	Mihama No. 3	The Kansai Electric Power Co., Inc.	826	December 1976
	Sendai No. 1	Kyushu Electric Power Co., Inc.	890	July 1984
	Takahama No. 3	The Kansai Electric Power Co., Inc.	870	January 1985
	Takahama No. 4	The Kansai Electric Power Co., Inc.	870	June 1985
	Sendai No. 2	Kyushu Electric Power Co., Inc.	890	November 1985
	Ikata No. 3	Shikoku Electric Power Co., Inc.	890	December 1994
4-Loop PWR	Ohi No. 1*	The Kansai Electric Power Co., Inc.	1175	March 1979
	Ohi No. 2*	The Kansai Electric Power Co., Inc.	1175	December 1979
	Tsuruga No. 2	The Japan Atomic Power Co., Inc.	1160	February 1987
	Ohi No. 3	The Kansai Electric Power Co., Inc.	1180	December 1991
	Ohi No. 4	The Kansai Electric Power Co., Inc.	1180	February 1993
	Genkai No. 3	Kyushu Electric Power Co., Inc.	1180	March 1994
	Genkai No. 4	Kyushu Electric Power Co., Inc.	1180	July 1997

Notes:
* Indicates Primary system by Westinghouse Electric Company and Secondary system by MHI. All others are primary and secondary systems by MHI

More than thirty years have passed since nuclear power generation started on a commercial basis in Japan, and it now supplies approximately one-third of the country's electric power. Beginning with introducing the technology from abroad, MHI has progressed with the acquisition and improvement of the technology in every field of design, manufacturing, construction, operation, and maintenance. MHI has proceeded with the design of new plants based upon the experience accumulated to date, while also developing measures to maintaining operation of the existing fleet. Comparison of the proposed US-APWR design to similar operating plants, therefore, involves examination of lessons learned from all of the operating PWRs in Japan.

MHI entered into an agreement with Westinghouse Electric Company of the United States in 1959, with the purpose of introducing PWR technology in Japan. Based upon those early technology transfers and independent technology development, MHI has built PWRs throughout Japan - a total of 23 over the years, as indicated in table 1.9.4-1. Mihama Unit 1, the first Japanese PWR, began commercial operation in 1970, and Genkai Unit 4, the most recent Japanese PWR, commenced commercial operation in 1997. The fleet of PWRs in Japan can be classified into three-generation groups, the first group started commercial operation in 1970's, the second in 1980's and the third in

1990's. The first generation group (nine units) includes imported plants or domestic plants based on imported technology. The second-generation group (seven units) was constructed based on MHI's own technology, developed through the experience in construction and operation of the first group of plants. The third generation group (seven units) has been further improved based on domestic technology. Since introducing the technology for the first generation plants such as Mihama Unit 1, MHI has promoted domestic production of equipment and improvement of the technology introduced. During the ten years plus after nuclear power generation was introduced, MHI experienced problems such as bowing of fuel rods, crossflow from core baffle plate clearance, and degradation of steam generator (SG) tubes. To overcome these problems, MHI made thorough investigations of the root causes and took comprehensive measures such as improving the design of equipment in terms of materials and structures, improving operational controls (such as water quality), and implementing improved inspection and corrective maintenance techniques.

In the second-generation group, many design improvements were reflected based upon the experience obtained in the first generation. Modifications of inspection and maintenance procedures through the adoption of integrated reactor vessel head structures, increase of space for maintenance and inspection, and automated maintenance and inspection processes resulted in the shortening of periodic inspection durations and subsequent reduction of occupational radiation exposures. Some equipment was simplified as unit capacities increased, such as the adoption of super-sized moisture separators and heaters. Plant reliability was also increased by application of an integrated low-pressure turbine rotor. Along with the increased excellence of the utility companies in maintenance and operations, these improvements and enhancements contributed to increased plant availability. In the third generation group, the matured Japanese version PWR plants realized further improvements in operability, economic efficiency, and plant performance through the application of advanced technology such as digital controllers, an advanced main control board, and optimization of the original system and equipment designs and plant layouts. These additional improvements, based on operating experience, achieved another increment of simplification and operability of systems and equipment in Japan’s PWR fleet.

1.9.4.2 Plant Reliability and Safety Improvements Guided by Operating and Regulatory Experience

MHI has incorporated into the US-APWR design many measures to improve safety against internal events, including abnormal operational events and failures, detection and control of design basis accidents, and mitigation of severe accidents. The following table 1.9.4-2 summarizes the fundamental approaches to US-APWR design for these three classes of events, and safety and reliability improvements made in the US-APWR design. As indicated in the table, operating and regulatory experience has been utilized in all areas to guide the design enhancements.

Table 1.9.4-2 Summary of Major Reliability and Safety Improvements Guided by Operating and Regulatory Experience (sheet 1 of 2)

Class of Events	Fundamental Elements of Design US-APWR Approach	US-APWR Reliability and Safety Improvements
Prevention of abnormal operation and failures	• Conservative design and high quality in construction and operation • Careful selection of materials and use of qualified fabrication processes • Margins in the design of systems and plant components • Utilization of operating and regulatory experience	• Enhanced reliability of reactor coolant pressure boundary • Alloy 690 used at vessel head nozzle and T-cold at vessel head plenum temperature to provide greater resistance to primary water stress corrosion cracking (PWSCC) • Reduction of neutron fluence to reactor vessel using neutron reflector • Increased reliability of steam generators with high performance primary moisture separators and anti-vibration bars in tube bundle area • Advanced seal design for reactor coolant pumps and improved flow rate with improved impeller and diffuser designs • Shorter grid spacing and bottom nozzle debris filters in fuel assemblies reduces likelihood of damage to fuel Improved maintenance • Enhanced safety during on line maintenance using 4 train safety systems • Improved access and maintenance and no penetrations to the bottom reactor head, by adoption of upper mounted incore instrumentation system • Enhanced reliability during shut down operation • Shortened duration of mid loop operation • Automatic interlock to isolate the letdown line below mid loop water level • Reduction of operator actions • Computerized control room with enhanced operability • Enhanced reliability of I & C systems • Redundant digital control systems

Table 1.9.4-2 Summary of Major Reliability and Safety Improvements Guided by Operating and Regulatory Experience (sheet 2 of 2)

Class of Events	Fundamental Elements of Design US-APWR Approach	US-APWR Reliability and Safety Improvements
Detection of failures, control of abnormal plant states and accidents within the design basis	• Robust protection systems • Robust engineered safety features and critical support systems • Redundancy • Separation • Utilization of operating and regulatory experience	• Enhanced reliability of shut down capability • Reactor protection system - 4 train system with 4 train reactor trip breakers • Cold shutdown with safety components - Emergency core cooling system, Emergency letdown line, safety depressurization and vent system • Enhanced reliability of emergency core cooling systems • Advanced accumulator and 4 train high head safety injection system – direct vessel injection and no interconnection between trains • Longer accumulator injection period allows more time for safety injection pumps to start and reach design flow • Elimination of switchover of ECCS suction by installation of the refueling water storage pit inside the containment • Enhanced reliability of containment cooling system • 4-train containment spray system • Enhanced reliability of support systems - 4 train electric power system, CCWS, service water system, etc. • Installation of high reliability emergency gas turbine generators
Control of beyond design basis accidents	• Supplemental measures and accident management • Diverse measures against the design basis accidents • Utilization of operating and regulatory experience	• Enhanced measures against station blackout • Diverse ac/AC power sources • Enhanced measures against Interfacing systems LOCA • Upgraded piping of residual heat removal system • Measures against common mode failures in digital safety system • Diverse actuation functions (reactor trip, turbine trip, emergency feed water system initiation) • Enhanced measures against severe accidents after core damage • Design features to reduce hydrogen detonation, molten core concrete interaction, high pressure melt ejection

Digital Technology

With regard to its history of applications of digital technology in nuclear plants, MHI introduced the technology in operating plants for improvements of plant safety, availability, operability and monitoring capability. The technology achieves an excellent balance between cost and performance, and has been manufactured in compliance with US codes and standards. The phase-in of digital technology has been conservative with applications thus far made to all non-safety instrumentation and controls. With multiple applications in each plant, MHI has now made digital installations in five plants, each with an average of 10 years operation, for a total of over 20 million hours operating experience. No system malfunctions have been caused by failures in either software or hardware, and MHI is confident that this excellent non-safety history has provided ample operating experience for digital application to safety systems and the human-system interface (HSI) System of the US-APWR. The US-APWR features a fully digital I&C System and software-based HSI for all control and monitoring. The compact main control panel, featuring a minimum inventory of fixed position and conventional HSI devices, can be maneuvered by a single operator if required in normal operation or in a potential accident scenario. Conventional HSI is provided for manual system level actuation (RG 1.62) and critical functions and bypass or inoperable status functions (RG 1.47). The systems are based on the defense-in-depth and diversity concepts, with 4-train redundant configurations for safety systems and redundant configurations for non-safety systems. These systems feature maximum standardization and diverse electrical power backup for operational assurance.

1.9.4.3 Design Responses to Reportable Events at Operating PWRs

Event reports issued by the Japan Nuclear Energy Safety Organization (JNES), as required by Japan's "Law for the Regulation on Nuclear Source Material, Nuclear Fuel Material, and Reactors" and the "Electric Utility Law" are shown in Table 1.9.4-3. These events are for the 10-year period 1996 through 2006. In keeping with the intent of this section of the DCD to describe experience with similar power plants, events for operating Japanese PWRs are shown. A total of 63 PWR events are shown for the 10-year period.

JNES uses the International Atomic Energy Agency (IAEA) International Nuclear Event Scale (INES) to rate reportable events at its nuclear power plants (NPPs). This scale ranges from a value of zero (0), which is considered to be a deviation with no safety significance, to a value of seven (7), which is considered to be a major accident such as the Chernobyl event. All of the events in the table below are rated at a value of not greater than one (1) (considered to be an anomaly with no radioactive contamination), with the vast majority rated at either zero or "below scale".

JNES is a technical support organization for Japan's Nuclear and Industrial Safety Agency (NISA). In the table, the information in the "Date of Occurrence", "Power Station" and "Description" columns are reproduced from the JNES website. The columns "MHI Comment" indicates whether the event was considered to have a design impact, and if so, the action taken by MHI.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 1 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
May 21, 1996	Ikata Power Station (Unit 2)	A signification indication was found at the location of the tube expansion area at the hot leg side as a result of the eddy current test of the SG tubes during periodical inspection.	This event was attributed to PWSCC in the steam generator tubes, which were fabricated from Alloy 600 MA. Impacts on the US-APWR design have included adoption of Alloy 690 TT for its excellence in corrosion resistance; reduction in the difference between tube plate diameter and outer diameter of the heat transfer tubes, and reduction in residual stress by changing from the conventional roll tube expansion method.
August 25, 1996	Ohi Power Station (Unit 2)	During rated power operation, the reactor was manually shutdown, as the temperature at the lower part bearing in the motor of reactor coolant pump A was showing a tendency to increase. The cause was as follows: The flow of cooling water for the lower part of the motor bearing decreased due to a blockage of foreign material in the cooling water outlet valve of the bearing.	This event was attributed to faulty workmanship, and has had no impact on the US-APWR design.
September 16, 1996	Ohi Power Station (Unit 4)	During rated power operation, the alarms "Generator Internal Failure" and "Main Transformer Internal Earth Fault" actuated leading to generator and reactor automatic shutdown. The cause was as follows: During generators manufacturing stage, the phase ring connecting with the stator winding phase lead was not sufficiently fixed, so that the phase wire led to a fatigue break due to vibration etc. during plant operation. As a result, there was an electric discharge and a short circuit was produced at another phase lead.	This event was attributed to faulty fabrication, and has had no impact on the US-APWR design.
October 18, 1996	Ikata Power Station (Unit 1)	A significant indication was found at the location of the tube expansion area at the hot leg side as a result of the eddy current test of the SG tubes during periodical inspection.	This event was attributed to PWSCC in the steam generator tubes, which were fabricated from Alloy 600 MA. Impacts on the US-APWR design have included adoption of Alloy 690 TT for its excellence in corrosion resistance; reduction in the difference between tube plate diameter and outer diameter of the heat transfer tubes, and reduction in residual stress by changing from the conventional roll tube expansion method.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 2 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
December 24, 1996	Tsuruga Power Station (Unit 2)	During rated power operation, the reactor was manually shutdown, as the patrol found borated water leaking from the chemical and volume control system in the reactor containment vessel. The cause was as follows: During the manufacturing stage of the concerned elbow, foreign material adhered to the inner side of pipe. Cracks generated and evolved around the low melting point metal, leading to the through-wall cracks and leakage.	This event was attributed to faulty fabrication, and has had no impact on the US-APWR design.
May 9, 1997	Takahama Power Station (Unit 2)	During periodical inspection, the reactor had been started up and was under the critical state. A signal ¡Intermediate Range Neutron Flux High" was generated, and the reactor was automatically shutdown. The cause was as follows: During the work to adjust the trip setpoint of the intermediate range nuclear instrumentation unit (N-36), the trip setpoint next to the concerned unit card in the same control panel was also performed. At this time, the neutron flux detector was not isolated. An abnormal electric current was generated in the control power supply circuit of N-36 by the noise from the concerned detector which led to the fuse blowing, resulting in loss of the control power.	This event was attributed to faulty maintenance. Measures in the US-APWR design are taken, however, for the prevention of this type of overcurrent due to magnetic saturation.
September 1, 1997	Genkai Nuclear Power Station (Unit 2)	A significant indication was found at the location of the tube expansion area at the hot leg side as a result of the eddy current test of the SG tubes during periodical inspection.	This event was attributed to PWSCC in the steam generator tubes, which were fabricated from Alloy 600 MA. Impacts on the US-APWR design have included adoption of Alloy 690 TT for its excellence in corrosion resistance; reduction in the difference between tube plate diameter and outer diameter of the heat transfer tubes, and reduction in residual stress by changing from the conventional roll tube expansion method.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 3 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
October 3, 1997	Ikata Power Station (Unit 2)	A significant indication was found at the location of the tube expansion area at the hot leg side as a result of the eddy current test of the SG tubes during periodical inspection.	This event was attributed to PWSCC in the steam generator tubes, which were fabricated from Alloy 600 MA. Impacts on the US-APWR design have included adoption of Alloy 690 TT for its excellence in corrosion resistance; reduction in the difference between tube plate diameter and outer diameter of the heat transfer tubes, and reduction in residual stress by changing from the conventional roll tube expansion method.
November 10, 1998	Sendai Nuclear Power Station (Unit 1)	During rated power operation, it was found that the drain flow rate into the containment sump was increased. The reactor was manually shutdown to perform the check and the investigation. The cause was as follows: Foreign materials in the piping stuck to seat of the drain valve during the previous periodic inspection. A part of the foreign material was blown out due to the change of containment coolant drain tank water level and thus there remained gaps between the seat and the disk of the concerned valve. Consequently, the leakage from the closed drain valve occurred.	This event was attributed to faulty maintenance, and has had no impact on the US-APWR design.
November 30, 1998	Genkai Nuclear Power Station (Unit 2)	During periodic inspection, the eddy current test of steam generator tubes was performed. It was found that there were significant indications at the tube expansion zone of the hot leg side.	This event was attributed to PWSCC in the steam generator tubes, which were fabricated from Alloy 600 MA. Impacts on the US-APWR design have included adoption of Alloy 690 TT for its excellence in corrosion resistance; reduction in the difference between tube plate diameter and outer diameter of the heat transfer tubes, and reduction in residual stress by changing from the conventional roll tube expansion method.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 4 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
December 1, 1998	Ohi Power Station (Unit 2)	During periodic inspection, it was found that there was a crack near the joint portion of the drain valve of the Residual Heat Removal (RHR) pump outlet piping. The cause was as follows: The natural frequency of the concerned valve became the very close to the frequency of the flow induced vibration by the pump because the concerned drain valve was replaced by the new one during the previous periodic inspection. This led to the concerned drain valve's resonance whenever the pump was operated. The fatigue crack due to the welding defect occurred and developed and resulted in the trough-wall crack.	This event was determined to have been caused by faulty welding in the replacement of an RHR drain valve, but could also have had a design component. For small diameter piping likely to suffer from high cycle fatigue due to vibration, MHI will specify and/or recommend for the US-APWR butt welding techniques as much as possible, to improve reliability of the welded joints.
January 29, 1999	Genkai Nuclear Power Station (Unit 1)	During rated power operation, it was found that the seal water return flow of one of Reactor Coolant Pumps (RCP) was gradually increased. The reactor was manually shutdown to perform the check and the investigation.	This event was attributed to faulty maintenance, and has had no impact on the US-APWR design.
January 29, 1999	Ohi Power Station (Unit 2)	During adjustment operation, one control rod which was being withdrawn dropped into the reactor core. The reactor was manually shutdown to perform the check and the investigation. Another control rod slipped during shutdown operation.	This event was attributed to faulty maintenance, and has had no impact on the US-APWR design.
February 18, 1999	Ikata Power Station (Unit 2)	During periodic inspection, the eddy current test of steam generator tubes was performed. It was found that there were significant indications at the tube expansion zone of the hot leg side.	This event was attributed to PWSCC in the steam generator tubes, which were fabricated from Alloy 600 MA. Impacts on the US-APWR design have included adoption of Alloy 690 TT for its excellence in corrosion resistance; reduction in the difference between tube plate diameter and outer diameter of the heat transfer tubes, and reduction in residual stress by changing from the conventional roll tube expansion method.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 5 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
April 30, 1999	Mihama Power Station (Unit 2)	During operation at 330MWe power to inspect and repair the condensers, it was found that the drain flow rate into the containment sump increased. It was recognized that there was a small leakage from the excess letdown system piping. The reactor was manually shutdown to perform the inspection and the investigation. The cause was that the repeated thermal stress due to the change of the thermal stratification was generated at the elbow of the concerned piping.	This event was attributed to repeated thermal stress and fatigue in the CVCS excess letdown piping due to fluctuations in thermal stratification in the flow. Impact of the event on the US-APWR design is that routing of piping will be arranged so that thermal stratification effects such as this are considered and prevented.
May 27, 1999	Takahama Power Station (Unit 4)	During periodic inspection, the eddy current test of all tubes of three steam generators was performed. As a result of the test, it was recognized that there were significant indications at the tube expansion zone of the hot leg side.	This event was attributed to PWSCC in the steam generator tubes, which were fabricated from Alloy 600 TT. Impacts on the US-APWR design have included adoption of Alloy 690 TT for its excellence in corrosion resistance; reduction in the difference between tube plate diameter and outer diameter of the heat transfer tubes, and reduction in residual stress by changing from the conventional roll tube expansion method.
July 5, 1999	Takahama Power Station (Unit 4)	During periodic inspection, the reactor operation was started and was the critical condition. It was found that small amount of boric acid was precipitated near the seal portion of the in-core neutron flux monitoring guide tube. Therefore, the reactor was manually shut down. The cause was as follows: When the seal portion was inspected during this periodic inspection, a tool had touched with the seal, and it caused the small scratch. The reactor coolant leaked through a micro-size gap made by the scratch.	This event was attributed to faulty maintenance, and has had no impact on the US-APWR design.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 6 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
July 12, 1999	Tsuruga Power Station (Unit 2)	During rated power operation, the containment sump level alarm was initiated, indicating high level increasing rate. It was also recognized that the dust monitor indication inside reactor containment was increasing. Therefore, the reactor was manually shut down. The cause was the high cycle thermal fatigue caused by the inadequate structure of the regenerative heat exchangers.	The regenerative heat exchanger consists of an inner tube, where the main flow is cooled by heat transfer with a high temperature bypass flow that occurs in the clearance between the inner tube and shell. The cause of the damage to the heat exchanger was found to be thermal fatigue by temperature fluctuations. In consideration of the proven causes for this event, the US-APWR regenerative heat exchanger design will not be equipped with an inner tube, and the thermal fluctuations associated with confluence of main and bypass flows will be eliminated.
August 25, 1999	Sendai Nuclear Power Station (Unit 1)	During rated power operation, the reactor was automatically shut down by automatic steam turbine trip signal due to "turbine solenoid valve actuation". The cause was as follows: When the solenoid operated valve housing case was installed during the previous periodic inspection, several piping connections were insufficiently tightened. The insufficient tightening caused a gap. The O-ring was partly thrust into the gap, and resulted in fail.	This event was attributed to faulty maintenance, and has had no impact on the US-APWR design.
February 19, 2000	Ohi Power Station (Unit 2)	During inspection of the condenser tubes after reducing the electric power to about 60% of rated power since February 14, 2000, the turbine was manually shut down due to decreasing of the condenser vacuum, and the reactor resulted in automatic shutdown. The cause of decreasing of the condenser vacuum was that the vacuum pump performance was degraded due to freezing of the air ejector nozzle portion. The cause of manual shutdown of the turbine was that the operator mistook the value of the generator output for the value of the condenser vacuum on the CRT display.	The decreased vacuum pump performance aspect of this event was attributed to expected aging of the component, and the manual turbine shutdown was attributed to operator error. This event has had no impact on the US-APWR design.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 7 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
March 16, 2000	Takahama Power Station (Unit 3)	When the eddy current test of steam generator tubes was performed during periodic inspection, it was recognized that there were significant changes in the eddy current in the tube expansion zone of the hot leg side.	This event was attributed to PWSCC in the steam generator tubes, which were fabricated from Alloy 600 TT. Impacts on the US-APWR design have included adoption of Alloy 690 TT for its excellence in corrosion resistance; reduction in the difference between tube plate diameter and outer diameter of the heat transfer tubes, and reduction in residual stress by changing from the conventional roll tube expansion method.
March 31, 2000	Genkai Nuclear Power Station (Unit 2)	When the eddy current test of steam generator tubes was performed during periodic inspection, it was recognized that there were significant changes in the eddy current in the tube expansion zone of the hot leg side.	This event was attributed to PWSCC in the steam generator tubes, which were fabricated from Alloy 600 MA. Impacts on the US-APWR design have included adoption of Alloy 690 TT for its excellence in corrosion resistance; reduction in the difference between tube plate diameter and outer diameter of the heat transfer tubes, and reduction in residual stress by changing from the conventional roll tube expansion method.
April 7, 2000	Mihama Power Station (Unit 2)	During rated power operation, it was found that there was small leakage of reactor coolant from letdown piping of the chemical and volume control system. Therefore, the reactor was manually shut down. The cause was as follows: The orifice's inside surface was dented due to cavitation which was generated in the de-pressurizing orifice area during temporary pressure reduction operation. This resulted in generation of cavitation even during normal operation. High cycle fatigue due to fluid vibration with this cavitation caused the crack generation and propagation in the elbow weld portion.	This event was attributed to faulty operation and has had no impact on the US-APWR design.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 8 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
April 29, 2000	Mihama Power Station (Unit 2)	During the power increase (electric power about 20%), the generator voltage decreased. Therefore, the voltage recovery operation was performed. During the operation, the generator was automatically shut down by the actuation of the relay due to the loss of generator field, which resulted in automatic shutdown of the reactor. The cause was estimated as follows: Clamping of the cable terminal of one-phase of the three-phase cable, which supplies power from the permanent magnet generator to the automatic voltage regulator, was insufficient. This caused heat generation of the terminal and resulted in burning of the terminal plate. The nut of the terminal was loosened and then the cable between the terminals was broken.	This event was attributed to faulty maintenance, and has had no impact on the US-APWR design.
May 26, 2000	Ikata Power Station (Unit 2)	During periodical inspection, the eddy current test of steam generator tubes was performed. As a result of the test, it was recognized that there were significant indications at the tube expansion zone of the hot leg side.	This event was attributed to PWSCC in the steam generator tubes, which were fabricated from Alloy 600 MA. Impacts on the US-APWR design have included adoption of Alloy 690 TT for its excellence in corrosion resistance; reduction in the difference between tube plate diameter and outer diameter of the heat transfer tubes, and reduction in residual stress by changing from the conventional roll tube expansion method.
September 1, 2000	Mihama Nuclear Power Station (Unit 3)	During periodical inspection, the eddy current test of steam generator tubes was performed. As a result of the test, it was recognized that there were significant indications at the tube expansion zone of the hot leg side. The cause was estimated as follows: The secondary product, which was generated by metal cutting work during the previous periodical inspection, contacted with the concerned tubes and resulted in abrasion of the tubes.	This event was attributed to faulty workmanship. However, a design improvement of installing strainers in the feedwater ring has been incorporated into the US-APWR design, for the purpose of preventing the introduction of foreign materials into the steam generators.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 9 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
September 14, 2000	Sendai Power Station (Unit 1)	During periodical inspection, the eddy current test of steam generator tubes was performed. As a result of the test, it was recognized that there were significant indications at the tube expansion zone of the hot leg side.	This event was attributed to PWSCC in the steam generator tubes, which were fabricated from Alloy 600 TT. Impacts on the US-APWR design have included adoption of Alloy 690 TT for its excellence in corrosion resistance; reduction in the difference between tube plate diameter and outer diameter of the heat transfer tubes, and reduction in residual stress by changing from the conventional roll tube expansion method.
October 2, 2000	Takahama Power Station (Unit 4)	During periodical inspection, the eddy current test of steam generator tubes was performed. As a result of the test, it was recognized that there were significant indications at the tube expansion zone of the hot leg side.	This event was attributed to PWSCC in the steam generator tubes, which were fabricated from Alloy 600 TT. Impacts on the US-APWR design have included adoption of Alloy 690 TT for its excellence in corrosion resistance; reduction in the difference between tube plate diameter and outer diameter of the heat transfer tubes, and reduction in residual stress by changing from the conventional roll tube expansion method.
October 13, 2000	Ikata Power Station (Unit 1)	During periodical inspection, When pressure test of small diameter pipe in the Primary System(a part of charging pipe in the Chemical and Volume Control System) was performed under the pipe replacement work, it was recognized that small amount of water leaked from the pipe. It was recognized that there was a flaw of about 5 mm in length in the vicinity of the weld portion of the pipe. The cause was estimated as follows: The vinyl chloride tape, which was furnished on the pipe at plant construction, remained. Chloride stress corrosion cracks on the outside surface of the pipe generated and developed. The pressure at pressure test of the pipe caused a wall-through of the crack.	This event was attributed to faulty workmanship, and has had no impact on the US-APWR design.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 10 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
November 15, 2000	Mihama Power Station (Unit 3)	Under adjustment operation, it was recognized that there was a small amount of steam leakage from the plug on the pipe used for piping radiation inspection. The reactor was manually shut down for inspection and repair. The cause was estimated as follows: Control of covered electrode during shielded metal arc welding of the concerned plug was insufficient. Therefore, a crack at low temperature generated. Afterwards, the crack developed during the plant operation and resulted in a wall-through.	This event was attributed to faulty workmanship, and has had no impact on the US-APWR design.
December 2, 2000	Ohi Power Station (Unit 1)	Under adjustment operation, it was recognized that oil leaked like mist from flange portion of oil transfer pipe for turbine steam governor valve actuator. The reactor was manually shut down for inspection and repair. The cause was estimated as follows: In the flange assembly work, the O-ring was pinched between flange surfaces due to the narrow work space and so on. The O-ring was damaged due to repeated high oil pressure, which resulted in a leakage.	This event was attributed to faulty workmanship, and has had no impact on the US-APWR design.
December 30, 2000	Ikata Power Station (Unit 1)	Its adjustment operation was initiated and its power was being increased. It was recognized that steam was leaking from a drain line valve of moisture separator and reheater relief valve main pipe in the Secondary Cooling System. Therefore, the reactor was manually shut down. The cause was as follows: The valve was disassembled and inspected in the area where dusts including salt generated. Dusts including salt intruded into the valve and the salt dissolved into dewed water. At the temperature condition with plant startup, chloride stress corrosion cracks generated.	This event was attributed to faulty maintenance, and has had no impact on the US-APWR design.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 11 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
July 6, 2001	Takahama Power Station (Unit 3)	During the periodical inspection, as a result of Eddy Current Test (ECT) of tubes of steam generators, defect indications were found. The location of these defects was the tube expansion zone of the hot leg side.	This event was attributed to PWSCC in the steam generator tubes, which were fabricated from Alloy 600 MA. Impacts on the US-APWR design have included adoption of Alloy 690 TT for its excellence in corrosion resistance; reduction in the difference between tube plate diameter and outer diameter of the heat transfer tubes, and reduction in residual stress by changing from the conventional roll tube expansion method.
September 8, 2001	Ikata Power Station (Unit 2)	During the periodic inspection, the rust was recognized on the outside surface of the in-core nuclear instrumentation device thimble guides. Therefore, liquid penetrant test (PT) of the concerned portion was performed. As a result of the test, defect indications on the outside surface of the concerned two guides were recognized. As a result of the investigation, these defect indications on the outside surface were so shallow as all of them vanish during the surface polishing process. The cause was estimated as follows: Vinyl chloride tape got slipped into the gap between the reactor vessel metal insulator portion and the iron plate mold frame of the surrounding concrete at construction phase. This caused generation of hydrogen chloride from vinyl chloride tape due to high radiation during the operation. This caused generation of liquid drops including iron oxy-hydroxide and ammonium chloride, and the liquid drops deposited on the surface of the thimble guides and resulted in shallow pit-shaped dents.	This event was attributed to faulty workmanship, and has had no impact on the US-APWR design.
January 30, 2002	Takahama Power Station (Unit 4)	During the periodical inspection, as a result of Eddy Current Test (ECT) of tubes of steam generators, defect indication was found. The location of the defect was the tube expansion zone of the hot leg side of the steam generator.	This event was attributed to PWSCC in the steam generator tubes, which were fabricated from Alloy 600 TT. Impacts on the US-APWR design have included adoption of Alloy 690 TT for its excellence in corrosion resistance; reduction in the difference between tube plate diameter and outer diameter of the heat transfer tubes; and reduction in residual stress by changing from the conventional roll tube expansion method.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 12 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
November 15, 2002	Mihama Power Station (Unit 3)	During rated power operation, a leakage from the vicinity of the vent valve weld portion, which was installed on the seal water injection line of one reactor coolant pump, was found on November 12, 2002. Therefore, necessary actions were taken and the reactor operation was continued. However, the amount of seal water injection increased and therefore, the reactor was manually shut down. As a result of the investigation, the cause was estimated to be as follows: In addition to insufficient welding of the concerned weld portion, stresses were repeatedly added to the concerned weld portion due to pressure pulsation generated from the valve, which regulates the amount of the seal water injection. Therefore, a crack in the concerned weld portion generated due to high-frequency fatigue and resulted in wall-through.	This event was attributed to faulty workmanship, and has had no impact on the US-APWR design.
December 12, 2002	Tsuruga Power Station (Unit 2)	During rated power operation, it was recognized that smoke was going up from the vicinity of the heat insulator near the high-pressure turbine casing cover. Therefore, removal of the heat insulator in the concerned portion was performed. During the removal work, fire ignition was recognized. Immediately, the fire was fought out by using fire extinguishers. However, because fire ignition was recognized again, the reactor was manually shut down. As a result of the investigation, the cause was estimated to be as follows: Lubricant oil was not drained and the exit pipe portion of the air ejector was blocked, due to slug accumulation on the U seal portion of the oil drain line of the main oil tank gas extractor system. Actuation of the air ejector under this situation led the released gas to the main oil tank. This led increasing of the No.2 bearing box internal pressure and it became positive pressure. The oil leaked from the concerned bearing portion and the leaked oil strained into the heat insulator. A fire ignited due to high environmental temperature around the heat insulator.	This event was attributed to faulty maintenance, and has had no impact on the US-APWR design.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 13 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
May 15, 2003	Sendai Nuclear Power Station (Unit 1)	During the periodical inspection, as a result of Eddy Current Test (ECT) of tubes of steam generators, defect indications were found. The location of these defects was the tube expansion zone of the hot leg side.	This event was attributed to PWSCC in the steam generator tubes, which were fabricated from Alloy 600 TT. Impacts on the US-APWR design have included adoption of Alloy 690 TT for its excellence in corrosion resistance; reduction in the difference between tube plate diameter and outer diameter of the heat transfer tubes, and reduction in residual stress by changing from the conventional roll tube expansion method.
May 22, 2003	Takahama Power Station (Unit 4)	During the periodical inspection, as a result of Eddy Current Test (ECT) of tubes of steam generators, defect indications were found. The location of these defects was the tube expansion zone of the hot leg side.	This event was attributed to PWSCC in the steam generator tubes, which were fabricated from Alloy 600 TT. Impacts on the US-APWR design have included adoption of Alloy 690 TT for its excellence in corrosion resistance; reduction in the difference between tube plate diameter and outer diameter of the heat transfer tubes, and reduction in residual stress by changing from the conventional roll tube expansion method.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 14 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
September 10, 2003	Tsuruga Power Station (No.2)	During the periodical inspection, when heat insulator was taken off to inspect the piping nozzle stub for pressurizer, boric acid precipitation was recognized on the surface of the piping nozzle stub for pressurizer relief valve. Ultrasonic test was performed on all piping nozzle stubs of the upper part of the pressurizer, including the piping nozzle stub concerned, and on piping nozzle stub for pressurizer surge. As a result of the test, significant indications were recognized on the piping nozzle stub for pressurizer relief valve and the piping nozzle stub for pressurizer safety valve. As a result of the investigation, it was found that the locations, where the indications were recognized, were both the weld portions modified. Therefore, the cause of this event was estimated to be PWSCC, which was generated by the combined three factors of increase of tensile stress of the circumference direction due to addition of its operational stress to residual stress (tensile stress) after modified welding of the piping nozzle stub portion, use of 600 type nickel-based alloy with SCC sensitivity for the concerned weld metal, and water quality environment of primary coolant.	This event was attributed to stress corrosion cracking (SCC) in the pressurizer piping nozzles. Impacts on the US-APWR design have included adoption of 690 series nickel based alloy, for its excellence in corrosion resistance, for the pressurizer nozzles, as well as continued emphasis on rigid controls on welding and primary coolant water quality.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 15 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
September 10, 2003	Tomari Power Station (Unit 2)	During rated thermal power operation, leakage from the regenerative heat exchanger room was recognized. After the system line of the regenerative heat exchanger room was isolated, it was recognized that there was leakage on the welding portion between the outlet piping nozzle stub of the shell side of the regenerative heat exchanger and the elbow portion. Based on the inspection result, the plant was shutdown to investigate in detail and repair the location of leakage. As a result of the investigation, the cause of this event was estimated that cracks were generated by changing stress due to temperature fluctuation (main factor) generated by mixture of main flow (low temperature water) and bypass flow (high temperature water) near the shell side outlet of the regenerative heat exchanger with inner tubes, and by changing stress due to mechanical vibration caused by small cavitation generated in the downstream side of the extracting orifice, developed, and resulted in wall-through.	The regenerative heat exchanger consists of an inner tube, where the main flow is cooled by heat transfer with a high temperature bypass flow that occurs in the clearance between the inner tube and shell. The cause of the damage to the heat exchanger was found to be thermal fatigue by temperature fluctuations near the outlet of the heat exchanger. In consideration of the proven causes for this event, the US-APWR regenerative heat exchanger design will not be equipped with an inner tube, and the thermal fluctuations associated with confluence of main and bypass flows will be eliminated.
November 9, 2003	Mihama Power Station (Unit 2)	During adjustment operation, leakage from the close plug of the vent line installed on the pressurizer spray pipe was recognized. Therefore, the reactor was shut down for inspection. As a result of the investigation, the cause of this event was estimated that a seat leakage of the concerned valve during the plant start-up generated due to decrease of holding force of the valve disc due to insufficient tightening of the pressurizer spray pipe vent valve and difference of thermal expansion between the valve stem and the valve box during the system temperature increase, and the O ring of the close plug had been damaged, resulting in leakage.	This event was attributed to faulty maintenance, and has had no impact on the US-APWR design.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 16 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
December 5, 2003	Ohi Power Station (Unit 1)	During rated thermal power operation, it was recognized that the water-level increasing rate of the containment sump A had shown the trend of increasing. Therefore, the reactor was manually shut down for inspection. As a result of the investigation, The cause of this event was estimated as follows: Surface roughness of the contact portion between the seal insert and the seal housing of the D-reactor coolant pump No.3 seal portion developed with progress of the unit's operation. The sliding resistance of the concerned contact portion became large and smooth movement of the seal ring for the upper and lower directions was obstructed. The close adhesion to the seal runner, which was retained by self-weight of the seal ring, etc. and the rebound spring, was lost. Finally, openness of the seat surface became large. Therefore, makeup water, which passed through the seat surface, increased, which resulted in leakage from the pump.	This event was attributed to faulty maintenance, expected aging, and deterioration from operational use, and has had no impact on the US-APWR design.
January 22, 2004	Takahama Power Station (Unit 3)	During the periodical inspection, when the Eddy Current Test (Intelligent ECT) of all tubes of steam generators was performed, there were significant signal indications found on the tubes, which showed denting of the tube outer surfaces. The ECT used multi-coil type probe. As a result of the investigation, the significant signal indications were found in the locations, where old anti-vibration holding bars were installed. Therefore, the cause of this event was estimated to be wearing and denting of tubes, which had generated in the concerned portion.	This event was attributed to faulty maintenance. MHI points out that conventional methods were unable to detect the damage, however, with .the improved detection accuracy of the Intelligent ECT, together with enhanced precision in depth assessment, successful detection was achieved. Impact on the US-APWR design has been the inclusion of an increased number of anti-vibration bars in the steam generators to reduce tube vibration.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 17 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
March 15, 2004	Ikata Power Station (Unit 3)	During rated thermal power operation, on March 9, 2004, the signal, which shows flow rate decrease of the seal water injection system to the primary coolant pump, actuated. Leakage of the cleaned-up primary coolant from the seal water portion of the charging pump C, which returns primary coolant to the primary coolant system after cleaning-up primary coolant, was recognized The charging pump 3C was stopped immediately and was switched to the pump 3B (spare pump). Operation of the reactor was continued. Afterwards, as a result of the inspection, on March 15, 2004, break of the concerned pump main shaft and failure of wear, etc. due to contacts inside the pump and at its seal water portion, etc. was recognized. As a result of the investigation, the cause of this event was estimated as follows : 1) In manufacturing process of the main shaft in factory, the radius of curvature of part of the seventh stage sprit ring groove portion, which was broken, was smaller than its design value and the stress concentration was large. The concerned split ring and the main shaft contacted to each other and stress generated in the concerned groove portion. 2) Because the charging pump was operated under the periodical inspection with pressure of the volume and control tank being open condition to atmosphere, air void generated at the orifice portion flew into the pump and vibration generated. At that time, stress generated at the seventh stage split ring groove portion. 3) Cracks of the main shaft of the charging pump C generated due to the combined factors above mentioned. 4) In the later periodical inspection, cracks developed due to the same mechanism, and finally resulted in break of the main shaft.	This event was attributed to faulty fabrication and operating of the pumps under condition of cavitation, and has had no impact on the US-APWR design.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 18 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
May 5, 2004	Ohi Power Station (Unit 3)	During the periodical inspection, when preparatory work for visual inspection of the piping nozzle stubs (70 locations in total) installed to the reactor vessel upper head was preformed, white adhesive material was recognized near the root of one location of piping nozzle stub (No. 47) for installment of the control rod drive mechanism. As a result of analysis of the adhesive material, it was recognized that the adhesive material was boric acid, which is included in primary coolant. Therefore, detailed visual inspection of the concerned piping nozzle stub was performed, and leakage from the concerned piping nozzle stub was recognized on May 5, 2004. Other 69 locations of piping nozzle stub were also inspected. As a result of the inspection, adhesive material was recognized on the piping nozzle stub (No. 67) for installment of the temperature meter. The leakage at the piping nozzle stub (No. 47) was estimated to be due to the combination of following 3 factors: 1) Condition (water chemistry of the primary cooling system), 2) Material (stress corrosion cracks could occur in the nickel-based alloy (600 type) used for welding in the present case) and 3) stress (as a result of visual inspection of the weld portion, its surface would be under tensile stress without buff finishing). These factors would have lead to the stress corrosion cracks and allowed them develop completely through to the other side, which eventually resulted in the leakage. As to the leakage at the piping nozzle stub (No. 67), it was estimated that leaked boric acid was not wiped out appropriately when the primary coolant containing boric acid had leaked from the seal cover around the upper part of the stub at the time of commissioning after construction, which has allowed the boric acid to remain.	This event was attributed to PWSCC in the pressurizer piping nozzles. Impacts on the US-APWR design have included adoption of 690 series nickel based alloy, for its excellence in corrosion resistance, for welding the nozzles to the reactor vessel closure head, as well as continued emphasis on rigid controls on welding and primary coolant water quality.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 19 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
June 10, 2004	Ohi Power Station (No.1)	During the periodical inspection, when water-filling into the reactor cavity from the refueling water storage tank was performed for preparation of fuels removal, transformation of the upper part of the tank was recognized on June 10, 2004. The cause of the tank transformation was estimated as follows: While installation of the duct hose to the 6 inch-diameter air-vent-pipe should be performed only during water-filling, the duct hose was installed before the tank water drainage and filling water into the reactor cavity was performed. As a result of it, the loosened duct hose inclined and some portions of the duct hose were almost blocked, which resulted in smaller pressure of the tank inside than that of the outside.	This event was attributed to faulty operation, and has had no impact on the US-APWR design.
July 5, 2004	Ohi Power Station (Unit 1)	During the periodical inspection, thickness measurements of the feedwater pipes from the main feedwater isolation valves to the steam generators was performed by ultrasonic test method and partial denting was recognized at the downstream elbow portions of the main feedwater isolation valves of three (A, B, C) of four. The cause of the thinning generation was estimated that largely turbulent water flow, caused by the main feedwater isolation valve (glove valve)'s structure, became further turbulent in the pipe elbow portion, and resulted in generation of thinning, followed by slow development of the thinning.	This event was attributed to faulty maintenance, and has had no impact on the US-APWR design.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 20 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
July 16, 2004	Ohi Power Station (Unit 1)	During the periodical inspection, repair work of the transformed refueling water storage tank was performed and inspection for preparing pressure-resisting test of the tank has been performed. On July 14, 2004, a small amount of water ooze from the vicinity of the support backplate weld of the concerned tank return pipe was found. Afterwards, detailed inspection was performed and penetrant indication patterns on the vicinity of the four backplate welds except the location of ooze concerned were newly recognized. All were recognized as chlorine-typed stress corrosion cracking. It was estimated that the chlorine-typed stress corrosion crack generated, because 1) A treelike intergranular crack was recognized in five locations in total, including the location of ooze recognized and other four backplate locations, 2) The tank is made of stainless steel and tensile residual stress exists in the backplate weld portion concerned, 3) The tank had been installed in the outside without coating for the long period and sea salt particles were easy to stay under the environment. The cause of water ooze was estimated as follows: The chlorine-typed stress corrosion crack developed, and resulted in a wall-through. Furthermore, in addition to aging progress of the coating due to liquid contact, heat influence was added, when the tank return pipe support, which was installed to the backplate concerned, was cut and welded to repair the tank. This led to separation of the coating film of the concerned portion, and resulted in water ooze with the tank water filling.	This event was attributed to faulty maintenance, and has had no impact on the US-APWR design. Also, this event is not directly applicable to the US-APWR, which has no RWST.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 21 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
August 9, 2004	Mihama Power Station (Unit 3)	During the rated thermal power operation, the fire alarm was activated and the mismatch trip signal alarm under the condition of feed-water flow rate smaller than steam flow rate to the 3A steam generator was activated, followed by the reactor automatic shutdown. Eleven workers at the facility were burned (5 workers were killed) by flashed steam from the broken mouth of the condensate water piping. As the result of field investigation, it was confirmed that a broken mouth was located at the condensate water piping near the ceiling of the second floor of the TB (the condensate water piping between the fourth low-pressure feedwater heater and the deaerator) and secondary condensate water (steam) flashed. The causes of this event were :1) Thinning of concerned piping had not been detected for a long time due to the inadequate management by Kansai Electric Power Co. Inc., Mitsubishi Heavy Industries, Ltd. and Nihon Arm Co. Ltd., 2) Inadequate management and quality assurance activities by each company mentioned above.	This event was attributed to faulty maintenance, and has had no impact on the US-APWR design.
September 6, 2004	Takahama Power Station (Unit 4)	During the periodical inspection, the Eddy Current Test with multi-coil type probe (Intelligent ECT) was performed on all tubes of steam generators to confirm their integrity. Significant signal indications of denting were recognized on the outer surface of 339 tubes. The signal indications detected by the Intelligent ECT were found in lines at the positions where anti-vibration-holding bars and then they were due to the thinning by wearing.	This event was attributed to faulty maintenance. MHI points out that conventional methods were unable to detect the damage, however, with the improved detection accuracy of the Intelligent ECT, together with enhanced precision in depth assessment, successful detection was achieved. Impact on the US-APWR design has been the inclusion of an increased number of anti-vibration bars in the steam generators to reduce tube vibration.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 22 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
September 10, 2004	Sendai Power Station (Unit 1)	During the periodical inspection, the Eddy Current Test with multi-coil type probe (Intelligent ECT) was performed on all tubes of steam generators to confirm their integrity. Significant signal indications of denting were recognized on the outer surface of 292 tubes. It was estimated that the significant signal indications detected (at 5 tubes) in the tube expansion zone would represent the occurrence of PWSCC on the inner surface of tubes. Combination of local residual stresses by the expansion of tubes at the time of fabrication of the steam generators and the stresses by internal pressure during operation would have finally caused it. It was also estimated that the significant signal indications (at 287 tubes) in the U-type tube zones would represent the thinning by wearing of tubes in the past at the positions of old anti-vibration-holding bars.	This event was attributed to faulty maintenance. MHI points out that conventional methods were unable to detect the damage, however, with the improved detection accuracy of the Intelligent ECT, together with enhanced precision in depth assessment, successful detection was achieved. Impact on the US-APWR design has been the inclusion of an increased number of anti-vibration bars in the steam generators to reduce tube vibration. This event was attributed to PWSCC in the steam generator tubes, which were fabricated from Alloy 600 TT. Impacts on the US-APWR design have included adoption of Alloy 690 TT for its excellence in corrosion resistance; reduction in the difference between tube plate diameter and outer diameter of the heat transfer tubes, and reduction in residual stress by changing from the conventional roll tube expansion method.
September 21, 2004	Tomari Power Station (Unit 2)	During the periodical inspection, the Eddy Current Test with multi-coil type probe (Intelligent ECT) was performed on all tubes of steam generators to confirm their integrity. Significant signal indications of denting were recognized on the outer surface of 56 tubes. The signal indications detected by the Intelligent ECT were found in lines at the positions where anti-vibration-holding bars and then they were due to the thinning by wearing.	This event was attributed to faulty maintenance. MHI points out that conventional methods were unable to detect the damage, however, with .the improved detection accuracy of the Intelligent ECT, together with enhanced precision in depth assessment, successful detection was achieved. Impact on the US-APWR design has been the inclusion of an increased number of anti-vibration bars in the steam generators to reduce tube vibration.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 23 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
October 25, 2004	Mihama Power Station (Unit 1)	During the periodical inspection conducted since September 5, 2004, wall thickness of turbine-driven auxiliary feedwater pipes was measured with using ultrasonic waves. Their wall thicknesses were revealed to be 5.7 mm and 5.6 mm respectively, at the portion upstream to the flow regulating valve for the steam generator A and close to its weld zone with the pipe, and at another portion downstream from the flow regulating valve for the steam generator B and close to its weld zone with the pipe. It was confirmed that the wall thickness was thinner than the level to require their report to the agency, namely less than 5.8 mm. It was estimated that disagreement between the center of the edge preparation device and those of the relevant pipes, when the inner surface of the pipes was grinded at the time of construction of the power station, would have caused the eccentricity of the pipes to lead to the reduced wall thickness at the certain portions to the level which requires report to the agency.	This event was attributed to faulty workmanship, and has had no impact on the US-APWR design.
December 15, 2004	Sendai Power Station (Unit 2)	During the periodical inspection, the Eddy Current Test with multi-coil type probe (Intelligent ECT) was performed on all tubes of steam generators to confirm their integrity. Significant signal indications of denting were recognized on the outer surface of 426 tubes. The signal indications detected by the Intelligent ECT were found in lines at the positions where anti-vibration-holding bars and then they were due to the thinning by wearing.	This event was attributed to faulty maintenance. MHI points out that conventional methods were unable to detect the damage, however, with .the improved detection accuracy of the Intelligent ECT, together with enhanced precision in depth assessment, successful detection was achieved. Impact on the US-APWR design has been the inclusion of an increased number of anti-vibration bars in the steam generators to reduce tube vibration.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 24 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
December 24, 2004	Ikata Power Station (Unit 1)	During the periodical inspection, on December 23, through-wall cracks were found at the horizontal duct part of the A/B exhaust stack during the inspection of the vicinity of location where a new inspection opening of the concerned exhaust stack was to be installed. Detailed inspection of the exhaust stack was conducted on December 24, and identified 12 cracks at the inner surface including 4 cracks of through-wall. 19 cracks were recognized in the vicinity of the intermittent weld portion of the A/B exhaust stack, and a crack was recognized in the seal weld portion of the steel plate for connection of the A/B exhaust stack and the containment exhaust stack, respectively. The cause of cracks in the vicinity of the intermittent weld portion was estimated as follows: The interval of the steel plates for reinforcement was long in the horizontal duct part and was short in the vertical duct part. Both parts were located in the slipstream of the bent portions. Therefore, the duct vibrated due to the pressure fluctuation, the fluctuated stress exceeded the fatigue limit in the intermittent weld portion, which has a shape on which stress concentrates, and resulted in the crack generation and development from the outer surface of stainless steel plate in the vicinity of the weld portion. The cause of cracks in the seal weld portion was estimated as follows: Rain water entered into the gap among the intermittent weld portions of the steel plates for connection of the duct upper surface, generated corrosion, and resulted in wall-through of the inner and outer surfaces of the seal weld portion.	This event was attributed to faulty seal weld design of the ventilation stacks where they are connected to their support structures, and weld corrosion and cracking from exposure to rain. The design impacts of this event on the US-APWR are that the fluctuation of pressure, corrosion and vibration are re-considered for improved design of the US-APWR exhaust stacks.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 25 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
January 18, 2005	Tsuruga Power Station (Unit 2)	During the periodical inspection, the Eddy Current Test with multi-coil type probe (Intelligent ECT) was performed on all tubes of steam generators to confirm their integrity. Significant signal indications of denting were recognized on the outer surface of 475 tubes. The signal indications detected by the Intelligent ECT were found in lines at the positions where anti-vibration-holding bars and then they were due to the thinning by wearing.	This event was attributed to faulty maintenance. MHI points out that conventional methods were unable to detect the damage, however, with .the improved detection accuracy of the Intelligent ECT, together with enhanced precision in depth assessment, successful detection was achieved. Impact on the US-APWR design has been the inclusion of an increased number of anti-vibration bars in the steam generators to reduce tube vibration.
March 19,2005	Mihama Power Station (Unit 1)	During the rated thermal power operation, it was found that three bolts of manifold cover of B-charging pump in the charging pump room of the first basement of the reactor A/B were broken. And it was also found a bolt of another manifold cover of the concerned charging pump was broken.	This event was attributed to faulty maintenance, and has had no impact on the US-APWR design.
April 28, 2005	Mihama Power Station (No.1)	As an operational feedback of the event of cracks occurred at the reactor A/B exhaust stack of the Ikata Power Station Unit-1 (on December 24, 2004), visual inspection of the A/B exhaust stack was performed during the periodical inspection. It was recognized that two drain pipings, which are installed to the bottom of the A/B exhaust stack in the annulus portion (outside of the building), were taken off. Cracks were also recognized at the part of the bottom of the stack concerned. As the result of the investigation, it was estimated that fatigue cracks generated because the cyclic stress beyond the fatigue limit applied on the thin welding portion of the exhaust stack part and the drain piping by the vibration of bottom plate due to the exhaust gas passing through the exhaust stack. In the development process of crack on the circumference of the weld metal, cyclic stress also applied on the concerned exhaust stack bottom plate. As a result, cracks generated on the bottom plate and the concerned drain piping was finally separated from the bottom plate of the exhaust stack due to ductile fracture.	This event was attributed to faulty fabrication, and has had no impact on the US-APWR design.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 26 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
May 12, 2005	Ikata Power Station (No.3)	The reactor had been operated at the rated thermal power. During the periodical inspection of the Chiller-D of the main control room HVAC system, very slight abnormal noise was recognized when the test operation of the concerned chiller was conducted on April 27, 2005, and the chiller was stopped. When the concerned chiller was overhauled on May 12, 2005, several flaws were found at the parts of impellers and seal ring of the chiller. As the result of the investigation, the cause of the event was estimated as follows: The suction portion of the impeller was apt to contact to the seal ring, because the center (main shaft) of the suction portion of the impeller and the center of the impeller cover shifted slightly, and then they were assembled after overhauling of the concerned chiller. The vane openness to adjust flow rate of the refrigerant gas became fully closed when the automatic stop test was performed, and vibration of the impeller became larger than one at the normal operation. Therefore, in subsequent load test operation, the suction portion of the impeller contacted with the seal ring, they became high in temperature, strength of them reduced, and then the parts of them fractured due to friction force.	This event was attributed to faulty maintenance, and has had no impact on the US-APWR design.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 27 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
September 29, 2005	Mihama Power Station (No.1)	As countermeasures of leakage from the welding portion of thermometer stub of the heating steam drain piping of the moisture separator and heater train-B, which occurred during the adjustment operation of the 21st periodical inspection, in order to repair the concerned leakage portion and the same portion of the train-A, the power-decrease-operation was initiated at 17:15 on September 29, 2005. During the power-decrease-operation, the alarm of standpipe water level-low at primary coolant pump-A was announced at 19:08. Therefore, the makeup water to the standpipe was supplied. However, the same alarm was announced again at 19:43 and it was confirmed by TV-monitor in the containment vessel that seal water leaked from the splashguard of the primary coolant pump-A. Therefore, the reactor was manually shut down. As the result of the investigation, it was recognized that spring force of the seal ring spring of the No.3 shaft seal portion of the pump had reduced gradually during the operation period. At generation of the event, shooting force in the seal portion had become smaller than the friction force (which worked to the opposite direction of the seating force). The cause of the event was estimated as follows : At the condition, when the seal runner stretched at the shaft direction due to heat caused by temperature change in the containment, the seal ring could not follow up the seal runner and openness between the seat surfaces generated. Therefore, the amount of leakage of seal water increased. The cause of the event was also estimated as follows: Seal water was made up into the shaft seal portion through the stand pipe, followed by leakage of seal water. However, seal water leaked from the splashguard because the leakage amount of seal water became larger than the recovery capability of the seal water recovery line.	This event was attributed to expected aging and deterioration due to operational use and to faulty maintenance, and has had no impact on the US-APWR design.

Table 1.9.4-3 Reportable Events at Operating Japanese PWRs for the Period 1996-2006 (sheet 28 of 28)

Date of Occurrence	Power Station	Description	MHI Comment
January 6, 2006	Tomari Power Station (No.1)	During the periodical inspection, when in-house part of the emergency exhaust stack was inspected, six cracks (through-wall cracks) were identified near the welding portion of reinforcement metal on the exhaust stack. As the result of the investigation, it was recognized that the emergency exhaust stack was under the influence of vibrations of the main exhaust stack through the common support, and it was vibrating all the time. When the emergency exhaust stack is in operation, fluctuations of pressure within this exhaust stack will result in additional vibration. Especially in the downstream to its elbow portion, pressure fluctuates significantly within the stack to produce more vibration. Under those conditions that both the main exhaust stack and the emergency exhaust stack are in operation, the emergency exhaust stack will be exposed to repeated stress of larger than its fatigue limit near the welding portions in the downstream to its elbow portion. It was presumed that this stress generated and aggravated cracks at the outer surface of its stainless steel plate near the welding portions, with some cracks running through its wall completely.	This event was attributed to faulty weld design of the ventilation stack where it is connected to its lower support structure. The design impacts of this event on the US-APWR are that the effects of vibration are re-considered for the improved design of the US-APWR exhaust stacks.
January 13, 2006	Sendai Nuclear Power Station (No.1)	During the periodical inspection, the Eddy Current Test was conducted on all tubes of three steam generators. Significant signal indications of defects were recognized on 13 tubes. The locations of the defects were its tube expansion zone of the hot leg side. The cause of the concerned defects found on the tubes was estimated as follows: The combined physical conditions by the residual stress, which was locally caused during the tube expansion process during the steam generator manufacturing, and by the stress caused due to high internal pressure during the power operation, caused the stress corrosion cracking on the inner surface of the tubes.	This event was attributed to PWSCC in the steam generator tubes, which were fabricated from Alloy 600 TT. Impacts on the US-APWR design have included adoption of Alloy 690 TT for its excellence in corrosion resistance; reduction in the difference between tube plate diameter and outer diameter of the heat transfer tubes, and reduction in residual stress by changing from the conventional roll tube expansion method.

1.9.5 Advanced and Evolutionary Light-Water Reactor Design Issues

Language from Reg Guide 1.206 section C.I.1.9.5 states that applicants should address the licensing and policy issues applicable to the proposed facility design that have been developed by the NRC and documented in the Office of the Secretary of the Commission (SECY) documents, and associated staff requirements memoranda, for advanced and evolutionary light-water reactor designs. The list of SECY documents provided in the guidance is as follows:

- SECY-89-013, “Design Requirements Related to the Evolutionary Advanced Light-Water Reactors (ALWRs)”
- SECY-90-016, “Evolutionary Light-Water Reactor (ELWR) Certification Issues and Their Relationship to Current Regulatory Requirements”
- SECY-90-241, “Level of Detail Required for Design Certification under Part 52”
- SECY-90-377, “Requirements for Design Certification under 10CFRPart 52”
- SECY-91-074, “Prototype Decisions for Advanced Reactor Designs”
- SECY-91-178, “ITAAC for Design Certifications and Combined Licenses”
- SECY-91-210, “ITAAC Requirements for Design Review and Issuance of FDA”
- SECY-91-229, “Severe Accident Mitigation Design Alternatives for Certified Standard Designs”
- SECY-91-262, “Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light-Water Reactor (LWR) Designs”
- SECY-92-053, “Use of Design Acceptance Criteria During the 10CFRPart 52 Design Certification Reviews”
- SECY-92-092, “The Containment Performance Goal, External Events Sequences, and the Definition of Containment Failure for Advanced LWRs”
- SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs”
- SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Design (RTNSS)”
- SECY-94-302, “Source-Term-Related Technical and Licensing Issues Relating to Evolutionary and Passive Light-Water-Reactor Designs”
- SECY-95-132, “Policy and Technical Issues Associated with Regulatory Treatment of Non-Safety Systems in Passive Plant Designs”

In SRP 1.0 section I.9, “Conformance with Regulatory Criteria”, the subsection entitled “Advanced and Evolutionary Light-Water Reactor Design Issues” suggests “A table that identifies the information addressing the applicable licensing and policy issues developed by the NRC and documented in SECY-93-087 and the associated SRM for advanced and evolutionary light-water reactor designs is reviewed.”

1.9.5.1 Summary of SECY Letters

For completeness, all of the SECY letters listed in Reg Guide 1.206 section C.I.1.9.5 are presented in Table 1.9.5-1, along with a general summary of each document in the column entitled “Comment”. Those SECY letters that require additional detailed treatment of requirements are so indicated in Table 1.9.5-1, with direction as appropriate to tables 1.9.5-2, 1.9.5-3 and 1.9.5-4.

Table 1.9.5-1 General Summary of SECY Letters Cited in RG 1.206 Section C.I.1.9.5 (sheet 1 of 2)

Document Number	Title	Comment
SECY-89-013	Design Requirements Related to the Evolutionary Advanced Light-Water Reactors (ALWRs)	The topics (design issues) in this letter are carried forward and repeated in SECY-90-016 and SECY-93-087. See the table entry for SECY-93-087 below.
SECY-90-016	Evolutionary Light-Water Reactor (ELWR) Certification Issues and Their Relationship to Current Regulatory Requirements	The topics (design issues) in this letter are carried forward and repeated in SECY-93-087. See the table entry for SECY-93-087 below.
SECY-90-241	Level of Detail Required for Design Certification under Part 52	The appropriate regulatory recommendations in this letter have been carried forward into 10CFR52 and implementing NRC guidance documents. Conformance is addressed in sections 1.9.1 and 1.9.2 of this DCD.
SECY-90-377	Requirements for Design Certification under 10CFRPart 52	The appropriate regulatory recommendations in this letter have been carried forward into 10CFR52 and implementing NRC guidance documents. Conformance is addressed in sections 1.9.1 and 1.9.2 of this DCD.
SECY-91-074	Prototype Decisions for Advanced Reactor Designs	MHI position is that US-APWR is evolutionary and does not represent a “significant deviation” from standard and known technologies, and thus does not require prototyping. Data for testing of components such as advanced accumulators, fuel, etc. is described in topical reports referenced in section 1.5 of this DCD.
SECY-91-178	ITAAC for Design Certifications and Combined Licenses	The appropriate regulatory recommendations in this letter have been carried forward into 10CFR52 and implementing NRC guidance documents. Conformance is addressed in sections 1.9.1 and 1.9.2 of this DCD.
SECY-91-210	ITAAC Requirements for Design Review and Issuance of FDA	The appropriate regulatory recommendations in this letter have been carried forward into 10CFR52 and implementing NRC guidance documents. Conformance is addressed in sections 1.9.1 and 1.9.2 of this DCD.
SECY-91-229	Severe Accident Mitigation Design Alternatives for Certified Standard Designs	Severe accidents are addressed in section 19 of this DCD, and Severe Accident Mitigation Design Alternatives (SAMDAs) are addressed in section 19.2.6
SECY-91-262	Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light-Water Reactor (LWR) Designs	This SECY document does not present any new requirements that are applicable to the US-APWR.

Table 1.9.5-1 General Summary of SECY Letters Cited in RG 1.206 Section C.I.1.9.5 (sheet 2 of 2)

Document Number	Title	Comment
SECY-92-053	Use of Design Acceptance Criteria During the 10CFRPart 52 Design Certification Reviews	The appropriate regulatory recommendations in this letter (ITAAC, DAC and the two-tiered approach) have been carried forward into 10CFR52 and implementing NRC guidance documents. Conformance is addressed in sections 1.9.1 and 1.9.2 of this DCD.
SECY-92-092	The Containment Performance Goal, External Events Sequences, and the Definition of Containment Failure for Advanced LWRs	This SECY document provides a status update on several issues and does not present any new requirements.
SECY-93-087	Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs	See table 1.9.5-2 for treatment of the requirements of this document.
SECY-94-084	Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Design (RTNSS)	SECY-94-084 and SECY-95-132 were consolidated into one document by memo to file from D. M. Crutchfield dated 7/24/95. See table 1.9.5-3 for treatment of the requirements of these documents.
SECY-94-302	Source-Term-Related Technical and Licensing Issues Relating to Evolutionary and Passive Light-Water-Reactor Designs	See table 1.9.5-4 for treatment of the requirements of this document.
SECY-95-132	Policy and Technical Issues Associated with Regulatory Treatment of Non-Safety Systems in Passive Plant Designs	SECY-94-084 and SECY-95-132 were consolidated into one document by memo to file from D. M. Crutchfield dated 7/24/95. See table 1.9.5-3 for treatment of the requirements of these documents.

SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs"

A total of 42 issues were presented in SECY-93-087, and they were broken down into 3 categories:

- I. SECY-90-016 Issues (A through N)
- II. Other Evolutionary and Passive Design Issues (A through T)
- III. Issues Limited to Passive Designs (A through H)

Table 1.9.5-2, "Detailed Treatment of Requirements of SECY-93-087", is presented below and shows the 42 issues grouped as they are in SECY-93-087 and reiterated above. Twenty-one (21) of the issues were approved by the Commissioners in the related Staff Requirements Memorandum (SRM) dated July 21, 1993, and these are so noted in the table. Each table entry contains the number and title of the issue, requirements that are applicable to the US-APWR, and a comment on where the issue is treated in this DCD.

Table 1.9.5-2 Detailed Treatment of Requirements of SECY-93-087 (sheet 1 of 15)

Issue Number	Description	Requirements	Comment
I.A	Use of a Physically Based Source Term	This issue is addressed in SECY-94-302.	See table 1.9.5-4
I.B	Anticipated Transient Without Scram	SECY-93-087 states that the NRC staff considers this issue to be resolved.	No new requirements
I.C	Mid-Loop Operation	SECY-93-087 states that the NRC staff considers this issue to be resolved.	No new requirements
I.D	Station Blackout	This issue is addressed in the consolidated SECY-94-084 and SECY-95-132.	See table 1.9.5-3.
I.E	Fire Protection	Approved by Commission in SRM dated 7/21/93. “The Commission approves the staff's position that the passive plants should also be reviewed against the enhanced fire protection criteria approved in the Commission's SRM of June 26, 1990.” These are in turn drawn from SECY-90-016, which also mentions the requirements of 10CFR50.48 and Appendix R, and are cited as follows:” Therefore the evolutionary ALWR designers must ensure that safe shutdown can be achieved, assuming that all equipment in any one fire area will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible. Because of its physical configuration, the control room is excluded from this approach. Provided an independent alternative shutdown capability that is physically and electrically independent of the control room is included in the design. Evolutionary ALWRs must provide fire protection for redundant shutdown systems in the reactor containment building that will ensure, to the extent practicable, that one shutdown division will be free of fire damage. Additionally the evolutionary ALWR designers must ensure that smoke, hot gases, or the fire suppressant will not migrate into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions. Because the layout of a nuclear plant is design specific, plant-specific design details will be reviewed by the staff on an individual basis. The staff will require a description of safety-grade provisions for the fire-protection systems to ensure that the remaining shutdown capabilities are protected, as well as demonstration that, the design complies with the migration criteria discussed above.”	Addressed for US-APWR in DCD sections 3.1.1, 9.5.1 and Appendix 9A

Table 1.9.5-2 Detailed Treatment of Requirements of SECY-93-087 (sheet 2 of 15)

Issue Number	Description	Requirements	Comment
I.F	Intersystem Loss-of-Coolant Accident	Approved by Commission in SRM dated 7/21/93. “The Commission approves the staff's position that the passive plants should also be reviewed for compliance with the intersystem LOCA criteria approved in the Commission's SRM of June 26, 1990”, which says: “The Commission (with all Commissioners agreeing) has approved the staff's position on intersystem LOCA provided that, as recommended by the ACRS, all elements of the low pressure system are considered (e.g. instrument lines, pump seals, heat exchanger tubes, and valve bonnets.) The original text drawn from SECY-90-016 states: “The staff concludes that designing, to the extent practicable, low-pressure systems to withstand full RCS pressure is an acceptable means for resolving this issue. However, the staff believes that for those systems that have not been designed to withstand full RCS pressure, evolutionary ALWRs should provide (1) the capability for leak testing of the pressure isolation valves, 2) valve position indication that is available in the control room when isolation valve operators are de-energized and (3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed. Imposition of these requirements exceed Commission regulations and guidance; therefore, the staff recommends that the Commission approve these positions for evolutionary ALWRs.”	SECY-93-087 Issue No. I.F requires that: [1] The low pressure systems should be designed to withstand full RCS pressure, or [2] For those systems that have not been designed to withstand full RCS pressure, evolutionary ALWRs should provide (1) the capability for leak testing of the pressure isolation valves, (2) valve position indication that is available in the control room when isolation valve operators are de-energized and (3) high-pressure alarms to warn control room operators and both isolation valves are not closed. The US-APWR design provisions for these requirements are described in DCD Section 3.12.5. Section 9.3.1 describes Interface LOCA which is addressed in risk assessment; therefore, the provisions for above-mentioned requirements are not described.

Table 1.9.5-2 Detailed Treatment of Requirements of SECY-93-087 (sheet 3 of 15)

Issue Number	Description	Requirements	Comment
I.G	Hydrogen Control	Approved by Commission in SRM dated 7/21/93. “The Commission approves the staff's position that the passive plants should be designed, as a minimum, to the same requirements applied to evolutionary designs. Specifically, passive plants must: Accommodate hydrogen generation equivalent to a 100% metal-water reaction of the fuel cladding; Limit containment hydrogen concentration to no greater than 10%; and Provide containment-wide hydrogen control (such as igniters or inerting) for severe accidents. The Commission approves the staff's clarification, as expressed at the Commission briefing, that the possible use of passive autocatalytic hydrogen recombiners should not be precluded from consideration a priori. The staff is cautioned to consider carefully the relatively slow time response of autocatalytic recombiners as a possible impediment to their efficiency.”	Addressed for US-APWR in DCD sections 6.2.5 and 19.2.3.3.2 Non-safety related igniters are located in containment adequately.

Table 1.9.5-2 Detailed Treatment of Requirements of SECY-93-087 (sheet 4 of 15)

Issue Number	Description	Requirements	Comment
I.H	Core Debris Coolability	Approved by Commission in SRM dated 7/21/93. “The Commission approves the staff's position that both the evolutionary and passive ALWR designs meet the following criteria: Provide reactor cavity floor space to enhance debris spreading. Provide a means to flood the reactor cavity to assist in the cooling process. Protect the containment liner and other structural members with concrete, if necessary. Ensure that the best estimate environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed Service Level C for steel containments or Reactor Load Category for concrete containments, for approximately 24 hours. Ensure that the containment capability has margin to accommodate uncertainties in the environmental conditions from core-concrete interactions. With regard to the 0.02m2/MWt reactor vessel cavity floor area, the staff should continue its research activities and supporting analyses, as documented in its May 19, 1993 letter to the ACRS. With respect to the containment response to ex-reactor vessel core debris, the staff should not limit licensees to only one method for addressing containment responses to severe accident events but also permit other technically justified means for demonstrating adequate containment response.”	Addressed for US-APWR in DCD sections 19.2.3.3.3 Generic Letter No. 88-20 states the debris coolable criterion as less than approximately 25cm. The US-APWR reactor cavity floor provides sufficient area to meet this coolable criterion. The US-APWR provides dependable reactor cavity flooring system, which consists of two independent water supply systems, one is CSS with a drain line and the other is firewater injection to the reactor cavity. Reactor cavity floor concrete is provided to protect from direct impact to the steel liner plate by relocated core debris. PCCV has sufficient resistance to pressure and temperature rise resulting from core-concrete interactions which is analyzed using sever accident analysis program (MAAP). Core coolability is confirmed using severe accident analysis program (MAAP).

Table 1.9.5-2 Detailed Treatment of Requirements of SECY-93-087 (sheet 5 of 15)

Issue Number	Description	Requirements	Comment
I.I	High-Pressure Core Melt Ejection	Approved by Commission in SRM dated 7/21/93. “The Commission approves the staff's position for the general criteria that the evolutionary and passive LWR designs: provide a reliable depressurization system; and provide cavity design features to decrease the amount of ejected core debris that reaches the upper containment.”	Addressed for US-APWR in DCD section 19.2.3.3.4 Severe accident dedicated primary system depressurization system is installed. Debris capture structure is provided at the corner of reactor cavity tunnel and reactor cavity floor.
I.J	Containment Performance	Approved by Commission in SRM dated 7/21/93. “The recommendations on containment performance, as outlined in SECY 93-087, could be read to imply that the staff is no longer proposing to use the concept of conditional containment failure probabilities (CCFP). However, based on discussions held during the Commission meeting on this subject, the staff informed the Commission that it intends to continue to apply the 0.1 CCFP in implementing the Commission's defense in depth regulatory philosophy and the Commission's policy on Safety Goals. Therefore, the Commission approves the staff's position to use the following deterministic containment performance goal in the evaluation of the passive ALWRs as a complement to the CCFP approach approved by the Commission in its SRM of June 26, 1990: "The containment should maintain its role as a reliable, leak-tight barrier (for example, by ensuring that containments stresses do not exceed ASME Service Level C limits for metal containments, or Factored Load Category for concrete containments) for approximately 24 hours following the onset of core damage under the more likely severe accident challenges and, following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products. The Commission approves the staff's interim approach subject to the staff's review and recommendations resulting from public comments on the Advance Notice of Proposed Rulemaking on Severe Accident Plant Performance Criteria for Future ALWRs."	Addressed for US-APWR in DCD sections 3.8.1, 3.8.2, 6.2.1, and 19.2.4 PCCV has high resistance to pressure rise, and PCCV withstands approximately 24 hours following onset of core damage.

Table 1.9.5-2 Detailed Treatment of Requirements of SECY-93-087 (sheet 6 of 15)

Issue Number	Description	Requirements	Comment
I.K	Dedicated Containment Vent Penetration	Approved by Commission in SRM dated 7/21/93. “The Commission approves the staff's position that the need for a containment vent for the passive plant designs should be evaluated on a design-specific basis.”	Addressed for US-APWR in DCD section 19.2.3.3.9 Dedicated containment vent penetration is provided.
I.L	Equipment Survivability	Approved by Commission in SRM dated 7/21/93. “The Commission approves the staff's position that the passive plant design features provided only for severe accident mitigation need not be subject to the environmental qualification requirements of 10CFRSection 50.49; quality assurance requirements of 10CFRPart 50, Appendix B; and redundancy/diversity requirements of 10CFRPart 50, Appendix A.”	Addressed for US-APWR in DCD section 19.2.3.3.7 Severe accident dedicated systems are not subject to the environmental qualification requirements of 10 CFR Section 50.49; quality assurance requirements of 10 CFR Part 50, Appendix A. But those necessary in severe accident countermeasures are confirmed to withstand those environmental conditions practically.

Table 1.9.5-2 Detailed Treatment of Requirements of SECY-93-087 (sheet 7 of 15)

Issue Number	Description	Requirements	Comment
I.M	Elimination of Operating-Basis Earthquake	Approved by Commission in SRM dated 7/21/93. “The Commission approves the staff's recommendation to account for earthquake cycles in the fatigue analyses of piping systems performed until the new guidance is issued, using two SSE events with 10 maximum stress cycles per event (20 full cycles of the maximum SSE stress range). Alternatively, the number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE Standard 344-1987. The Commission approves the staff's recommendation that the effects of anchor displacements in the piping caused by an SSE be considered with the Service Level D limit. The Commission approves the staff's recommendation to eliminate the OBE from the design of systems, structures, and components. When the OBE is eliminated from the design, no replacement earthquake loading should be used to establish the postulated pipe rupture and leakage crack locations. The Commission approves the staff's recommendation that the mechanistic pipe break and high-energy leakage crack locations determined by the piping high stress (without the OBE) and fatigue locations may be used for equipment environmental qualification and compartment pressurization purposes.	Addressed for US-APWR in DCD sections 3.2.1, 3.7, 3.10.2 and 3.12.5

Table 1.9.5-2 Detailed Treatment of Requirements of SECY-93-087 (sheet 8 of 15)

Issue Number	Description	Requirements	Comment
I.M	Elimination of Operating-Basis Earthquake (continued)	The Commission agrees that with the elimination of the OBE, two alternatives exist that will essentially maintain the requirements provided in IEEE Standard 344-1987 to qualify equipment with the equivalent of five OBE events followed by one SSE event (with 10 maximum stress cycles per event). Of these alternatives, the equipment should be qualified with five one-half SSE events followed by one full SSE event. Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half SSE events may be used in accordance with Appendix D of IEEE Standard 3441987 when followed by one full SSE. The Commission agrees that the above requirements should also apply to passive ALWRs. The Commission understands that the OBE will continue to be used as a threshold criterion for conducting inspections following an earthquake event. The staff should keep the Commission and the ACRS informed as the staff's further analysis and review proceed.”	
I.N	Inservice Testing of Pumps and Valves	Approved by Commission in SRM dated 7/21/93. “The Commission has no objection to the staff's position, but understands that further elaboration on this issue will be forthcoming from the staff.” The language to which the Commission was referring from SECY-93-087 was: “In SECY-90-016, the staff recommended that the Commission approve the position that the following provisions should be applied to all safety-related pumps and valves, and not limited to ASME Code Class 1, 2, and 3 components: Piping design should incorporate provisions for full flow testing (maximum design flow) of pumps and check valves. Designs should incorporate provisions to test motor-operated valves under design-basis differential pressure. Check valve testing should incorporate the use of advanced, non-intrusive techniques to address degradation and performance characteristics. A program should be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation that cannot be detected through the use of advanced, non-intrusive techniques. The staff concluded that these requirements are necessary to provide an adequate assurance of operability.” This issue is also addressed in the consolidated SECY-94-084 and SECY-95-132.	Addressed for US-APWR in DCD sections 3.1.4, 3.9.6, 6.6, 13.4 and Chapter 16 (Tech Spec section 5.5) See table 1.9.5-3 for requirements from SECY-94-084 and SECY-95-132.

Table 1.9.5-2 Detailed Treatment of Requirements of SECY-93-087 (sheet 9 of 15)

Issue Number	Description	Requirements	Comment
II.A	Industry Codes and Standards	Approved by Commission in SRM dated 7/21/93. “The Commission approves the staff's position that consistent with past practice, that staff will review both evolutionary and passive plant design applications using the newest codes and standards that have been endorsed by the NRC. Unapproved revisions to codes and standards will be reviewed on a case-by-case basis.”	Addressed for US-APWR in DCD sections 3.2.4, 3.8.1, 3.8.3, 3.8.4, 3.8.5, 3.12.2, 3.12.6, and 7.1.2
II.B	Electrical Distribution	This issue is addressed in the consolidated SECY-94-084 and SECY-95-132.	See table 1.9.5-3.
II.C	Seismic Hazard Curves and Design Parameters	SECY-93-087 indicates that this issue was provided for completeness and for information only, with no policy questions.	No new requirements.
II.D	Leak-Before-Break	Approved by Commission in SRM dated 7/21/93. “The Commission approves the staff's recommendation that the leak before break approach should be applied to both the evolutionary and the passive ALWRs seeking design certification under 10CFRPart 52. This approval should be limited to instances in which appropriate bounding limits are established using preliminary analysis results during the design certification phase and verified during the COL phase by performing the appropriate ITAAC.”	Addressed for US-APWR in DCD section 3.6.3
II.E	Classification of Main Steamlines in Boiling Water Reactors	Approved by Commission in SRM dated 7/21/93. “The Commission approves the staff's position that neither the main steam drain and bypass lines from the first valve up to the condenser inlet, nor the piping between the turbine stop valve and the turbine inlet should be classified as safety-related or as seismic Category I. Rather, these lines should be analyzed using a dynamic seismic analysis to demonstrate structural integrity under SSE loading conditions. The turbine stop, control, and bypass valves and the main steam lines from the turbine control valves to the turbine shall meet all of the quality group and quality assurance guidelines specified in SRP Section 3.2.2, Appendix A. Further, that seismic analyses be performed to ensure that the condenser anchorages and the piping inlet nozzle to the condenser are capable of maintaining their structural integrity during and after the SSE. The Commission approves the above-described approach to resolve the main steamline classification for both evolutionary and passive ALWRs.”	BWR specific; not applicable to US-APWR

Table 1.9.5-2 Detailed Treatment of Requirements of SECY-93-087 (sheet 10 of 15)

Issue Number	Description	Requirements	Comment
II.F	Tornado Design Basis	Approved by Commission in SRM dated 7/21/93. “The Commission approves the staff's position that a maximum tornado wind speed of 482 km/hr (300 mph) be used in the design-basis tornado employed in the design of evolutionary and passive ALWRs.”	Not Applicable. A maximum tornado wind speed of 300 mph that is based on the RG 1.76 R0 is not used in the design-basis tornado. A maximum tornado wind speed of 230 mph that is based on the RG 1.76 R1 is used in the design-basis tornado.
II.G	Containment Bypass	Issue does not identify any new requirements that are applicable to the US-APWR design.	No new requirements.
II.H	Containment Leak Rate Testing	Approved by Commission in SRM dated 7/21/93. “The Commission approves the staff's position that until the rule change proceedings for Appendix J of 10CFRPart 50 are completed, the maximum interval between Type C leakage rate tests for both evolutionary and passive plant designs should be 30 months, rather than the 24 months maximum interval currently required in Appendix J to 10CFRPart 50.”	Addressed for US-APWR in DCD sections 3.1.4 and 3.1.5

Table 1.9.5-2 Detailed Treatment of Requirements of SECY-93-087 (sheet 11 of 15)

Issue Number	Description	Requirements	Comment
II.I	Post-Accident Sampling System	Approved by Commission in SRM dated 7/21/93. "The Commission approves the staff's position as modified below. The Commission approves the staff's recommendation that the post-accident sampling systems for evolutionary and passive ALWRs of the pressurized water reactor type be required to have the capability to have the capability to determine the gross amount of dissolved gases (not necessarily a pressurized sample) as an acceptable means of satisfying the intent of 10CFR50.34(f) (2)(viii) and Item II.B.3 of NUREG-0737. The Commission agrees that the time for taking these samples can be extended to 24 hours following the accident. The Commission agrees that for evolutionary and passive ALWRs of the boiling water reactor type, there would be no need for the post-accident sampling system to analyze dissolved gases. The Commission approves the deviation from the requirements of Item II.B.3. of NUREG-0737 with regard to requirements for sampling reactor coolant for boron concentration and radioactivity measurements using the post-accident sampling system in evolutionary and passive ALWRs. The modified requirement would require the capability to take boron concentration samples and radioactivity measurements 8 hours and 24 hours, respectively, following the accident. The Commission approval is based on the fact that the PASS system is an existing requirement and on the belief that a relatively simple system can be designed to meet the modified requirement. It is the Commission's understanding that a system can be designed which is simple, does not require chemical analysis of the gases in solution, and will provide the reactor operator information as to whether significant amounts of non-condensible gases exist in the reactor coolant."	The US-APWR post accident sampling system is designed in accordance with these requirements to have capability to analyze dissolved gases and chloride within 24 hours and to take boron concentration sample and activity measurements 8 hours and 24 hours, respectively, following the accident. The US-APWR design provisions for these requirements are described in DCD Section 9.3.2.
II.J	Level of Detail	SECY-93-087 indicates that this issue was provided for completeness and for information only, with no policy questions.	No new requirements.
II.K	Prototyping	SECY-93-087 indicates that this issue was provided for completeness and for information only, with no policy questions.	No new requirements.
II.L	ITAAC	SECY-93-087 indicates that this issue was provided for completeness and for information only, with no policy questions.	No new requirements.
II.M	Reliability Assurance Program	This issue is addressed in the consolidated SECY-94-084 and SECY-95-132.	See table 1.9.5-3.

Table 1.9.5-2 Detailed Treatment of Requirements of SECY-93-087 (sheet 12 of 15)

Issue Number	Description	Requirements	Comment
II.N	Site-Specific Probabilistic Risk Assessments and Analysis of External Events	Approved by Commission in SRM dated 7/21/93. "The Commission approves, in part, and disapproves, in part, the staff's position on site-specific probabilistic risk assessment and analysis of external events, as listed below. The Commission approves the position that the analyses submitted in accordance with 10CFR52.47 should include an assessment of internal and external events. The Commission disapproves the staff's recommendation to use two times the Design Basis SSE for margins-type assessment of seismic events. The Commission approves the use of 1.67 times the Design Basis SSE for a margin-type assessment of seismic events. The Commission approves the following staff recommendation, as modified: PRA insights will be used to support a margins-type assessment of seismic events. A PRA-based seismic margins analysis will consider sequence-level High Confidence, Low Probability of Failures (HCLPFs) and fragilities for all sequences leading to core damage or containment failures up to approximately one and two thirds the ground motion acceleration of the Design Basis SSE. The Commission approves the staff's position that the simplified probabilistic methods, such as but not limited to EPRI's FIVE methodology, will be used to evaluate fires. The Commission approves the staff's position that traditional probabilistic techniques should be used to evaluate internal floods. The Commission approves the staff's position that the ALWR vendors should perform bounding analyses of site-specific external events likely to be a challenge to the plant (such as river flooding, storm surge, tsunami, volcanism, high winds, and hurricanes). The Commission approves the staff's position that when a site is chosen, its characteristics should be compared to those assumed in the bounding analyses to ensure that the site is enveloped. The Commission approves the staff's position that if the site is enveloped, the COL Applicant need not perform further PRA evaluations for these external events. The COL Applicant should perform site-specific PRA evaluations to address any site-specific hazards for which a bounding analysis was not performed or which are not enveloped by the bounding analyses to ensure that no vulnerabilities due to siting exist."	Addressed for US-APWR in DCD Chapter 19.1 PRA covers internal external events. PRA covers seismic margin analysis. Fire PRA Methodology for Nuclear Power Facilities of NUREG/CR-6850 is used to perform Fire PRA for US-APWR. PRA covers internal flooding event. External events considered in design are treated in PRA according to their occurrence frequencies.

Table 1.9.5-2 Detailed Treatment of Requirements of SECY-93-087 (sheet 13 of 15)

Issue Number	Description	Requirements	Comment
II.O	Severe Accident Mitigation Design Alternatives	SECY-93-087 indicates that this issue was provided for completeness and for information only, with no policy questions.	No new requirements.
II.P	Generic Rulemaking Related to Design Certification	SECY-93-087 indicates that this issue was provided for completeness and for information only, with no policy questions.	No new requirements.
II.Q	Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems	Approved by Commission in SRM dated 7/21/93. "The Commission approves, in part, and disapproves, in part, the staff's recommendation. The Commission has approved a revised position, as follows: 1. The applicant shall assess the defense-in-depth and diversity of the proposed instrumentation and control system to demonstrate that vulnerabilities to common mode failures have adequately been addressed. 2. In performing the assessment, the vendor or applicant shall analyze each postulated common-mode failure for each event that is evaluated in the accident analysis section of the safety analysis report (SAR) using best estimate methods. The vendor or applicant shall demonstrate adequate diversity within the design for each of these events. 3. If a postulated common-mode failure could disable a safety function, then a diverse means, with a documented basis that the diverse means is unlikely to be subject to the same common-mode failure, shall be required to perform either the same function or a different function. The diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary function under the associated event conditions. 4. A set of displays and controls located in the main control room shall be provided for manual, system-level actuation of critical safety functions and monitoring of parameters that support the safety functions. The displays and controls shall be independent and diverse from the safety computer system identified in items 1 and 3 above.	Addressed for US-APWR in DCD sections 7.1.1, 7.1.3 and 7.8

Table 1.9.5-2 Detailed Treatment of Requirements of SECY-93-087 (sheet 14 of 15)

Issue Number	Description	Requirements	Comment
II.Q	Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems (continued)	The staff's position has been modified in essentially two respects: First, inasmuch as common mode failures are beyond design-basis events, the analysis of such events should be on a best-estimate basis. Second, the staff indicates in its discussion of the third part of its position that "The diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary function under the associated event conditions." Therefore, this clarification has been added to the fourth part of the staff's position (which refers to a subset of the safety functions referred to in the third part) by removing the safety grade requirement. Further, the remainder of the discussion under the fourth part of the staff position is highly prescriptive and detailed (e.g., "shall be evaluated," "shall be sufficient," shall be hardwired," etc.). The Commission approves only that such prescriptiveness be considered as general guidance, the practicality of which should be determined on a case-by-case basis.”	
II.R	Steam Generator Tube Ruptures	Approved by Commission in SRM dated 7/21/93. “Multiple Steam Generator Tube Ruptures The Commission approves the staff's position to require that analysis of multiple steam generator tube ruptures (STGRs) involving two to five steam generator tubes be included in the application for design certification for the passive PWRs. The Commission understands that, as discussed in the Commission meeting on this SECY paper, since the steam generator multi-tube rupture event is beyond the design basis requirements for PWRs, realistic or best-estimate analytical assumptions may be used to assess plant responses. Containment Bypass Potential Resulting From SGTRs The Commission approves the staff's recommendation that the applicant for design certification for a passive or evolutionary PWR assess design features to mitigate the amount of containment bypass leakage that could result from steam generator tube ruptures.”	Addressed for US-APWR in DCD section 15.6.3
II.S	PRA Beyond Design Certification	Issue does not identify any new requirements.	No new requirements.

Table 1.9.5-2 Detailed Treatment of Requirements of SECY-93-087 (sheet 15 of 15)

Issue Number	Description	Requirements	Comment
II.T	Control Room Annunciator (Alarm) Reliability	Approved by Commission in SRM dated 7/21/93. “The Commission approves the staff's recommendation that the alarm system for ALWRs should meet the applicable EPRI requirements for redundancy, independence, and separation. In addition, alarms that are provided for manually controlled actions for which no automatic control is provided and that are required for the safety systems to accomplish their safety functions, shall meet the applicable requirements for Class 1E equipment and circuits.”	Addressed for US-APWR in DCD section 7.5.1.
III.A	Regulatory Treatment of Nonsafety Systems in Passive Designs	This issue is addressed in the consolidated SECY-94-084 and SECY-95-132.	See table 1.9.5-3.
III.B	Definition of Passive Failure	This issue is addressed in the consolidated SECY-94-084 and SECY-95-132.	See table 1.9.5-3.
III.C	SBWR Stability	This is specifically a boiling water reactor (BWR) issue.	Not applicable to the US-APWR.
III.D	Safe Shutdown Requirements	This issue is addressed in the consolidated SECY-94-084 and SECY-95-132.	See table 1.9.5-3.
III.E	Control Room Habitability	This issue is addressed in the consolidated SECY-94-084 and SECY-95-132.	See table 1.9.5-3.
III.F	Radionuclide Attenuation	The concern of this issue was over the lack of a containment spray system in some advanced plant designs. The US-APWR has a containment spray system.	Not applicable to the US-APWR.
III.G	Simplification of Offsite Emergency Planning	SECY-93-087 was inconclusive on this issue and did not invoke any new requirements.	No new requirements.
III.H	Role of the Passive Plant Control Room Operator	Approved by Commission in SRM dated 7/21/93. “The Commission approves the staff's recommendation that sufficient man-in-the-loop testing and evaluation must be performed. In addition, a fully functional integrated control room prototype is likely to be necessary for passive plant control room designs to demonstrate that functions and tasks are properly integrated into the man/machine interface.”	Addressed for US-APWR in DCD section 18.10.

SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Design (RTNSS)" and SECY-95-132, "Policy and Technical Issues Associated with Regulatory Treatment of Non-Safety Systems in Passive Plant Designs"

The net results of SECY-94-084, SECY-95-132, the Commission's Staff Requirements Memoranda (SRM) of June 30, 1994 and June 28, 1995, and the NRC staff's consolidation of the two SECY letters in a July 24, 1995 memo, were approved positions on eight issues. They are repeated from the 7/24/95 memo as follows:

A. Regulatory Treatment of Non-Safety Systems (RTNSS)
B. Definition of passive failure
C. Safe shutdown requirements
D. Control room habitability
E. Reliability assurance program
F. Station blackout
G. Electrical distribution
H. Inservice testing of pumps and valves

Table 1.9.5-3, "Detailed Treatment of Requirements of SECY-94-084 and SECY-95-132", is presented below and shows the eight issues as they are designated in the consolidated SECY letters and reiterated above. Each table entry contains the letter designation and title of the issue, requirements that are applicable to the US-APWR, and a comment on where the issue is treated in this DCD. Because the descriptions in the SECY letters were in some cases quite lengthy and contained a lot of history relating to the issues, the "Requirements" column presents excerpts from the letters that are intended to capture the requirements without being a substitute for the letters themselves.

Table 1.9.5-3 Detailed Treatment of Requirements of SECY-94-084 and SECY-95-132 (sheet 1 of 16)

Issue Designation	Description	Requirements	Comment
A	Regulatory Treatment of Non-Safety Systems (RTNSS)	This issue was driven primarily by NRC's concern over the prevalent reliance on passive systems and consideration of active systems as non-safety systems in advanced reactor designs. They have cited "residual uncertainties associated with passive safety system performance increase the importance of active systems in providing defense-in-depth functions to the passive systems" and "exclusive reliance on passive systems in meeting current licensing criteria is a departure from current design philosophy and licensing practice". In this SECY, NRC developed with EPRI "a process for maintaining appropriate regulatory oversight of these active systems in the passive ALWR designs" that "specifies requirements concerning design and performance of active systems and equipment that perform non-safety, defense-in-depth functions". The process depends on a "focused PRA" to select non-safety active systems important to risk.	For the US-APWR, RTNSS applies primarily to fire protection systems, the station blackout event, and the anticipated transient without scram (ATWS) event. The RTNSS program for control of RTNSS items is described in the DCD in chapter 17(MHI'S QAPD). Should other RTNSS equipment outside the realm of fire protection, SBO and ATWS be added as a result of PRA insights, the QA program exists for ensuring the reliability of these items. The US-APWR treatment of station blackout, addressing the requirements appropriate for design certification, is presented in DCD sections 8.1.5, 8.2.2, 8.4 and 19.2.2. Fire Protection for US-APWR is addressed in DCD section 9.5.1. ATWS for US-APWR is addressed in DCD section 15.8. In the area of preventing interactions among structures that do and do not contain safety-related equipment, seismic classification, analysis and design is addressed in DCD sections 3.2.1, 3.7.1, 3.7.2, 3.7.3 and 3.8.4. Much of this issue as written in SECY-94-084/SECY-95-132 is not applicable to the US-APWR. MHI's position is that the US-APWR is designed as an evolutionary plant, based on traditional concepts of defense in depth that are heavily reliant on safety-related active systems. The MHI position is that the active systems required to protect safety and risk will already be recognized as safety-related under the US-APWR design, and do not need to be selected by a "focused PRA". MHI will use PRA as required for insights into improved plant performance and severe accidents, and that analysis will be presented in Chapter 19 of the

Table 1.9.5-3 Detailed Treatment of Requirements of SECY-94-084 and SECY-95-132 (sheet 2 of 16)

Issue Designation	Description	Requirements	Comment
			DCD.
B	Definition of Passive Failure	"In SECY-77-439, the staff discussed the distinction between active and passive failures of a system or component. An active failure in a fluid system is (1) the failure of a component which relies on mechanical movement to complete its intended function on demand, or (2) an unintended movement of the component. Examples include the failure of a motor- or air-operated valve to move or to assume its correct position on demand, the spurious opening or closing of a motor- or air-operated valve, or the failure of a pump to start or stop on demand. Such failures can be induced by operator error. A passive failure in a fluid system is a breach in the fluid pressure boundary or a mechanical failure which adversely affects a flow path. Examples include the failure of a check valve to move to its correct position when required and the leakage of fluid from failed components (such as pipes and valves), particularly through a failed seal at a valve or pump or line blockage. Motor-operated valves which have the source of power locked out are allowed to be treated as passive components. In defining a single failure in Appendix A to 10CFRPart 50, the NRC stated that fluid and electric systems are considered to be designed against an assumed single failure if the system maintains its ability to perform its safety functions in the event of either (1) a single failure of any active component (assuming passive components function properly) or (2) a single failure of a passive component (assuming active components function properly). The NRC further noted that single failures of passive components in electric systems should be assumed in designing against a single failure. Thus, no distinction is made between failures of active and passive components for electric systems, and all such failures must be considered in applying the single failure criterion. Appendix A also states that the conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are being developed. In SECY-77-439, the staff stated the following: on the basis of the licensing review experience accumulated in the period since 1969, it has been judged in most instances that the probability of	Addressed for US-APWR in DCD sections 7.1.3, 7.2.1, 7.3.1and 15.0.4.

Table 1.9.5-3 Detailed Treatment of Requirements of SECY-94-084 and SECY-95-132 (sheet 3 of 16)

Issue Designation	Description	Requirements	Comment
B	Definition of Passive Failure (continued)	most types of passive failures in fluid systems is sufficiently small that they need not be assumed in addition to the initiating failure in the application of single failure criterion to assure safety of a nuclear power plant. In keeping with the defense-in-depth approach, the staff does consider the effects of certain passive failures (e.g., check valve failure, medium- or high-energy pipe failure, and valve stem or bonnet failure) as potential accident initiators. In licensing reviews, however, only on a long-term basis does the staff consider passive failures in fluid systems as potential accident initiators in addition to initiating events. For example, Section 6.3 of the Standard Review Plan (SRP) requires consideration of passive failures in the emergency core cooling system during the recirculation cooling mode following emergency cooling injection, but does not define such a failure. The staff finds no reason to alter this regulatory practice for the passive ALWR designs, except for check valves as discussed below…" "The staff recommends that the Commission approve the staff's proposal to maintain the current licensing practice for passive component failures on the passive ALWR designs, and to redefine check valves, except for those whose proper function can be demonstrated and documented in the passive safety systems as active components subject to single failure consideration."	
C	Safe Shutdown Requirements	"In GDC 34 of Appendix A to 10CFRPart 50, the NRC regulations require that the design include a residual heat removal (RHR) system to remove residual heat from the reactor core so that specified acceptable fuel design limits (SAFDLs) and the design conditions of the reactor coolant pressure boundary are not exceeded. GDC 34 further requires suitable redundancy of the components and features of the RHR system to ensure that the system safety functions can be accomplished, assuming a loss-of-offsite power or onsite power, coincident with a single failure. The NRC promulgated these requirements to ensure that the RHR system is available for long-term cooling to ensure a safe shutdown state…" "The regulation does not define safe shutdown of the plant after normal operation or a design basis accident, nor does it define what constitutes a safe shutdown state. In implementing the	Because the US-APWR Residual Heat Removal (RHR) System is a multi-train, safety grade, active system, much of the underlying concern of this issue is not applicable. The function of the US-APWR RHR system and its ability to maintain safe shutdown are addressed in DCD sections 3.1.4, 5.4.7 and 7.4.

Table 1.9.5-3 Detailed Treatment of Requirements of SECY-94-084 and SECY-95-132 (sheet 4 of 16)

Issue Designation	Description	Requirements	Comment
C	Safe Shutdown Requirements (continued)	GDC 34 requirements, the staff specified in Regulatory Guide (RG) 1.139 "Guidance for Residual Heat Removal,' and Branch Technical Position (BTP) RSB 5-1 the conditions for cold shutdown (93.3 0C (200 OF) for a PWR and 100 'C (212 OF) for a BWR) using only safety-grade systems within 36 hours. In the regulatory guide, the staff presents the basis for this requirement as follows: even though it may generally be considered safe to maintain a reactor in a hot standby condition for a long time, experience shows that there have been events that required eventual cooldown and long-term cooling until the reactor coolant system was cold enough to perform inspection and repairs. It is therefore obvious that the ability to transfer heat from the reactor to the environment after a shutdown is an important safety function for both PWRs and BWRs. Consequently, it is essential that a power plant have the capability to go from hot standby to cold shutdown conditions. .under any accident conditions…” “The staff is concerned that, with the passive system design basis of 72-hour capability, the passive RHR system water pool, without refill, will have water capacity to permit only 72 hours of operation after a scram. A long-term safe stable condition, however, can be maintained if a reliable non-safety support system or equipment is available to replenish the water pool to sustain long term operation of the passive RHR system after 72 hours. The passive URD requires that non-safety equipment necessary for plant recovery after the assumed 72-hours accident duration be designed for the expected environment, and that only simple, unambiguous operator actions and easily accomplished offsite assistance be necessary after 72 hours to prevent fuel damage. The staff recommended in Section A of this paper that the Commission approve an acceptable process for resolving the RTNSS issue. With an acceptable resolution of the RTNSS issue, the staff expects that non-safety support systems and equipment and active decay heat removal systems will be evaluated for their risk significance and will meet appropriate design and reliability criteria to provide backup capability to passive systems beyond 72 hours. This will ensure proper operation of the passive RHR system to maintain a safe stable condition over the long term, as well as reliable	

Table 1.9.5-3 Detailed Treatment of Requirements of SECY-94-084 and SECY-95-132 (sheet 5 of 16)

Issue Designation	Description	Requirements	Comment
C	Safe Shutdown Requirements (continued)	non-safety systems that will be necessary to bring the plant to cold shutdown conditions. The staff concludes that cold shutdown is not the only safe stable shutdown condition which can maintain the fuel and reactor coolant boundary within acceptable limits, and that the EPRI proposed 215.6 °C (420 °F) as a safe stable shutdown condition is acceptable on the basis of acceptable passive safety system performance and acceptable resolution of the regulatory treatment of non-safety systems. The staff recommends that the Commission approve the EPRI's proposed 215.6 °C (420 °F) or below, rather than the cold shutdown condition required by RG 1.139 as a safe stable condition which the passive decay heat removal systems must be capable of achieving and maintaining following non-LOCA events, This recommendation is predicated on an acceptable passive safety system performance and an acceptable resolution of the issue of regulatory treatment of non-safety systems."	
D	Control Room Habitability	"General Design Criterion (GDC) 19 of Appendix A to 10CFRPart 50 states that (1) a control room should be provided from which actions can be taken to operate the nuclear power plant safely under normal conditions and to maintain it in a safe condition under accident conditions including a loss-of-coolant accident and (2) adequate radiation protection should be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. SRP Section 6.4, "Control Room Habitability Systems," defines the acceptable operator dose criteria in terms of specific whole-body and critical organ doses (5 rem to the whole body and 30 rem each to the thyroid and skin). In current plants, safety-grade, filtered control room heating, ventilation, and air conditioning (HVAC) systems with charcoal absorbers are used to ensure that radiation doses to operators will be maintained within the GDC 19 limits in the event of an accident…" "The staff recommends that the Commission approve the following positions on control room habitability for passive plants:	Control room habitability is addressed in DCD sections 3.1.2, 6.4 and 15.

Table 1.9.5-3 Detailed Treatment of Requirements of SECY-94-084 and SECY-95-132 (sheet 6 of 16)

Issue Designation	Description	Requirements	Comment
D	Control Room Habitability (continued)	1. The concept of using a passive. safety-grade control room pressurization system is acceptable. The proposed design would use bottled air to keep operator doses within the limits of GDC 19 and Section 6.4. Revision 2 of the SRP for the first 72 hours of the event, and safety-grade connections for the pressurization system to allow the use of offsite portable air supplies if needed after 72 hours to minimize operator doses for the duration of the accident. 2. COL holders must demonstrate, through performance of the applicable ITAAC and periodic surveillance tests, the capability of the pressurization system and the capability and availability of backup air supplies to maintain control room habitability for the duration of the accident. 3. The regulatory treatment of the Portable air supply and the nonsafety-grade ventilation system should be in accordance with the staff's position on the regulatory treatment of non-safety systems process described in Section A of this paper.”	
E	Reliability Assurance Program	“In SECY-89-013, "Design Requirements Related to the Evolutionary ALWR," the staff stated that the reliability assurance program (RAP) would be required for design certification to ensure that the design reliability of safety significant SSCs is maintained over the life of a plant…” “The staff considers the RAP for advanced reactors to have two stages. The first stage applies prior to initial fuel load, and is referred to as the design reliability assurance program (D-RAP). The D-RAP can be divided into the design certification phase, the COL application phase, and the COL holder phase. An applicant for design certification would be required, by the D-RAP applicable regulation, to establish the scope, purpose, objective, and essential elements of an effective RAP and would implement those portions of the D-RAP that apply to design certification. A combined license (COL) applicant will be responsible for augmenting and completing the remainder of the D-RAP to include any site-specific design information and identify and prioritize the risk-significant SSCs as required by the D-RAP applicable regulation. Once the site-specific D-RAP has been established and the risk significant SSCs identified and prioritized, the procurement, fabrication, construction, and preoperational testing would be implemented in accordance with	The US-APWR Reliability Assurance Program, addressing the requirements appropriate for design certification, is presented in DCD section 17.4.

Table 1.9.5-3 Detailed Treatment of Requirements of SECY-94-084 and SECY-95-132 (sheet 7 of 16)

Issue Designation	Description	Requirements	Comment
E	Reliability Assurance Program (continued)	the COL holder's D-RAP or other programs and would be verified using the ITAAC process." "The second stage applies to reliability assurance activities for the operations phase of the plant life cycle. These activities can be integrated into existing programs (e.g., maintenance, surveillance testing, inservice inspection, inservice testing, and quality assurance)…" "An application for advanced reactor design certification or a combined license must contain: (1) the description of the reliability assurance program used during the design that includes scope, purpose and objectives: (2) the process used to evaluate and prioritize the structures, systems and components in the design, based on their degree of risk significance: (3) a list of the structures. systems and components designated as risk significant: and (4) for those SSCs designated as risk significant: (i) a process to determine dominant failure modes that considered industry experience, analytical models. and applicable requirements: and (ii) key assumptions and risk insights from probabilistic, deterministic, or other methods that considered operations, maintenance, and monitoring activities. Each licensee that references the advanced reactor design must implement the design reliability assurance program approved by the NRC. The applicable regulation for D-RAP must be satisfied for the design certification, the COL application, and by the COL holder. The design certification D-RAP will be verified using the staff's safety evaluation review process. The COL Applicant's D-RAP will be approved by the staff prior to granting a COL. The COL Applicant's D-RAP should incorporate all aspects of reliability assurance that will be accomplished prior to fuel load (i.e., procurement, fabrication, construction, and preoperational testing phases). The D-RAP shall be verified using the inspections, tests, analyses, and acceptance criteria (ITAAC) process. The SSAR should include the details of the D-RAP, including the conceptual framework, program structure, and essential elements. The SSAR for the D-RAP should also (1) identify, prioritize, and list the risk-significant SSCs based on the design certification PRA, deterministic methods, such as, but not	

Table 1.9.5-3 Detailed Treatment of Requirements of SECY-94-084 and SECY-95-132 (sheet 8 of 16)

Issue Designation	Description	Requirements	Comment
E	Reliability Assurance Program (continued)	limited to, nuclear plant operating experience and relevant component failure data bases; (2) describe the methods used to ensure that the design certification applicant's design organization determines that significant design assumptions, such as equipment reliability and unavailability, are realistic and achievable; (3) include design assumption information for the equipment procurement process; and (4) provide these design assumptions for the COL Applicant's consideration in planning operations-phase reliability assurance activities. A COL Applicant would augment the design certification D-RAP with site-specific design information and would implement the balance of the D-RAP, including information for the procurement, fabrication, construction, and preoperational testing phases that will be completed prior to fuel load. The COL Applicant would incorporate into the existing maintenance and QA programs operations-phase reliability assurance activities. The COL Applicant's D-RAP will be reviewed and approved by the NRC staff at the time the COL is issued, with all subsequent changes subject to NRC staff approval prior to implementation, similar to current QA Programs. The staff would verify implementation of the D-RAP with the ITAAC process as well as inspections and audits during detailed design, procurement, fabrication, construction, and preoperational testing prior to fuel load and would continue to inspect and audit implementation of the operations-phase reliability assurance activities for the duration of the license using the maintenance and quality assurance regulations (i.e., 10CFR50.65 and 10CFRPart 50, Appendix B)."	
F	Station Blackout	"The station blackout rule (10CFR50.63) allows design alternatives to ensure that an operating plant can be safely shut down if all ac power (offsite and onsite) is unavailable. In SECY-90-016, "Evolutionary LWR Certification Issues and Their Relationship to Current Regulatory Requirements," the staff concluded that the preferred method of demonstrating compliance with 10CFR50.63 for evolutionary designs is by installing a spare (full-capacity) alternate ac power source of a diverse design. The passive ALWR designs do not require ac power for 72 hours following an event and will include provisions for offsite	The US-APWR treatment of station blackout, addressing the requirements appropriate for design certification, is presented in DCD sections 8.1.5, 8.2.2, 8.4 and 19.2.2.3.

Table 1.9.5-3 Detailed Treatment of Requirements of SECY-94-084 and SECY-95-132 (sheet 9 of 16)

Issue Designation	Description	Requirements	Comment
F	Station Blackout (continued)	assistance (including additional ac power) beyond 72 hours. Thus, EPRI and the passive plant designers have not made the same provisions for certain ac power system features found in existing plants or in the evolutionary plant designs. The passive designs lack an alternate ac power source and a normally available second offsite power circuit. They also use non-safety-grade emergency generators (typically diesel generators on existing plants) and non-safety-grade ac electrical distribution systems. Each of these is addressed below. An alternate ac power source or the ability to cope with a station blackout for a specified duration are the options available to comply with the requirements of the station blackout rule. The staff prefers the use of an alternate ac power source to meet the requirements of the rule in evolutionary plant designs because it offers several advantages. An alternative ac power source could power a larger complement of shutdown equipment and bring the plant to cold shutdown, it could be used for other purposes in addition to station blackout, it is not limited by time while providing power during a station blackout, and it provided for a uniform hardware approach requiring less analysis and fewer specialized operating procedures. However, EPRI and the passive plant designers stated that the passive plants will be designed to remain in a safe and stable condition for 72 hours without ac power, and without operator actions. This period can be extended well beyond 72 hours with preplanned offsite assistance and simple operator actions. This strong coping capability, reduced reliance on ac power, and minimal required operator actions would seem to obviate the need for an alternate ac source. However, EPRI also reduced the requirements on certain other ac power system features on which the station blackout requirements were premised. GDC 17 requires that two offsite power circuits be available during plant operating modes. In the URD for the passive plant designs, however, EPRI required that the design include only a single offsite power circuit to supply the plant loads during operating modes. A second circuit is required in the passive plant designs for use only "in the event of an extended unavailability of the normal power supply, e.g., during plant outages." In the passive URD, EPRI stated as rationale for this	

Table 1.9.5-3 Detailed Treatment of Requirements of SECY-94-084 and SECY-95-132 (sheet 10 of 16)

Issue Designation	Description	Requirements	Comment
F	Station Blackout (continued)	requirement that it will ensure that adequate power supply will be maintained (either from another offsite source at the same site or from offsite) at all times during plant shutdown modes when major maintenance is required on one of the onsite power sources or on the normal offsite circuit. The staff believes that if two offsite circuits are not available during plant operating modes, the frequency of loss-of-offsite power events and the time needed to recover offsite power will likely be greater than they are for existing plants. The designer should evaluate these difficulties against the stronger coping capability of the passive plant designs. The passive URD also requires that Installed spare main and auxiliary transformers be available to replace their counterparts in no more than 12 hours, which should help to reduce the likelihood of an extended loss of the single normally available offsite power circuit. In addition, EPRI and the passive plant designers are providing non-safety grade onsite emergency generators (diesel generators or combustion turbine generators) and non-safety-grade ac electrical distribution systems. The staff believes that at least two aspects of this approach could directly affect station blackout. EPRI specified an overall reliability of 0.9 for the emergency generators. The maintenance unavailability and the start/run reliability that EPRI indicates would be consistent with this overall reliability are worse than typically seen on safety-grade diesel generators in existing plants. Secondly, EPRI stated that the emergency generators could be used as peaking units to supply power to the grid. EPRI and the passive plant designers, however, have not provided for a distribution system design that would facilitate the use of the emergency generator in this manner, since it would require that the power be delivered to the grid through the plant buses and distribution circuits. Both of the foregoing provisions could increase the likelihood of a station blackout. Each of the ac power system features discussed in this section shares two aspects. They are viewed as non-safety systems or components for the passive plant designs, and their potential negative effects on station blackout must be judged against the strong coping capability of the passive plants. The staff, therefore, concludes that this issue is a good candidate to be	

Table 1.9.5-3 Detailed Treatment of Requirements of SECY-94-084 and SECY-95-132 (sheet 11 of 16)

Issue Designation	Description	Requirements	Comment
F	Station Blackout (continued)	addressed by the process for the regulatory treatment of non-safety systems described in Section A of this paper. The staff recommends that the Commission approve the staff's proposal to resolve the station blackout issue and related GDC 17 issues on passive ALWR designs by evaluating the ac power system features discussed above under the process defined herein for resolving the regulatory treatment of non-safety systems issue. The staff will pursue regulatory treatment of these features if they are found to be risk significant or are relied on to meet the R/A missions.”	
G	Electrical Distribution	“In SECY-91-078, "Chapter 11 of the Electric Power Research Institute's (EPRI's) Requirements Document and Additional Evolutionary Light-Water Reactor (LWR) Certification Issues," March 25, 1992, the staff recommended that the Commission approve its position that an evolutionary plant design should include the following elements: an alternate source of power to the non-safety loads unless the designer can demonstrate that the design margins will result in transients for a loss of non-safety power event that are no more severe than those associated with the turbine-trip-only event in current plants at least one offsite circuit to each redundant safety division supplied directly from one of the offsite power sources with no intervening nonsafety buses in such a manner that the offsite source can power the safety buses upon a failure of any non-safety bus. In the staff requirements memorandum (SRH) of August 15, 1991, the Commission approved the staff's positions. In a letter of May 5, 1992, EPRI stated that this issue does not apply to passive designs. The first position identified above involved the lack of a second source of power on evolutionary plant designs (typically an offsite circuit on existing plants) to the traditional non-safety electrical buses that power plant loads required for unit operation. These loads include the reactor coolant pumps (recirculation pumps for BWRs), feedwater pumps, condensate pumps, and circulating water pumps. In SECY-91-078, the staff took this position to ensure that a second power source be	This aspect of electrical distribution is addressed for US-APWR in DCD sections 3.1.2.8, 3.1.2.9, 8.1 and 8.2.

Table 1.9.5-3 Detailed Treatment of Requirements of SECY-94-084 and SECY-95-132 (sheet 12 of 16)

Issue Designation	Description	Requirements	Comment
G	Electrical Distribution (continued)	provided to a sufficient string of these traditional non-safety loads so that forced circulation could be maintained, and the operator would have the normal complement of non-safety equipment available to bring the plant to a stable shutdown condition after a loss of the normal power supply and plant trip. In the passive plant designs, the same complement of loads identified above (with the exception of the recirculation pumps in the BWRs that are no longer used) are fed from traditional non-safety load buses with only a single source of offsite power available to them. However, recognizing the strong coping capability without ac power of the passive plant designs, EPRI has not required that a second offsite power source normally be available to any of the plant loads, non-safety or safety. The staff took the second position in SECY-91-078 to address the connection of at least one offsite circuit directly to the safety buses with no intervening non-safety buses. In the evolutionary designs, this was accomplished with a direct connection of the second offsite circuit to the safety-grade diesel generator buses. The configuration shown in the passive URD is similar to that for the evolutionary plant, except that the second circuit is only intended to be available during extended plant outages as a maintenance type feed. Furthermore, the diesel generator buses to which the second circuit is connected and most of the ac distribution system are non-safety-grade. Thus, intervening-non-safety buses and one transformer are located between the second circuit and the safety-grade ac bus that is now located at the 480-volt motor control center level. The one normally available offsite power circuit connection to the safety buses also has a number of intervening non-safety buses and transformers. Both of the positions on this issue are closely tied to the lack of a second normally available offsite circuit identified in Section F of this paper. The staff, therefore, concludes that this issue is a good candidate to be addressed by the process for the regulatory treatment of non-safety systems described in Section A of this paper. The staff recommends that the Commission approve the staff's proposal to resolve the electrical distribution issue on passive ALWR designs by evaluating the ac power system features	

Table 1.9.5-3 Detailed Treatment of Requirements of SECY-94-084 and SECY-95-132 (sheet 13 of 16)

Issue Designation	Description	Requirements	Comment
G	Electrical Distribution (continued)	using the process defined herein for resolving the regulatory treatment of non-safety systems. The staff will pursue regulatory treatment of these features if they are found to be risk significant or are relied on to meet the R/A missions.“	
H	Inservice Testing of Pumps and Valves	“In SECY-90-016, the staff recommended that the Commission approve the following four positions for the inservice testing of safety-related pumps and valves beyond the current regulatory requirements in 10CFR50.55(a) for ASME Code Class 1, 2, and 3 components: Piping design should incorporate provisions for full-flow testing (maximum design flow) of pumps and check valves. Designs should incorporate provisions to test MOVs under design-basis differential pressure. Check valve testing should incorporate the use of advanced, non-intrusive techniques, to address degradation and performance characteristics. A program should be established to determine the frequency necessary to disassemble and inspect pumps and valves to detect unacceptable degradation that cannot be detected through the use of advanced, non-intrusive techniques…” …“The staff recommended that the Commission approve the position that these requirements should be imposed on passive ALWRs. The staff also concluded that additional inservice testing requirements may be necessary for certain pumps and valves in passive plant designs. The unique passive plant design relies significantly on passive safety systems, but also depends on non-safety systems (which are traditionally safety-related systems in current light-water reactors) to prevent challenges to passive systems. Therefore, the reliable performance of individual components is a very significant factor in enhancing the safety of passive plant design. The staff recommends that the following provisions be applied to passive ALWR plants to ensure reliable component performance. 1. Important non-safety-related components are not required to meet criteria similar to safety-grade criteria. However. the non-safety-related piping systems with functions that have been identified as being important by the RTNSS process should be designed to accommodate testing of Rumps and Specific positions	In-service inspection of pumps and valves is addressed for US-APWR in DCD sections 3.1.4.3, 3.1.4.7, 6.6, 13.4 and Chapter 16 (Tech Spec section 5.5).

Table 1.9.5-3 Detailed Treatment of Requirements of SECY-94-084 and SECY-95-132 (sheet 14 of 16)

Issue Designation	Description	Requirements	Comment
H	Inservice Testing of Pumps and Valves (continued)	on the inservice testing requirements for those components will be determined as a part of the staff's review of plant specific implementation of the regulatory treatment of non-safety systems for passive reactor designs. 2. ASME/ANSI OM Part 10, referenced in Section XI, ASNE Code, 1989 Edition, provides for the relaxation in the valve testing frequency from quarterly intervals to cold shutdowns or refueling outages if testing during normal plant operations or cold shutdown conditions is not practical. The rules of ON 10 do not accommodate quarterly testing because they address the testing of valves in currently operating reactors, where the detailed piping system designs were completed before the NRC promulgated the inservice testing requirements. The vendors for advanced passive reactors, for which the final designs are not complete, have sufficient time to include provisions in their piping system designs to allow testing at power. Quarterly testing is the base testing frequency in the Code and the original intent of the Code. Furthermore, the COL holder may need to test more frequently than during cold shutdowns or at every refueling outage to ensure that the reliable performance of components is commensurate with the importance of the safety functions to be performed and with system reliability goals. Therefore, to the extent practicable, the passive ALWR piping systems should be designed to accommodate the applicable Code requirements for the quarterly testing of valves. However, design configuration changes to accommodate Code-required quarterly testing should be done only if the benefits of the test outweigh the potential risk. 3. The passive system designs should incorporate provisions (1) to permit all critical check valves to be tested for performance, to the extent practicable, in	

Table 1.9.5-3 Detailed Treatment of Requirements of SECY-94-084 and SECY-95-132 (sheet 15 of 16)

Issue Designation	Description	Requirements	Comment
H	Inservice Testing of Pumps and Valves (continued)	both forward- and reverse-flow directions, although the demonstration of a non-safety direction test need not be as rigorous as the corresponding safety direction test. and (2) to verify the movement of each check valve's operator during inservice testing by observing a direct instrumentation indication of the valve position such as a position indicator or by using non-intrusive test methods. 4. The passive system designs should incorporate provisions to test safety related power-operated valves under design-basis differential pressure and flow, Prior to installation, the design capability of these types of valves should be demonstrated by a qualification test, Prior to initial startup. the valve capability under design-basis differential pressure and flow should be verified by a preoperational test. During the operational phase, the valve capability under design-basis differential pressure and flow should be verified periodically through a program similar to that being developed for MOVs in GL 89-10. Similarly, to the extent practicable, the design of non-safety-related piping systems with functions under design-basis condition that have been identified as being important by the RTNSS process should incorporate provisions to periodically test power operated valves in the system during operations to assure that the valves meet their intended functions under design-basis conditions. 5. To the extent practicable, provisions should be incorporated in the design to assure that MOVs in safety-related systems are capable of recovering from mis-positioning. Mis-positioning may occur through actions taken locally (manual or electrical), at a motor control center, or in the control room, and includes deliberate changes of valve position to perform surveillance testing. The staff will determine if and the extent to which this concept should be applied to	

Table 1.9.5-3 Detailed Treatment of Requirements of SECY-94-084 and SECY-95-132 (sheet 16 of 16)

Issue Designation	Description	Requirements	Comment
H	Inservice Testing of Pumps and Valves (continued)	MOVs in important non-safety-related systems when the staff reviews the implementation of the regulatory treatment of non-safety systems.	

SECY-94-302, "Source-Term-Related Technical and Licensing Issues Relating to Evolutionary and Passive Light-Water-Reactor Designs"

In SECY-94-302, issued in December 1994, the NRC staff noted that the review process for NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants", was essentially complete, and presented positions on:

- Closure of source term-related issues in its Safety Evaluation Reports (SERs) for EPRI requirements documents for both evolutionary and passive plant designs, and
- Generic implementation of source term-related issues in evolutionary and passive LWR design certification reviews.

NUREG-1465 was initiated to develop alternative source terms for use in accident analysis, as summarized below in language cited from the preface of the final version of NUREG-1465, issued in February 1995:

"In 1962, the Atomic Energy Commission issued Technical Information Document (IMD) 14844, "Calculation of Distance Factors for Power and Test Reactors." In this document, a release of fission products from the core of a light-water reactor (LWR) into the containment atmosphere ("source term") was postulated for the purpose of calculating off-site doses in accordance with 10CFRPart 100, "Reactor Site Criteria." The source term postulated an accident that resulted in substantial meltdown of the core, and the fission products assumed released into the containment were based on an understanding at that time of fission product behavior. In addition to site suitability, the regulatory applications of this source term (in conjunction with the dose calculation methodology) affect the design of a wide range of plant systems. In the past 30 years, substantial information has been developed updating our knowledge about severe LWR accidents and the resulting behavior of the released fission products. The purpose of this document is to provide a postulated fission product source term released into containment that is based on current understanding of LWR accidents and fission product behavior. The information contained in this document is applicable to LWR designs and is intended to form the basis for the development of regulatory guidance, primarily for future LWRs. This report will serve as a basis for possible changes to regulatory requirements. However, acceptance of any proposed changes will be on a case-by-case basis. Source terms for future reactors may differ from those presented in this report, which are based upon insights derived from current generation light-water reactors. An applicant may propose changes in source term parameters (timing, release magnitude, and chemical form) from those contained in this report, based upon and justified by design specific features."

The NRC staff did not request a review of the positions presented in SECY-94-302 by the Commission, because they were primarily technical applications of previous Commission policy decisions. Hence, there is no Staff Requirements Memo (SRM) pertaining to SECY-94-302. The staff stated that they intended to use the reactor accident source terms given in NUREG-1465 in radiological consequence assessments in the following areas of evolutionary and passive LWR design certification reviews:

6. equipment qualification

7. control room habitability

8. engineered safety features atmosphere cleanup systems

9. primary containment leak rate

10. containment isolation timing

11. post-accident sampling

12. shielding and vital area access

The NRC staff presented and highlighted the five (5) "most significant issues" in the transmittal letter for SECY-94-302, and included the other seven issues in Attachment 1 to the letter. They were presented in the context of ongoing design reviews of several specific advanced light water reactors at the time of publication in 1994, but seemed to have some potential for generic significance in future reviews. It would appear from the presentation of the 5 most significant issues summarized in the letter that three of them would apply to the US-APWR, and the other two, while possibly generic, are nonetheless BWR issues and thus not applicable to the US-APWR. The other seven (7) issues were presented in Attachment 1 of the letter, and are all design-specific and thus not applicable to the US-APWR. All 12 issues are summarized in Table 1.9.5-4, Detailed Treatment of Requirements of SECY-94-302, below.

Table 1.9.5-4 Detailed Treatment of Requirements of SECY-94-302 (sheet 1 of 3)

Issue Number	Description	Requirements	Comment
1	Selective Use of Accident Source Terms Given in Draft NUREG-1465	“Selectively use the source terms given in draft NUREG-1465 using only "Gap Release" and "Early In-Vessel Release" (excluding "Ex-Vessel Release" and "Late In-Vessel Release" associated with vessel failure and core-concrete interaction) in evaluating radiological consequences for DBAs, the DBA radiation environmental qualification of electrical and mechanical equipment important to safety, post-accident shielding, and Three Wile Island-related requirements (Issue 1).”	Addressed in US-APWR DCD sections 3.11.5.2 and 15.0.3.
2	Iodine Chemical Form	“Use the chemical forms of iodine of at least 95 percent cesium iodine as stated in draft NUREG-1465 with 4.85 percent of elemental iodine and hydrogen iodide and 0.15 percent organic iodide (Issue 2).”	Addressed in US-APWR DCD section 15.0.3.
3	Equipment Survivability for Design Features Needed for Severe Accident Mitigation and Containment Integrity	“In radiological assessments of equipment survivability as a result of a severe reactor accident, the staff will require that the equipment and features needed for severe accident prevention, mitigation, and post-accident sampling be designed to provide a reasonable level of confidence that they will operate in a severe accident environment. This environment would include the exvessel release, with proper credit for design features to mitigate that release, and the late in-vessel release In addition to the releases for a DBA (Issue 3).”	Addressed in US-APWR DCD section 19.2.3.3.7 Severe accident dedicated systems are not subject to the environmental qualification requirements of 10 CFR Section 50.49; quality assurance requirements of 10 CFR Part 50, Appendix A. But those necessary in severe accident countermeasures are confirmed to withstand those environmental conditions practically.
4	Radioactive Iodine Deposition on BWR Main Steamlines and Condensers	“The staff will accept the passive BWR plant design without a LCS and allow an appropriate credit for iodine removal In the main steamlfne and condenser following a DBA (Issue 4).”	Included among the top 5 most significant issues, however BWR-specific and not applicable to the US-APWR.

Table 1.9.5-4 Detailed Treatment of Requirements of SECY-94-302 (sheet 2 of 3)

Issue Number	Description	Requirements	Comment
5	Fission-Product Holdup in the Secondary Containment	“The staff is not reviewing the source term related technical and licensing issues for the SBWR design at the present time. However, when the staff resumes its review of the SBWR design, the staff will allow appropriate credit for the SBWR safety envelope based on fission-product holdup and decay within this envelope if (1) the vendor specifies that the secondary containment leakage and mixing performance be consistent with the values used by the staff in its radiological assessment and (2) the COL combined license applicant incorporates the secondary containment leakage value specified by the vendor into the plant-specific technical specifications (Issue 5).”	Included among the top 5 most significant issues, however BWR-specific and not applicable to the US-APWR.
6	Fission-Product Release Timing (for System 80+, AP600, and SBIR)	Not included among the top 5 most significant issues, and presented in the context of resolving a design-specific issue that existed in 1994.	Not applicable to the US-APWR.
7	Aerosol Deposition in Primary Containment (for AP600 and SBWR)	Not included among the top 5 most significant issues, and presented in the context of resolving a design-specific issue that existed in 1994.	Not applicable to the US-APWR.
8	Aerosol Removal by BWR Suppression Pool (for ABWR and SBWR)	Not included among the top 5 most significant issues, and presented in the context of resolving a design-specific issue that existed in 1994.	Not applicable to the US-APWR.
9	Fission-Product Removal by Containment Spray (For AP600)	Not included among the top 5 most significant issues, and presented in the context of resolving a design-specific issue that existed in 1994.	Not applicable to the US-APWR.
10	Radioactive Aerosol and Iodine Removal by Engineered Safety Features (ESF) Atmosphere Cleanup System (for AP600 and SBWR)	Not included among the top 5 most significant issues, and presented in the context of resolving a design-specific issue that existed in 1994.	Not applicable to the US-APWR.

Table 1.9.5-4 Detailed Treatment of Requirements of SECY-94-302 (sheet 3 of 3)

Issue Number	Description	Requirements	Comment
11	Atmospheric Diffusion Model for Control Room Habitability Assessment (for CE System 80+, AP600 and SBWR)	Not included among the top 5 most significant issues, and presented in the context of resolving a design-specific issue that existed in 1994.	Not applicable to the US-APWR.
12	Failure of Heat Exchanger Tubes in the Passive Containment Cooling System (for SBWR)	Not included among the top 5 most significant issues, and presented in the context of resolving a design-specific issue that existed in 1994.	Not applicable to the US-APWR.

1.9.6 Combined License Information

COL 1.9(1) *The COL Applicant is to address an evaluation of the applicable RG, SRP, Generic Issues including Three Mile Island (TMI) requirements, and operational experience for the site-specific portion and operational aspect of the facility.*

1.9.7 References

References for MHI test documents and topical reports have been provided previously in sections 1.5.4 and 1.6 of this DCD. References for NRC SECY letters that are the basis for the ALWR design issues evaluation are compiled in the introductory portion of section 1.9.5 of this DCD.

Other general references used in the development of Chapter 1 are as follows:

- 10CFR50, Domestic Licensing of Production and Utilization Facilities.
- 10CFR52, Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants.
- Regulatory Guide 1.206, Combined License Applications for Nuclear Power Plants (LWR Edition) , Revision 0, June 2007.
- NUREG 0800, Standard Review Plan Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants , March 2007.
- US-APWR Design Description, Mitsubishi Heavy Industries, Ltd., October 2006.