



NuScale Standard Plant  
Design Certification Application

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Chapter One  
**Introduction and  
General Description  
of the Plant**

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**PART 2 - TIER 2**

Revision 0  
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## CHAPTER 1 INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

### 1.1 Introduction

This document represents the Final Safety Analysis Report (FSAR) required under 10 CFR 52.47(a) to be provided as part of an application for a standard design certification under 10 CFR 52, Subpart B and will be referred to as such throughout. It describes the NuScale design, including (1) the design bases and limits on its operation; (2) a safety analysis of the structures, systems, and components and of the facility as a whole; and (3) the information prescribed in 10 CFR 52.47(a) that is relevant to the NuScale design.

A NuScale Power Module (NPM) shown in Figure 1.2-6 and Figure 1.2-7, is a collection of systems, sub-systems, and components that together constitute a modularized, movable, nuclear steam supply system (NSSS). The NPM is composed of a reactor core, a pressurizer, and two steam generators integrated within a reactor pressure vessel (RPV) and housed in a compact steel containment vessel.

The NuScale advanced small modular reactor plant design is scalable, such that from one (1) to twelve (12) NPMs operate within a single Reactor Building. The information provided in this FSAR includes the design of an individual NPM, as well as plant design and interfaces for a 12 NPM facility. In general, chapters describe a single module. Multi-module information is only noted where warranted (e.g., shared systems or analyses such as seismic).

The NuScale design features:

- No AC or DC power required for safe shutdown and cooling
- Compact helical coil steam generators with reactor pressure on the outside of the tubes
- High-strength steel containment immersed in a pool of water
- Sub-atmospheric containment pressure during normal operation
- Small core with a correspondingly small source term
- Comprehensive digital I&C monitoring and control

Important features of a multi-unit plant include:

- a scalable plant design, which allows for incremental plant capacity growth.
- a compact nuclear island.
- the ability to operate in "island mode" (no reliance on off-site power).

#### 1.1.1 Plant Location

The NuScale Power Plant is designed to be located on a site having site characteristics (e.g., seismology, hydrology, meteorology, geology, and other site-related characteristics) bounded by the parameters described in Chapter 2, Site Characteristics.

COL Item 1.1-1: A COL applicant that references the NuScale Power Plant design certification will identify the site-specific plant location.

### 1.1.2 Containment Type

The NuScale containment vessel (CNV) is a supported, cylindrical vessel-type containment that is designed to withstand limiting high-pressure transients. The CNV is an ASME BPV Code Class MC (steel) containment that is designed, analyzed, fabricated, inspected, tested and stamped as an ASME BPV Code Class 1 pressure vessel. The CNV internal pressure is maintained at a vacuum during normal operations and as such insulation materials are not required between the reactor vessel and the CNV. The containment vessels are mounted to the Reactor Building module compartment walls and at the bottom within the Reactor Building pool.

### 1.1.3 Reactor Type

The NuScale nuclear steam supply system (NSSS) is a passive NuScale-designed small modular pressurized water reactor. This design is comprised of an integral power module consisting of a reactor core, two steam generator tube bundles, and a pressurizer contained within a single reactor vessel, along with the containment vessel that immediately surrounds the reactor vessel. This design eliminates the need for external piping to connect the steam generators and pressurizer to the RPV. Natural circulation provides reactor coolant system flow, thereby eliminating the need for reactor coolant pumps.

### 1.1.4 Power Output

A NuScale Power Plant consists of from one to 12 NPMs. Each NPM is rated at 160 MWt (1,920 MWt, total), with approximately 50 MWe (600 MWe total) output. Electrical output is dependent on environmental conditions. When considering house loads, the total net output is approximately 570 MWe for a 12 NPM facility. Design power assumes an additional 2% to account for measurement uncertainty.

### 1.1.5 Schedule

COL Item 1.1-2: A COL applicant that references the NuScale Power Plant design certification will provide the schedules for completion of construction and commercial operation of each power module.

### 1.1.6 Format and Content

#### 1.1.6.1 Regulatory Guide 1.206

The format and content of this FSAR generally follow the format and content guidelines of Regulatory Guide 1.206. However, where applicable, Sections may be skipped or additional sections inserted. In addition, this FSAR includes Chapter 20, Mitigation of Beyond-Design-Basis Events and Chapter 21, Multi-Module Design Considerations, which are not included in the RG 1.206.

#### 1.1.6.2 Standard Review Plan - NuScale Design Specific Review Standard

A NuScale design specific review standard (DSRS) has been developed by the NRC as a supplement to NUREG-0800, "Standard Review Plan for the Review of Safety Analysis

Reports for Nuclear Power Plants: LWR Edition" (SRP). Accordingly, the preparation of this FSAR used the technical guidance provided in the DSRS and SRP as the basis for the NuScale design. A detailed evaluation of conformance with the NuScale DSRS and the SRP is provided in Section 1.9.

#### **1.1.6.3 Text, Tables, and Figures**

Tables and figures are typically identified by the "X.Y" section in which they appear and are numbered sequentially. For example, Table 1.1-1 and Figure 1.1-1 would be the first table and figure appearing in Section 1.1. Figures consist of diagrams, plots, pictures, graphs, or other illustrations. Tables and figures are located at the end of the applicable "X.Y" section immediately following the text. The exception to this is for large "X.Y.Z" sections, in which the tables and figures are numbered sequentially in that section. For example, Table 3.9.3-1 and Figure 3.9.3-1 would be the first table and figure appearing in Section 3.9.3. Again, the tables and figures are located at the end of the applicable section intermediately following the text.

#### **1.1.6.4 Page Numbering**

Section pages are numbered sequentially and are typically identified by the "X.Y" section followed by a sequential number. The exception to this is for chapter appendices, which are numbered by the chapter number and appendix letter followed by a sequential number. For example, 3A-1 is the first page of Appendix A to Chapter 3.

#### **1.1.6.5 Proprietary Information**

This FSAR has no proprietary or safeguards information. Some portions of this FSAR 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 provided with the non-public version of the FSAR. A separate public version of the FSAR is provided that removes the withheld material. Proprietary or safeguards information that is necessary for the complete review of the design certification is provided to the NRC separately in the form of Topical or Technical Reports. A list of Topical and Technical Reports that are incorporated by reference are listed in Tables 1.6-1 and 1.6-2, respectively.

#### **1.1.6.6 Acronyms and Abbreviations**

A list of acronyms and abbreviations used in this FSAR is provided in Table 1.1-1, Acronyms and Abbreviations.

**Table 1.1-1: Acronyms and Abbreviations**

<b>Acronym/Abbreviation</b>	<b>Description</b>
AAC	alternate AC power
AAPS	auxiliary AC power source
AASHTO	American Association of State Highway and Transportation Officials
ABS	auxiliary boiler system
ABVS	Annex Building HVAC system
ABWR	Advanced Boiling Water Reactor
AC	alternating current
ACI	American Concrete Institute
ACM	Availability Controls Manual
ACRS	Advisory Committee on Reactor Safeguards
AEA	Atomic Energy Act
AFU	air filtration unit
AFWS	auxiliary feedwater system
AHJ	authority having jurisdiction
AHU	air handling unit
AIA	Authorized Inspection Agency
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
ALARA	as low as reasonably achievable
ALU	actuation logic unit
ALWR	advanced light water reactor
AMCA	Air Movement and Control Association International, Inc.
ANB	Annex Building
ANS	American Nuclear Society
ANSI	American National Standards Institute
AO	axial offset
AOA	axial offset anomaly
AOO	anticipated operational occurrence
AOV	air-operated valve
API	American Petroleum Institute
APWR	Advanced Pressurized Water Reactor
ARM	airborne radiation monitor
ARO	all rods out
ARS	acceleration response spectra
ASCE	American Society of Civil Engineers
ASD	adjustable speed drive
ASHRAE	American Society of Heating, Refrigerating, and Air-Conditioning Engineers
ASM	American Society for Metals International
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATB	Administration and Training Building
ATWS	anticipated transient without scram
AVT	all-volatile treatment
AWS	American Welding Society
AWWA	American Water Works Association
BAS	boron addition system
BAST	boric acid storage tank
BDBE	beyond design basis event
BDBEE	beyond design basis external event
BDG	backup diesel generator

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym/Abbreviation	Description
BOC	beginning of cycle
BOL	beginning of life
BOP	balance-of-plant
BPDS	balance-of-plant drain system
BPE	bioprocessing equipment
BPSS	backup power supply system
BRL	Ballistic Research Laboratory
BRVS	battery room ventilation system
BTP	Branch Technical Position
BPVC	Boiler and Pressure Vessel Code
BWR	boiling water reactor
CAM	continuous air monitor
CARS	condenser air removal system
CAS	central alarm station
CAS	compressed air system
CCBE	common cause basic event
CCDP	conditional core damage probability
CCF	common cause failure
CCFL	counter current flow limitation
CCFP	conditional containment failure probability
CDF	core damage frequency
CDI	conceptual design information
CDM	certified design material
CEA	control element assembly
CES	containment evacuation system
CET	containment event tree
CEUS	central and eastern United States
CFDS	containment flooding and drain system
CFR	Code of Federal Regulations
CFT	containment flange tool
CHF	critical heat flux
CHFR	critical heat flux ratio
CFWS	condensate and feedwater system
CHRS	containment heat removal system
CHWS	chilled water system
CILRT	containment integrated leak rate test
CIM	civil interface macro
CIP	clean-in-place
CIS	containment isolation system
CIV	containment isolation valve
CLRT	containment leakage rate testing
CMAA	Crane Manufacturers Association of America
CMS	code management software
CMTR	certified material test report
CNTS	containment system
CNV	containment vessel
CNVF	containment vessel failure
COC	certificate of compliance
COL	combined license
COLA	combined license application
COLR	core operating limits report



**Table 1.1-1: Acronyms and Abbreviations (Continued)**

<b>Acronym/Abbreviation</b>	<b>Description</b>
COMS	communication system
CPS	condensate polishing system
CQC	complete quadratic combination
CRA	control rod assembly
CRB	Control Building
CRDM	control rod drive mechanism
CRDS	control rod drive system
CRE	control room envelope
CRHS	control room habitability system
CRM	control rod misoperation
CRVS	normal control room HVAC system
CSA	core support assembly
CSDRS	certified seismic design response spectra
CSDRS-HF	certified seismic design response spectra - high frequency
CSS	containment sampling system
CST	condensate storage tank
CTG	combustion turbine generator
CVAP	Comprehensive Vibration Assessment Program
CVCS	chemical and volume control system
CWS	circulating water system
D3	diversity and defense in depth
DAC	design acceptance criteria
DAS	distributed antenna system
DAS	diverse actuation system
DAW	dry active waste
DBA	design basis accident
DBE	design basis event
DBPB	design basis pipe break
DBST	design basis source term
DBT	design basis tornado
DC	direct current
DCA	Design Certification Application
DCD	Design Control Document (Note - this is synonymous with FSAR in this document)
DCH	direct containment heating
DCS	distributed control system
DDC	distributed Doppler coefficient
DDG	dry dock gate
DGB	Diesel Generator Building
DGBVS	Diesel Generator Building HVAC system
DHRS	decay heat removal system
DIM	display interface module
DMA	dimethylamine
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOE	Department of Energy
DOT	Department of Transportation
DRAP	Design Reliability Assurance Program
DSRS	Design Specific Review Standard
DSS	digital safety system
DSW	dry solid waste
DWS	demineralized water system

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym/Abbreviation	Description
EAB	exclusion area boundary
EAL	Emergency Action Level
ECCS	emergency core cooling system
ECL	effluent concentration limit
EDL	equivalent dead load
EDMG	extensive damage mitigation guidelines
EDNS	normal DC power system
EDSS	highly reliable DC power system
EDSS-C	EDSS-common
EDSS-MS	EDSS-module-specific
EDV	engineering design verification
EFDS	equipment and floor drainage system
EFPY	effective full-power years
EHVS	13.8 kV and switchyard system
EIM	equipment interface module
ELAP	extended loss of AC power
ELVS	low voltage AC electrical distribution system
ELWR	evolutionary light water reactor
EMC	electromagnetic compatibility
EMDAP	evaluation model development and assessment process
EMDM	electromagnetic drive mechanism
EMI	electromagnetic interference
EMVS	medium voltage AC electrical distribution system
EOC	end of cycle
EOF	emergency operations facility
EOL	end of life
EOP	emergency operating procedure
EPA	Environmental Protection Agency
EPG	emergency procedure guidelines
EPRI	Electric Power Research Institute
EPZ	emergency planning zone
EQ	equipment qualification
EQDP	equipment qualification data package
EQRF	equipment qualification record file
ERDA	Energy Research and Development Administration
ERDS	emergency response data system
ERF	emergency response facility
ERO	Emergency Response Organization
ERS	equipment requirement specification
ESAS	emergency safeguards actuation system
ESBWR	Economic Simplified Boiling Water Reactor
ESF	engineered safety feature
ESFAS	engineered safety features actuation system
ESL	equivalent static load
ESP	early site permit
ETA	ethanolamine
ETAP	Electrical Transient Analyzer Program
FA	functional analysis
FAC	flow-accelerated corrosion
FAT	factory acceptance test
FATT	fracture appearance transition temperatures

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym/Abbreviation	Description
FCU	fan coil unit
FDA	final design approval
FDS	fire detection system
FEM	Federation Europeenne de la Manutention
FERC	Federal Energy Regulatory Commission
FFD	fitness-for-duty
FFT	fast Fourier transform
FHA	fire hazards analysis
FHE	fuel handling equipment
FHM	fuel handling machine
FIRS	foundation input response spectra
FIT	flow-indicating transmitter
FIV	flow-induced vibration
FLEX	Diverse and flexible coping strategies based on NRCs Fukushima task force recommendations
FLPRA	flooding probabilistic risk assessment
FMEA	failure modes and effects analysis
FOAK	first-of-a-kind
FOM	figure of merit
FPGA	field programmable gate array
FPP	Fire Protection Program
FPRA	fire probabilistic risk assessment
FPS	fire protection system
FRA	functional requirements analysis
FRP	fiber-reinforced polymer
FSAR	Final Safety Analysis Report (Note - this is synonymous with DCD in this document)
FSG	FLEX support guidelines
FSSA	fire safe shutdown analysis
FSSD	fire safe shutdown
FWB	Fire Water Building
FWH	feedwater heater
FWIV	feedwater isolation valve
FWLB	feedwater line break
FWRV	feedwater regulating valve
FWS	feedwater system
FWTS	feedwater treatment system
GAC	granulated activated charcoal
GDC	General Design Criteria
GLPS	grounding and lightning protection system
GMRS	ground motion response spectra
GQA	graded quality assurance
GRWS	gaseous radioactive waste system
GSI	generic safety issue
GTAW	gas tungsten arc weld
GTS	generic technical specifications
HAZ	heat-affected zone
HCLPF	high confidence of low probability of failure
HCW	high-conductivity waste
HDP	Hardware Development Plan
HDPE	high-density polyethylene
HED	human engineering discrepancy
HEI	Heat Exchanger Institute

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym/Abbreviation	Description
HELB	high-energy line break
HEP	human error probability
HEPA	high-efficiency particulate air
HFE	human factors engineering
HFEITS	human factors engineering issue tracking system
HFP	hot full power
HIC	high integrity container
HIPS	highly integrated protection system
HLHE	heavy load handling equipment
HMI	human machine interface
HOV	hydraulic-operated valve
HP	high pressure
HP-FWH	high pressure feedwater heater
HPM	human performance monitoring
HPN	health physics network
HRA	human reliability analysis
HRS	hardware requirement specification
HSI	human-system interface
HVAC	heating ventilation and air conditioning
HVD	heater vents and drains
HWM	hard-wired module
HZP	hot zero power
I&C	instrumentation and controls
IAB	inadvertent actuation block
IAS	instrument air system
IBC	International Building Code
ICIS	in-core instrumentation system
ICS	integrated control system
ID	inside diameter
IDD	interface design description
IE	infrequent event
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
IES	Illuminating Engineering Society of North America
IET	integral effects test
IGSCC	intergranular stress-corrosion cracking
IHA	important human action
ILRT	integrated leak rate testing
INL	Idaho National Laboratory
INPO	Institute of Nuclear Power Operations
IOTBS	inadvertent opening of the turbine bypass system
IP	implementation plan
IP	intermediate pressure
IP-FWH	intermediate pressure feedwater heater
ISA	integrated safety analysis
ISG	interim staff guidance
ISI	in-service inspection
ISM	independent support motion
ISO	International Organization for Standardization
ISR	integral jet impingement shield and pipe whip restraint
ISRS	in-structure response spectra

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym/Abbreviation	Description
IST	in-service testing
ISV	integrated system validation
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
ITM	inspection, testing, and maintenance
ITP	Initial Test Program
IVR	in-vessel retention
JLD	Japan Lessons-Learned Directorate
LBB	leak-before-break
LCO	limiting condition for operation
LCS	local control station
LCW	low-conductivity waste
LER	Licensee Event Report
LHGR	linear heat generation rate
LLRT	local leak rate test
LLHS	light load handling system
LOCA	loss-of-coolant accident
LOLA	loss of large areas
LOOP	loss of offsite power
LP	low pressure
LP-FWH	low pressure feedwater heater
LPSD	low power and shut down
LPZ	low population zone
LRA	lower riser assembly
LRF	large release frequency
LRVP	liquid ring vacuum pump
LRW	liquid radioactive waste
LRWS	liquid radioactive waste system
LSH	level switch, high
LSL	level switch, low
LSSS	limiting safety system setting
LTC	load manual tap changers
LTCC	long-term core cooling
LTOP	low temperature overpressure protection
LUHS	loss of normal access to the ultimate heat sink
LWMS	liquid waste management system
LWR	light water reactor
MAE	module assembly equipment
MC	main condenser
MCC	motor control center
MCHFR	minimum critical heat flux ratio
MCNP	Monte Carlo N-Particle Transport Code
MCR	main control room
MCS	module control system
MEL	master equipment list
MEMS	meteorological and environmental monitoring system
MFW	main feedwater
MHA	maximum hypothetical accident
MHS	module heatup system
MIB	monitoring and indication bus
MIC	microbiologically induced corrosion
MIT	Massachusetts Institute of Technology

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym/Abbreviation	Description
MLA	module lifting adapter
MLD	master logic diagram
MMI	multi-module issue
MMS	moment magnitude scale
MOC	middle of cycle
MOV	motor-operated valve
MPS	module protection system
MPT	main power transformer
MPU	magnetic speed pickup
MSI	main steam isolation
MSIBV	main steam isolation bypass valves
MSIV	main steam isolation valve
MSLB	main steam line break
MSO	multiple spurious operations
MSPI	mitigating system performance index
MSS	main steam system
MSSV	main steam safety valve
MTC	moderator temperature coefficient
MTU	metric tons, uranium
MWe	megawatt electric
MWS	maintenance workstation
MWt	megawatt thermal
NDE	non-destructive examination
NDS	nitrogen distribution system
NDT	non-destructive testing
NEI	Nuclear Energy Institute
NERC	North American Electric Reliability Corporation
NFA	new fuel assembly
NFE	new fuel elevator
NFJC	new fuel jib crane
NFPA	National Fire Protection Association
NIC	network interface controller
NIST	National Institute of Standards and Technology
NIST-1	NuScale Integral System Test Facility
NMS	neutron monitoring system
NPM	NuScale Power Module
NPP	NuScale Power Plant
NPS	nominal pipe size
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NRF	nuclear reliability factor
NSA	neutron source assembly
NSAC	Nuclear Safety Analysis Center
NSSS	nuclear steam supply system
NTTF	Near-Term Task Force
OBE	operating basis earthquake
OCS	operational condition sampling
OD	outside diameter
ODC	overspeed detection circuit
ODCM	Offsite Dose Calculation Manual
OE	operating experience

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym/Abbreviation	Description
OER	operating experience review
OHLH	overhead heavy load handling system
ORNL	Oak Ridge National Laboratory
ORPP	Operational Radiation Protection Program
OSC	operational support center
OSHA	Occupational Safety and Health Administration
OSP	overspeed protection system
OSU	Oregon State University
P&ID	pipng and instrumentation diagram
PA	protected area
PA/GA	public address/general alarm
PACS	priority actuation and control system
PAM	post-accident monitoring
PBX	private branch exchange
PCA	primary coolant activity
PCCV	prestressed concrete containment vessel
PCP	Process Control Program
PCS	plant control system
PCT	peak cladding temperature
PCUS	pool cleanup system
PDC	power distribution center
PDC	principal design criteria
PDIL	power dependent insertion limit
PDIT	differential pressure indicating transmitter
PFT	process feed tank
PGA	peak ground acceleration
PID	proportional integral derivative
PIRT	phenomena identification and ranking table
PIT	pressure indicating transmitter
PLC	programmable logic controller
PLD	pool leakage detection
PLDD	programmable logic design description
PLDP	Programmable Logic Development Plan
PLDS	pool leakage detection system
PLHGR	peak linear heat generation rate
PLM	priority logic module
PLRS	programmable logic requirement specification
PLS	plant lighting system
PLVVP	Programmable Logic Verification and Validation Plan
PMF	probable maximum flood
PMP	probable maximum precipitation
PORV	power-operated relief valve
POS	plant operating state
POV	power-operated valve
PPE	personnel protective equipment
PPS	plant protection system
PRA	probabilistic risk assessment
PRV	pressure relief valve
PSCIV	primary system containment isolation valves
PSCS	pool surge control system
PSD	power spectra density

**Table 1.1-1: Acronyms and Abbreviations (Continued)**

<b>Acronym/Abbreviation</b>	<b>Description</b>
PSMS	power supply monitoring system
PSS	process sampling system
PST	phase separator tank
PSTN	public switched telephone network
PTAC	performance and test acceptance criteria band
PTS	pressurized thermal shock
PVC	polyvinyl chloride
PVMS	plant-wide video monitoring system
PWHT	post-weld heat treatment
PWR	pressurized water reactor
PWS	potable water system
PWSCC	primary water stress-corrosion cracking
PZR	pressurizer
QA	quality assurance
QAP	Quality Assurance Program
QAPD	Quality Assurance Program Description
QPD	quadrant power difference
QD	quick disconnect
QPF	quadrant power fractions
RAI	request for additional information
RAP	Reliability Assurance Program
RBC	Reactor Building crane
RBCM	Reactor Building components
RBVS	Reactor Building HVAC system
RCA	radiologically controlled area
RCCA	rod control cluster assembly
RCCWS	reactor component cooling water system
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCRA	Resource Conservation and Recovery Act
RCS	reactor coolant system
RDT	reactor drain tank
REA	rod ejection accident
RETS	Radiological Effluent Technical Specifications
RFI	radio frequency interference
RFP	refueling pool
RFT	reactor flange tool
RG	Regulatory Guide
RHR	residual heat removal
RIS	regulatory issue summary
RM	radiation monitoring
RMS	fixed-area radiation monitoring system
RMTS	risk-managed technical specifications
RO	reverse osmosis
ROCA	restricted owner controlled area
ROP	Reactor Oversight Process
RPCS	reactor pool cooling system
RPI	rod position indication
RPM	reactor power module
RPS	reactor protection system
RPV	reactor pressure vessel



Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym/Abbreviation	Description
RRS	required response spectrum
RRV	reactor recirculation valve
RSA	remote shutdown area
RSR	results summary report
RSS	remote shutdown station
RSV	reactor safety valve
RTB	reactor trip breaker
RTD	resistance temperature detector
RTM	requirements traceability matrix
RT <sub>NDT</sub>	reference temperature for nil-ductility transition
RTNSS	regulatory treatment of nonsafety systems
RTP	rated thermal power
RTPTS	reference temperature, pressurized thermal shock
RTS	reactor trip system
RVI	reactor vessel internals
RVV	reactor vent valve
RWB	Radioactive Waste Building
RWBVS	Radioactive Waste Building HVAC system
RWDS	radioactive waste drain system
RWMS	radioactive waste management system
RWSS	raw water supply system
RXB	Reactor Building
RXC	reactor core
S&Q	staffing and qualifications
SAFDL	specified acceptable fuel design limit
SAM	seismic anchor motion
SAMDA	severe accident mitigation design alternative
SAMG	severe accident management guideline
SAR	Safety Analysis Report
SAS	secondary alarm station
SAS	service air system
SAT	site acceptance testing
SBAC	smooth bounding analysis curve
SBLOCA	small-break loss-of-coolant accident
SBM	scheduling and bypass module
SBO	station blackout
SBVS	Security Building HVAC system
SC-I	Seismic Category I
SC-II	Seismic Category II
SC-III	Seismic Category III
SCB	Security Buildings
SCC	stress corrosion-cracking
SCDF	seismic core damage frequency
SCR	silicon controlled rectifier
SCS	secondary sampling system
SCWS	site cooling water system
SDB	safety data bus
SDD	system design description
SDIS	safety display and indication system
SDM	shutdown margin
SDOE	secure development and operational environment

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym/Abbreviation	Description
SDOF	single-degree-of-freedom
SDP	software development process
SDS	site drainage system
SECS	plant security system
SECY	Secretary of the Commission, Office of the NRC
SEL	seismic equipment list
SER	Safety Evaluation Report
SFA	spent fuel assembly
SFCP	Surveillance Frequency Control Program
SFM	safety function module
SFP	spent fuel pool
SFPCS	spent fuel pool cooling system
SFSS	spent fuel storage system
SG	separation group
SG	steam generator
SG	strain gauge
SGI	safeguards information
SGS	steam generator system
SGTF	steam generator tube failure
SICS	safety information and control system
SIL	software integrity level
SLB	steam line break
SLP	site layout plan
SM	single module
SMA	seismic margin analysis
SME	subject matter expert
SMR	small modular reactor
SMS	seismic monitoring system
SNL	Sandia National Laboratory
SNM	special nuclear material
SOCA	security owner controlled area
SOV	solenoid-operated valve
SPAR	standardized plant analysis risk
SPND	self-powered neutron detector
SPS	security power system
SQDP	seismic qualification data package
SQRF	seismic qualification record form
SQUG	Seismic Qualification Utility Group
SREC	standard radiological effluent control
SRI	Stanford Research Institute
SRM	staff requirements memorandum
SRP	Standard Review Plan
SRSS	square root of the sum of the squares
SRST	spent resin storage tank
SRV	sump recirculation valve
SRWS	solid radioactive waste system
SSA	safe shutdown analysis
SSC	structures, systems, and components
SSCIV	secondary system containment isolation valve
SSE	safe shutdown earthquake
SSI	soil-structure interaction

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym/Abbreviation	Description
SSS	secondary sampling system
SSSI	structure-soil-structure interaction
SST	station service transformer
STPA	System-Theoretic Process Analysis
SUNSI	sensitive unclassified non-safeguards information
SVM	schedule and voting module
SWIS	service water intake structure
SWMS	solid waste management system
SWV	shear wave velocity
SWYD	switchyard system
TA	task analysis
TBC	Turbine Building crane
TBS	turbine bypass system
TBVS	Turbine Building HVAC system
T/C	thermocouple
TDH	total dynamic head
TDS	total dissolved solids
TEDE	total effective dose equivalent
TGB	Turbine Generator Building
TGS	turbine generator system
TGSS	turbine gland sealing system
THD	total harmonic distortion
TIHA	treatment of important human actions
TIT	temperature indicating transmitter
TLD	thermoluminescent dosimeter
TLOS	turbine lube oil storage
TMI	Three Mile Island
TMR	triple module redundancy
T <sub>NDT</sub>	nil ductility temperature
TOC	top of concrete
TRS	test response spectrum
TS	technical specifications
TSC	technical support center
TSTF	Technical Specification Task Force
TUF	tubular ultrafiltration
UAT	unit auxiliary transformer
UCRW	uncontrolled control rod assembly withdrawal at power
UCRWS	uncontrolled control rod assembly withdrawal from a subcritical or low power or startup condition
UDC	uniform Doppler coefficient
UHS	ultimate heat sink
UPS	uninterruptible power supply
URD	Utility Requirements Document
URS	uniform response spectrum
URS	upper riser assembly/section
USGS	United States Geological Survey
USI	unresolved safety issue
USM	uniform support motion
UTC	coordinated universal time
UWS	utility water system
V&V	verification and validation
VDU	video display unit

**Table 1.1-1: Acronyms and Abbreviations (Continued)**

<b>Acronym/Abbreviation</b>	<b>Description</b>
VIT	vibration indicating transmitter
VLA	vented lead-acid
VRLA	valve-regulated lead-acid
VRT	voltage regulating transformer
WDT	watchdog timer
WMCR	waste management control room
WSW	wet solid waste
WTB	Waste Treatment Building
ZPA	zero period acceleration

## 1.2 General Plant Description

This section summarizes the plant design and provides a general description of the overall facility. The description includes:

- principal design criteria, operating characteristics, and safety considerations
- engineered safety features (ESFs) and emergency systems
- instrumentation, controls, and electrical systems
- power conversion system
- fuel, fuel handling, and storage systems
- plant cooling water systems
- radioactive waste management systems
- auxiliary systems (e.g., compressed air, non-radioactive drains, water systems)

Each COL Applicant will develop a Final Safety Analysis Report (FSAR) that incorporates by reference the NuScale FSAR. The NuScale FSAR includes COL items that identify where site specific information must be provided. However, in some instances, representative information is necessary to provide context for interface requirements as specified in 10 CFR 52.47(a)(24) and 10 CFR 52.47(a)(25). This representative or conceptual design information (CDI) is outside the scope of the NuScale Power Plant certified design. Where provided, CDI is delineated by double brackets ([[]]). The scope of the certified design and site-specific design is shown in Figure 1.2-2. The basic systems associated with power generation are shown in Figure 1.2-3. Although some components from these systems are physically located in buildings that are CDI, the system itself is not, with the exception of the clouded portion, which identifies the CDI cooling towers and certain circulating water systems.

### 1.2.1 Principal Site Characteristics

Figure 1.2-1 presents a representative conceptual layout of the overall site. The majority of the site buildings are located within the protected area (PA) and surrounded by a double fence and intrusion-detection equipment. The PA is located within the security owner controlled area (SOCA) surrounded by an additional single fence. An administration and training building and a warehouse are shown outside of the SOCA fence.

A NuScale Power Module (NPM) shown in Figure 1.2-6, is a collection of systems, sub-systems, and components that make up a modularized, movable, nuclear steam supply system (NSSS). Each NPM is comprised of a reactor core, a pressurizer, and two steam generators (SGs) integrated within a reactor pressure vessel (RPV) and housed in a compact steel containment vessel (CNV).

The NuScale Power Plant is designed for 1 to 12 NPMs with the associated primary and secondary systems and components necessary to produce power and maintain the facility. This includes main steam systems, turbine generator sets, condensate and feedwater systems, and shared external cooling water systems (Figure 1.2-3), plus module assembly equipment, fuel handling equipment, turbine maintenance equipment, and radioactive waste processing equipment. The net total output for a NuScale Power Plant with 12 operating NPMs is approximately 570 MWe.

The following structures are included in the NuScale certified design (Figure 1.2-1 and Figure 1.2-2):

- 1) Reactor Building (RXB): located above and below grade, houses the following facilities (among others that are not specifically discussed in this section):
  - ultimate heat sink (reactor pool, refuel pool, and spent fuel pool)
  - fuel handling areas
  - remote shutdown station
  - primary systems

Additional details of the RXB are provided in Section 1.2.2.1.

- 2) Control Building (CRB): located above and below grade, adjacent to the RXB, provides space for the following facilities:
  - main control room (MCR): located below grade, houses the equipment, controls, and indications for operation of the NPMs
  - technical support center-located above the MCR, outside the radiological controlled area, provides space to support emergency operations and personnel

Additional details of the CRB are provided in Section 1.2.2.2.

- 3) Radioactive Waste Building (RWB): located above and below grade, provides space for heating ventilating and air conditioning (HVAC) equipment; and radioactive waste treatment and storage equipment. Additional details of the RWB are provided in Section 1.2.2.3.

The following structures are discussed as CDI (Figure 1.2-1 and Figure 1.2-2):

- 1) Turbine Generator Buildings (TGBs): house the turbine generators and associated equipment. Additional details of the TGBs are provided in Section 1.2.2.5.1.
- 2) Annex Building (ANB): controls access into the radiologically controlled area (RCA) and provides space for health physics facilities, servicing potentially radioactive and non-radioactive tooling, fixtures, and instrumentation, security services, and various personnel services. Additional details of the ANB are provided in Section 1.2.2.5.2.
- 3) Security Buildings (SCBs): provide for controlled access into the SOCA and the PA of the plant. Additional details of the SCBs are provided in Section 1.2.2.5.3.
- 4) Central Utility Building (CUB): houses various equipment for the chilled water system and other ancillary equipment for balance of plant systems. Additional details of the CUB are provided in Section 1.2.2.5.4.
- 5) [[Diesel Generator Buildings (DGBs): house the backup diesel generators and associated equipment.]] Additional details of the DGBs are provided in Section 1.2.2.5.5.

### 1.2.1.1 Facility Description

#### Process Overview

The reactor core is located in a core support assembly, which is seated in the lower RPV assembly. A central hot leg riser is connected to the top of the core support assembly. The reactor core transfers heat into the reactor coolant and the heated reactor coolant flows upward through the core and lower and upper riser assemblies. The heated coolant exits the upper riser assembly and is redirected downwards into the SG region between the vessel wall and the upper riser assembly. As the reactor coolant transfers heat to the SGs, it cools and becomes denser, which drives the natural circulation flow. The coolant returns to the bottom of the vessel through the downcomer and back into the reactor core, where the cycle begins again (Figure 1.2-7).

On the secondary side, preheated feedwater is pumped into the tube side of the SGs where it boils. As the steam flows upward in the tubes, it is continually heated to produce superheated steam before exiting the top of the SGs.

The superheated steam is directed to a dedicated steam turbine. A generator, driven by the turbine, creates electric power that is delivered to the utility grid through a step-up transformer. A turbine bypass line provides up to 100% of the rated main steam flow directly from the associated steam generators to the main condenser in a controlled manner to remove heat from the reactor following a load reduction or loss of electrical load.

Steam that exits or bypasses the turbine is directed to the condenser. A shared circulating water loop removes heat and condenses the steam for up to 6 condensers. The condensate is pumped through condensate polishing equipment to the inlet of the variable speed feedwater pumps. A small amount of steam is extracted from turbine stages to preheat the feedwater and increase plant efficiency. Feedwater regulating valves control feed flow into the SGs.

[[Heat from the circulating water loop from up to 6 condensers is rejected to atmosphere by a set of evaporative mechanical-draft cooling towers. Two sets of cooling towers are provided for 12 NPMs.]]

#### 1.2.1.1.1 Principal Design Criteria

The design provides a simple, safe reactor and provides the following:

- reliable, passive safety systems that are simple in design and operation, and are not reliant on electrical power to fulfill their safety functions
- safety features that assure a core damage frequency significantly lower than the current light water reactor fleet
- the absence of RPV or containment penetrations below the top of the reactor core
- modularization to enable in-shop fabrication of reactor and containment components

### 1.2.1.1.2 Operating Characteristics

The NPM is designed to operate up to full power conditions using natural circulation as the means of providing reactor coolant flow, eliminating the need for reactor coolant pumps.

The NPMs are partially immersed in a reactor pool and protected by passive safety systems. Each NPM has a dedicated emergency core cooling system (ECCS) and decay heat removal system (DHRS).

Important features of the NPM include the following:

- a small, modular design
- an integral pressurized water reactor (PWR) NSSS that combines the reactor core, SGs, and pressurizer within the RPV, eliminating the need for external piping to connect the SGs and pressurizer to the RPV
- natural circulation provides the driving force for reactor coolant flow, eliminating the need for reactor coolant pumps
- an RPV housed in a steel containment partially immersed in water, providing an effective passive heat sink for long-term decay heat removal
- a steel containment operated at a vacuum, eliminating the need for insulation on the RPV
- passive safety systems that are not reliant on electrical power

Table 1.2-1 presents the overall characteristics of the NuScale Power Plant.

### Nuclear Steam Supply System

The NSSS consists of a reactor core, two helical-coil SGs, and a pressurizer integrated within the RPV. The RPV is enclosed in an approximately cylindrical CNV that sits in the reactor pool. The reactor core is located below the helical-coil SGs inside the RPV. Using natural circulation, the primary reactor coolant flow path is upward through the central hot leg riser, and then downward around the outside of the SG tubes with return flow to the bottom of the core via an annular downcomer. As the reactor coolant flows across the SG tubes, heat is transferred to the secondary side fluid inside the SG tubes. Concurrently, as the secondary side fluid progresses up through the inside of the SG tubes, it is heated, boiled, and superheated to produce high pressure steam for the turbine generator unit.

### Reactor Core

The core configuration for the NPM consists of 37 fuel assemblies and 16 control rod assemblies (CRAs). The CRAs are organized into two banks: a regulating bank and a shutdown bank. The regulating bank, consisting of four CRAs symmetrically located in the core, is used during normal plant operation to control reactivity. The shutdown bank (12 CRAs in three groups) is used during normal shutdown. All 16 CRAs are inserted for scram events.



The fuel assembly design is similar to a standard 17 x 17 PWR fuel assembly with 24 guide tube locations for control rods and a central instrument tube. The only significant differences are the fuel assembly is nominally half the height of a standard fuel assembly and is supported by five spacer grids. The fuel is uranium dioxide,  $\text{UO}_2$ , with gadolinium oxide,  $\text{Gd}_2\text{O}_3$ , as a burnable absorber homogeneously mixed within the fuel in select rod locations. The U-235 enrichment is less than 4.95%. A list of fuel design parameters is presented in Table 4.2-1.

### **Pressurizer**

The pressurizer provides the primary means for controlling reactor coolant system (RCS) pressure. It is designed to maintain a stable reactor coolant pressure during operation. Reactor coolant pressure is increased by applying power to a pair of heater bundles installed above the pressurizer baffle plate. Pressure in the RCS is reduced using spray provided by the chemical and volume control system (CVCS).

### **Steam Generator**

Each NPM uses two once-through, helical-coil SGs for steam production. The SGs are located in the annular space between the hot leg riser and the RPV inside diameter wall. The SG consists of tubes connected to feed and steam plenums with tube sheets. Preheated feedwater enters the lower feed plenum through nozzles on the RPV. As feedwater flows through the interior of the SG tubes, heat is transferred across the SG tube wall from the reactor coolant to the feedwater. The feedwater changes phase and exits the SG as superheated steam.

### **Reactor Pressure Vessel**

The RPV consists of an approximately cylindrical steel vessel with an inside diameter of approximately 9 ft and an overall height of approximately 58 ft that is designed for an operating pressure of approximately 1,850 psia. The upper and lower heads are torispherical, and the lower portion of the vessel has a flange to provide access for refueling.

The RPV consists of three sections: the RPV head section, the upper section, and the lower section. The RPV head is welded to the top of the upper section, and the upper and lower sections are flanged together using bolts.

The torispherical RPV head supports the control rod drive mechanisms (CRDMs) and includes penetrations ranging from 2" to 8" diameter for pressurizer spray, reactor vent valves, reactor safety valves, reactor high point degasification instrumentation and controls (I&C) instrument channels, and the CRDM nozzles.

The RPV upper section is cylindrical, approximately 9 ft in diameter with slightly thicker sections at the feedwater inlet and steam outlet areas. The upper section includes penetrations ranging from 2.25" to 25" diameter for main steam piping nozzles and main steam access ports, pressurizer heaters, feedwater piping nozzles

and feedwater access ports, reactor recirculation valves, CVCS, and pressure instrumentation.

The RPV lower section is cylindrical, approximately 9 ft in diameter and includes a torispherical lower head that is welded in place. There are no penetrations in the lower section of the RPV.

A steel pressurizer baffle plate integral with the RPV provides a barrier between the saturated water in the pressurizer and the RCS. The pressurizer baffle plate is integrated with the upper steam plenums, has flow holes to allow surges of water into and out of the pressurizer, and to act as a thermal barrier.

### **Containment Vessel**

The CNV is a cylindrical, steel pressure vessel housing the RPV, CRDMs, and associated NSSS piping and components. The CNV has an overall height of approximately 76 ft and an outside diameter of approximately 15 ft and consists of an upper CNV section with a welded torispherical top head and a lower CNV section with a welded head. The upper and lower CNV sections are flanged together using bolts. The flange connection permits the CNV to be separated to provide access to the RPV for refueling and maintenance.

The safety functions of the CNV are to contain the release of radioactive material following postulated accidents and to provide heat rejection to the reactor pool following ECCS actuation. The CNV also provides support for the RPV.

Manways provide access to components located inside the CNV. Penetrations on the CNV upper head are provided for process piping, electrical power, and instrumentation.

The CNV is supported laterally by support lugs located slightly below the steam plenum elevation and by the support skirt attached to the CNV lower head. The support skirt also provides vertical support for the CNV. Internal to the CNV, the RPV is laterally and vertically supported by four support plates located slightly below the steam plenum elevation and is laterally supported at the center of the lower RPV head.

The CNV is partially immersed in the reactor pool, which provides a passive heat sink for containment heat removal. The CNV is designed to withstand the external environment of the reactor pool as well as the internal pressure and temperature of a design-basis accident.

The CNV is maintained at a vacuum under normal operating conditions. The benefits of maintaining a vacuum in the CNV include:

- minimizes moisture content that could impact the reliability and contribute to corrosion of components within the CNV
- facilitates detection of leakage from the reactor coolant pressure boundary

- eliminates convective heat transfer and therefore, the need for RPV insulation, which reduces potential debris generated in the CNV
- limits the initial amount of oxygen in containment (severe accident combustible gas consideration)

Following an actuation of the ECCS, steam is vented from the RPV through the reactor vent valves. This results in an initial spike in containment pressure and temperature. Steam in contact with the inside surface of the CNV is passively cooled and condensed by conduction and convection to the reactor pool water. This passive process rapidly reduces containment pressure and temperature and maintains containment pressure and temperature at less than design conditions indefinitely.

#### **1.2.1.1.3 Safety Considerations**

NuScale has achieved an improvement in safety over existing plants through simplicity of design, reliance on passive safety systems, and small fuel inventory. The integral design of the NPM eliminates external coolant loop piping, which eliminates large-break LOCA scenarios. The availability of passive safety systems for decay heat removal, emergency core cooling, and control room habitability eliminates the need for external power under accident conditions. With these passive safety systems, small-break LOCAs do not challenge the safety of the plant. The result is a design with a core damage frequency that is lower than the current light water reactor fleet.

The reactor core has a small radioactive source term as compared to a conventional 1,000 MWe nuclear reactor. Based on the smaller fuel inventory, the amount of radioactive material available for release during a postulated accident is reduced. Table 1.2-2 provides a listing of some of the features of the NPM.

#### **1.2.1.2 Engineered Safety Features and Emergency Systems**

##### **1.2.1.2.1 Engineered Safety Feature Materials**

Details are provided in Section 6.1 related to the selection and fabrication methods for metallic and organic materials used in ESF components to ensure compatibility with fluids that the component may be exposed to during normal, accident, maintenance, and testing conditions.

##### **1.2.1.2.2 Containment Systems**

The containment is an integral part of the NPM and provides primary containment for the RCS. Section 6.2 provides further information for the containment system.

##### **1.2.1.2.3 Emergency Core Cooling System**

The ECCS provides a passive means of decay heat removal in the event of a LOCA. The ECCS consists of three independent reactor vent valves and two independent

reactor recirculation valves (Figure 1.2-9). All five valves are closed during normal operation.

During ECCS operation, the reactor vent valves vent steam from the RPV into the CNV where the steam condenses and collects in the bottom of the containment. The reactor recirculation valves allow water to reenter the RPV and be circulated through the core. When reactor coolant temperature is reduced to below the boiling point, core cooling continues via conduction directly into the reactor pool. The cooling function of the ECCS is entirely passive, with heat being conducted through the CNV wall to the reactor pool. Section 6.3 provides design and operational information for the ECCS.

#### **1.2.1.2.4 Control Room Habitability System**

The control room habitability system (CRHS) ensures that plant operators are adequately protected against the effects of accidental releases of toxic and/or radioactive gases. The CRHS is a passive system that provides clean, compressed, breathable air to the MCR in the event of a radioactive release or when AC power is not available. Areas served by the CRHS are maintained at positive pressure relative to adjacent areas. Compressed breathable air storage capacity can provide clean air to the MCR spaces for at least 72 hours following an initiating event. Section 6.4 provides design and operational information for the CRHS.

#### **1.2.1.2.5 Fission Product Removal and Control Systems**

The only fission product removal and control system credited in the design is the CNV in conjunction with the containment isolation system. Fission product control is inherent in the design of the NPM, wherein the CNV atmosphere is depleted through the passive process of aerosol deposition. Section 6.5 provides information for this ESF.

#### **1.2.1.2.6 In-service Inspection of Class 2 and 3 Components**

The in-service inspection program includes the pre-service examinations and the periodic inservice inspections and tests necessary to ensure that safety-related and risk-significant systems, structures, and components are capable of fulfilling their intended safety functions. Section 6.6 provides detailed information for the in-service inspection program.

#### **1.2.1.3 Instrumentation, Controls, and Electrical Systems**

The I&C architectural design philosophy incorporates clear interconnection interfaces, separation between safety and non-safety systems, and simplification of system functions. The I&C architecture primarily consists of the following systems, which are described in Section 7.0:

- module protection system (MPS) provides information from safety-related sensors monitoring temperature, flow, neutron flux, and pressure data on the NSSS.
- neutron monitoring system measures neutron flux as an indication of core power and provides safety inputs to the MPS.

- module control system (MCS) is a distributed control system that allows monitoring and control of module-specific plant components.
- plant control system supplies non-safety inputs to the human system interfaces in the MCR, the remote shutdown station, and other locations where necessary.
- fixed area radiation monitoring system continuously monitors in-plant radiation and airborne radioactivity levels.
- safety display and indication system provides visual display and indication in the MCR from the MPS and plant protection system.
- plant protection system monitors and controls systems that are common to all NPMs and are not specific to an individual NPM.
- health physics network provides the permanently installed communications infrastructure necessary to support a licensee-implemented radiation protection program.
- in-core instrumentation system monitors various parameters within the reactor core and RCS and sends the parameter values to the MCS for display and evaluation.

Under normal operating conditions the AC electrical power distribution system supplies continuous power to equipment required for startup, normal operation, and shutdown of the plant. The NuScale Power Plant does not require onsite or offsite AC electrical power to cope with design-basis events. Safety systems are not reliant on AC or DC electrical power for actuation.

The power systems within the plant are described below:

- The 13.8 KV and switchyard system provides power from the turbine generators and the auxiliary AC power source to the 13.8 kV AC buses and connects the onsite AC system to the switchyard.
- Medium voltage AC electrical distribution system provides power at 4,160V AC to buses servicing medium voltage loads.
- Low voltage AC electrical distribution system provides power at 120V AC and 480V AC to buses servicing low voltage loads.
- Highly reliable DC power system provides a failure-tolerant source of 125V DC power to plant loads including emergency lighting, MPS, PPS, and post-accident monitoring loads.
- Normal DC Power System provides power to non-safety control and instrumentation loads.
- Backup power is provided for onsite AC power. The backup diesel generators provide power at the 480VAC level and the auxiliary AC power source provides power at the 13.8kVAC level.

#### 1.2.1.4 Power Conversion System

The power conversion systems associated with an NPM consist of a main steam system, a turbine generator set, a standard condenser and cooling tower arrangement, and a condensate and feedwater system as shown in Figure 1.2-3.

With multiple NPMs per plant, individual NPMs can be placed into service incrementally to meet construction schedules and grid demand as permitted by the site license. NPMs can also be taken off-line individually for refueling outages and maintenance.

#### 1.2.1.5 Fuel Handling and Storage Systems

The fuel handling and reactor maintenance areas are located in the west end of the RXB and include space for the SFP, refueling pool, and dry dock. The pools are shown in Figure 1.2-16.

#### 1.2.1.6 Plant Cooling Water Systems

The plant cooling water systems include several systems that are important to supporting plant operation. These systems include the following:

- The reactor component cooling water system (RCCWS) is a nonsafety-related, closed-loop cooling system that transfers heat from various plant components to the site cooling water system. The RCCWS provides cooling to the CRDMs, the non-regenerative heat exchangers for each CVCS, and the primary sampling system coolers. (Section 9.2.2)
- The reactor pool cooling system and the spent fuel pool cooling system are nonsafety-related, closed-loop systems that transfer heat from the associated pool to the site cooling water system. (Section 9.1.3)
- The circulating water system is an open-loop system that provides a continuous supply of cooling water to the plant turbine condensers. Circulating water pumps draw water from a common basin to provide cooling water flow for up to six condensers in one TGB. Heated circulating water from the outlet of the condensers flows to a set of mechanical-draft cooling towers where excess heat is removed as the water gravity flows back to the common basin. (Section 10.4.5)
- The site cooling water system is an open-loop system that provides a continuous supply of cooling water to the chilled water system, the balance of plant component cooling water system, the spent fuel pool cooling system, the reactor pool cooling system, the RCCWS, and the condenser air removal system. Site cooling water pumps draw water from a common basin to provide cooling water flow to the systems serviced. Heated site cooling water from the outlet of the individual system heat exchangers continues to a dedicated set of mechanical-draft cooling towers where excess heat is removed as the water gravity flows back to the common basin. (Section 9.2.7)

### 1.2.1.7 Radioactive Waste Management System

The radioactive waste management system is discussed in detail in Chapter 11. Liquid, gaseous, and solid radioactive waste management systems are discussed in detail in Sections 11.2, 11.3, and 11.4, respectively. Process effluent radiation monitoring and sampling systems are discussed in Section 11.5.

## 1.2.2 General Arrangement of Major Structures and Equipment

Figure 1.2-4 presents the layout of a NuScale Power Plant. This figure includes an administration and training building and a warehouse that are outside the scope of the FSAR and not discussed further.

### 1.2.2.1 Reactor Building

As shown in Figure 1.2-4, the RXB is approximately central to the site. See Figure 1.2-5 and Figure 1.2-10 through Figure 1.2-20 for RXB drawings. Dimensions provided in Figure 1.2-5 are nominal or approximate values for illustrative purposes. The RXB houses the NPMs and systems and components required for plant operation and shutdown. The RXB is primarily a rectangular configuration that is approximately 350 ft long and 150 ft wide, and extends approximately 81 ft above nominal plant grade level. The bottom of the RXB foundation is 86 ft below grade except for the areas under the elevator pit and the refueling pool, which are approximately 92 ft below grade. The RXB is a Seismic Category I, reinforced concrete structure with design considerations for the effects of aircraft impact, environmental conditions, postulated design basis accidents (internal and external), and design basis threats. The RXB also provides radiation protection to plant operations and maintenance personnel.

Each NPM is located in the common reactor pool in its own three-walled bay with the open wall towards the center of the pool. The bays are arranged into two rows with six bays per row along the north and south walls of the reactor pool at the east end of the pool. A central channel is provided between the bays to allow for movement of the NPMs between the bays and the refueling pool. The bays are approximately 20 ft wide by 20 ft long by 98 ft deep with a normal reactor pool water depth of approximately 69 ft (this correlates to an elevation of approximately 94'). Each bay has a concrete bioshield to reduce radiation levels in the RXB and to prevent deposition of foreign materials onto an NPM. The bioshield consists of a two foot thick horizontal slab comprised of reinforced concrete and polyurethane with a stainless steel surface and steel vertical faceplate that extends into the pool. The horizontal slab is bolted to the top of the bay. The bioshields are designed to be removed to access the NPM. To accommodate the removed bioshield, each bioshield is designed to have another bioshield stacked on top of it to allow for NPM movement during refueling.

The NPM, reactor pool, and SFP are below grade. The surface of the reactor pool water is approximately 6 feet below grade. Also located below grade are most primary systems and some radioactive waste equipment. Hoisting and handling equipment is located above grade.

Pipe fittings and electrical connections are provided above the reactor pool water level to permit manual connection and disconnection during NPM installation, refueling outages, and during replacement or removal of NPMs.

There is no safety-related equipment on the 125'-0" elevation. With the exception of demineralized water isolation valves, which are located on the 50'-0" elevation, there is no safety-related equipment below the 75'-0" elevation. Table 3.2-1, Classification of Structures, Systems, and Components, provides the location and classification of systems, structures, and components.

#### **1.2.2.1.1 Fuel Handling and Reactor Maintenance Areas**

The fuel handling and reactor maintenance areas are located in the west end of the RXB and include space for the SFP, refueling pool, and dry dock. The pools are shown in Figure 1.2-16.

The operating areas at the west end, 100'-0" elevation of the RXB provide space for the operation of fuel handling equipment and accessing the upper portion of an NPM while the reactor core is being refueled.

The refueling pool is connected directly to the reactor pool accommodating transport of an NPM through the pool water using the Reactor Building crane (RBC). A weir between the refueling pool and SFP provides access for fuel assembly transport under water during the refueling process. The fuel handling and maintenance areas are designed to provide radiation protection for plant operations and maintenance personnel who are working in those areas.

The area west of the SFP contains a fuel receiving area and a jib crane for loading new fuel assemblies into the new fuel elevator. The area has pallet jack access to aid in new fuel receiving activities. Upon receipt, new fuel assemblies are inspected and temporarily stored in racks in the SFP before being placed in a reactor core.

The SFP provides storage space for the accumulated spent fuel assemblies prior to removal for dry storage and for temporary short-term storage for new fuel assemblies. Spent fuel assemblies removed from the reactor core are placed in spent fuel storage racks in the SFP.

The refueling pool contains the bolting tools to disassemble and reassemble the NPM during refueling. The reactor core remains in the lower head of the RPV while in the refueling pool for refueling and fuel management. A fuel handling machine moves new and used fuel through the weir between the refueling pool and SFP.

The dry dock area contains the module inspection rack and is separated from the refueling pool by a gate. With the gate closed, the dry dock water level can be lowered and maintenance activities on the upper NPM can be completed. Necessary inspection and testing equipment for the NPM are moved to this area during refueling.

The dry dock provides maintenance access to the upper section of the NPM. The dry dock is also used for placing new NPM components into the reactor pool and



preparing them for assembly. Additionally, it provides access for shipment of used NPMs off-site.

#### 1.2.2.1.2 Refueling Operations

Refueling operations for an individual NPM is independent of the operating status of the remaining NPMs.

During refueling, an NPM is moved from its operating bay in the reactor pool to the refueling pool using the RBC. The RBC lifts the NPM off its supports within the reactor bay and moves it to the open channel in the center of the reactor pool, which serves as a pathway to transport the NPM to the refueling area.

In the refueling area, the NPM is set into the containment flange tool where the CNV flange is unbolted. The crane lifts the NPM, separating the lower CNV from the upper CNV with RPV still attached and intact. Next, the crane moves the upper CNV and RPV to the reactor vessel flange tool where the RPV flange is unbolted. The crane again lifts the NPM, this time separating the upper and lower RPV, leaving the lower RPV including the reactor core, in the reactor vessel flange tool. Finally, the crane transports the upper NPM (now consisting of just the upper CNV with attached upper RPV) to the module inspection rack in the dry dock. Inspection, testing, and maintenance are performed while the core is being refueled using a dedicated fuel handling machine.

After inspection, maintenance, and testing are complete and the reactor core has been refueled, the upper portion of the NPM is moved from the dry dock to the refueling pool where the NPM is reassembled in reverse order using the dedicated flange tools. Following reassembly, the NPM is moved into the reactor pool and returned to its operating bay by the RBC. In the operating bay, startup tests are performed and the reactor is prepared for restart. After the NPM has passed necessary tests and inspections, and the reactor coolant is at startup conditions, the NPM is brought online, and steam and power production begins.

#### 1.2.2.2 Control Building

The CRB is located approximately 30 ft east of the RXB. See Figure 1.2-21 through Figure 1.2-27 for CRB drawings. The overall CRB footprint is rectangular, approximately 120 ft long by 80 ft wide at the 100'-0" elevation.

The following portions of the CRB are nonsafety-related and Seismic Category II:

- above the 120'-0" elevation
- inside the elevator shaft (full building height)
- inside the two stairwells (full building height)
- the fire protection vestibule located on the East side of the CRB

Structural steel and metal siding are used above the 120'-0" elevation. The remaining portion of the CRB, below the 120'-0" elevation, is a safety-related, Seismic Category I, concrete structure.

The lowest elevation of the CRB primarily houses electrical equipment and CRHS air bottles. There is a tunnel that connects the RXB to the CRB. However, the tunnel is for electrical equipment rather than personnel travel between the two buildings.

The MCR and the associated spaces are located below grade in the CRB. This is the area serviced by the CRHS. Associated spaces for the MCR include the following:

- conference room (shift turnover)
- open office area (auxiliary operator room)
- two offices
- storage room
- janitor closet
- three air locks
- viewing area
- shift manager's office
- reference room
- emergency equipment room
- lavatories
- break room
- telecommunication room

A tunnel allows for personnel travel between the CRB and RXB.

The technical support center (TSC) and the associated spaces are located at grade level in the CRB. Associated spaces for the TSC include:

- records storage
- three offices
- two conference rooms
- data equipment room
- lavatories
- data maintenance room
- break room

Additional equipment located in the CRB includes the control room HVAC system (CRVS) equipment, the chilled water system equipment supporting the CRVS, and an elevator machine room.

### 1.2.2.2.1 Main Control Room

The MCR contains control panels for all installed NPMs. Each reactor operator monitors and controls multiple NPMs from a control room panel. Figure 18.7-1 provides the layout for the MCR.

Digital control systems are implemented in a manner that provides independence between safety-related protection systems and nonsafety-related control systems. Each reactor control system display provides the monitoring for a specific reactor. Additional display stations, including a separate display for shared plant systems, provide control room operators with access to a wide range of plant information for trending and diagnostics.

The reactor operators monitor the automated control system for each NPM. The MCR contains all alarms, displays, and controls for effective monitoring and control by the operators. The control room supervisor station provides an overview of all NPMs using multiple monitors. All monitor displays are designed using human factors analysis to enhance simplicity. The display layout and design uses graphical representations of plant systems and components.

The following monitoring and control activities are typical control room functions:

- initiate NPM startup
- initiate NPM shutdown
- set or correct selected set points that control the NPM or plant functions
- take corrective actions if an NPM or plant system does not operate as intended

The MCR enhances supervisory control of the NPMs and plant systems by providing alarm annunciation on the plant group-view overview display monitor as part of the alarm management system. This system includes information from the individual NPMs via the MPS, the MCS, and the shared I&C systems common to all the NPMs. In the event that the MCR becomes uninhabitable, a remote shutdown station in the Reactor Building provides a secondary location for safe shutdown of the reactors.

### 1.2.2.2.2 Technical Support Center

A TSC is provided, compliant with the design requirements of NUREG-0696. Section 13.3 provides additional information.

### 1.2.2.3 Radioactive Waste Building

The RWB houses equipment and systems for processing radioactive gaseous, liquid, and solid waste and for preparing waste for off-site shipment. See Figure 1.2-28 through Figure 1.2-33 for RWB drawings. The building houses equipment to prepare low-level radioactive waste for compaction to reduce volume and provides temporary storage for radioactive waste. HVAC equipment for high-efficiency particulate air filtration of air from the RXB and RWB is located in the RWB. The building is designed to

maintain radiation exposures to operators and maintenance personnel as low as reasonably achievable.

#### **1.2.2.4 Major Systems**

##### **1.2.2.4.1 Decay Heat Removal System**

The DHRS provides secondary side reactor cooling for non-LOCA events when normal feedwater is not available. The system, as shown in Figure 1.2-8, is a closed-loop, two-phase natural circulation cooling system. Two trains of decay heat removal equipment are provided, one attached to each SG loop. Each train is capable of removing 100 percent of the decay heat load and cooling the RCS. Each train has a passive condenser immersed in the reactor pool. In the event of a SG tube rupture, the affected SG is isolated and the DHRS provides cooling through the intact SG.

On receipt of an actuation signal, feedwater and main steam isolation valves are closed and the DHRS valves open. Reactor coolant continues to circulate through the RPV collecting decay heat from the core. As water from the DHRS condenser travels through the SG tubes it is converted to steam absorbing decay heat from the reactor coolant. The steam then flows back to the DHRS condenser where it gives up excess heat to the reactor pool water and is condensed, and the cycle is repeated. This transfer of heat promotes natural circulation in both the RCS and the DHRS.

Section 5.4.2 provides design and operational information for the DHRS.

##### **1.2.2.4.2 Ultimate Heat Sink**

The ultimate heat sink is a large, stainless steel-lined, reinforced concrete pool located in the RXB below plant grade level. The ultimate heat sink consists of the reactor pool area, the refueling pool area, and the spent fuel pool area. The pool areas are shown in Figure 1.2-16. During normal plant operations, heat is removed from the pool through the reactor pool cooling system and rejected into the atmosphere through a cooling tower or other external heat sink. The spent fuel pool has an independent spent fuel pool cooling system.

In a design basis accident involving a sustained loss of all AC power, decay heat is removed from the NPMs through passive heat transfer to the pool resulting in pool heat up and boiling. Water inventory in the reactor pool is adequate to cool the NPMs for at least 72 hours without adding water.

The reactor pool provides an additional means of fission product retention beyond that of the fuel, fuel cladding, RPV, and the containment for certain events.

Section 9.2.5 provides design and operational information for the ultimate heat sink.

#### 1.2.2.4.3 Chemical and Volume Control System

The CVCS is simple in design and its operation is not credited during or after an accident. During normal operation, the CVCS recirculates a portion of the reactor coolant through demineralizers and filters to maintain reactor coolant cleanliness and chemistry. A portion of the recirculated coolant is used to supply pressurizer spray for controlling reactor pressure. Reactor coolant inventory is controlled by injection of additional water when reactor coolant levels are low or letdown of reactor coolant to the liquid radioactive waste system when coolant inventory is high.

Additionally, during the NPM startup process, the CVCS is used in conjunction with the module heatup system, to add heat to the reactor coolant to establish natural circulation flow in the RCS.

Boron concentration in the RCS is controlled by a feed-and-bleed process. Injection pumps provide borated water or clean demineralized water that is delivered into the RCS with excess reactor coolant being letdown to the radioactive waste system. Safety-related protection is provided for an anticipated operational occurrence involving unintended dilution of the RCS due to CVCS equipment failure or operating error.

Section 9.3.4 provides design and operational information of the CVCS.

#### 1.2.2.5 Other Site Structures

##### 1.2.2.5.1 Turbine Generator Building

A NuScale Power Plant has two separate TGBs. The TGBs are nonsafety-related structures. Each building houses six turbine generator sets along with their auxiliaries, the condensers, condensate systems, and the feedwater systems. A laydown area and overhead crane are provided for installation and maintenance activities in each TGB.

##### 1.2.2.5.2 Annex Building

The ANB is a nonsafety-related structure. The ANB houses several facilities and serves several functions, including:

- controlling access to both radiologically-controlled and nonradiologically-controlled areas of the RXB
- housing various personnel support services such as locker rooms, showers, toilet facilities, lunch and conference rooms, and first aid
- [[providing space for personnel and component decontamination equipment and employee dosimeter processing
- housing a portion of the facilities that support plant security such as secondary alarm station, security briefing room, armory, security manager's office, etc.]]

**1.2.2.5.3 Security Buildings**

The SCBs are non-safety related structures that include the following structures:

- primary access control building
- main security building
- vehicle barrier system

The SCBs provide the following nonsafety-related functions:

- control personnel and vehicle entry into the PA and screen personnel seeking unescorted access into the PA.
- verify identity and access status as well as search for contraband items.
- provide a structure or space to monitor access into areas of the plant as well as monitoring tamper alarm devices.

**1.2.2.5.4 Central Utility Building**

The CUB is a nonsafety-related structure that houses common utility plant services, which include the following:

- [[chiller equipment
- instrument air system
- service air system
- chemical treatment equipment for demineralized water
- maintenance area
- life safety
- demineralized water equipment
- security functions]]

**1.2.2.5.5 [[Diesel Generator Buildings**

The NuScale Power Plant design includes two DGBs, each housing a single backup diesel generator. The principal functions of each DGB are to provide support and housing for the backup diesel generators and their auxiliary equipment. The DGB houses no safety-related systems and has no functional requirements that support the ESFs. The DGBs house the following:

- diesel engines and associated support equipment
- generators
- DGB HVAC system
- maintenance area]]

**1.2.3 Plant Features of Special Interest****Human Factors Considerations**

The NuScale Power Plant design minimizes human error through fail-safe design functionality, allows multi-modular control capability from a single control room with effective automation design, employs digital display design and soft control technology to enhance usability, and provides optimum workload management.

The HFE program satisfies specific regulatory requirements and guidance, and leverages human performance and operating experience from nuclear and non-nuclear industries.

Chapter 18 describes the HFE Program.

**Table 1.2-1: Overall Characteristics of a NuScale Power Plant**

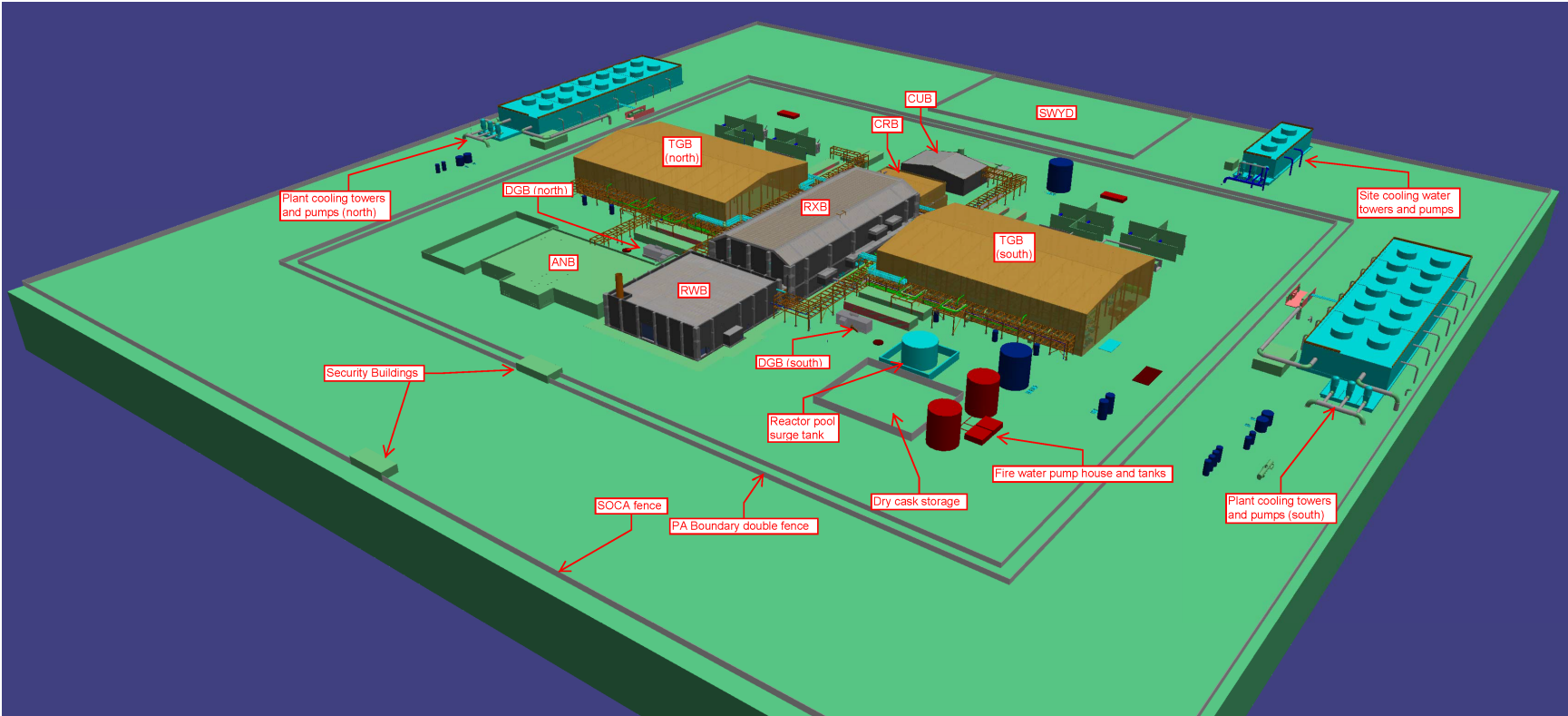
Overall Plant	
Nominal net output	570 MWe*
Number of power units	12
Power Module	
Number of reactors	One
Thermal power rating	160 MWth
Nominal gross electrical output	50 MWe
RCS normal operating pressure	1,850 psia
Steam generator number	Two
Steam generator type	Vertical helical tube
Steam cycle	Rankine-subcritical regenerative with superheat
Turbine type	3,600 rpm, condensing, with extraction
Reactor Core	
Fuel	UO <sub>2</sub> (<4.95% enrichment)
Refueling intervals	24 months
* Nominal net output is total gross electrical output minus house loads.	



Table 1.2-2: Design Features of a NuScale Power Module

NuScale Design Feature	Primary Impact	Safety Enhancement
RCS contained within the RPV	No large diameter primary coolant piping	Eliminates postulated large-break LOCA spectrum accidents
Natural-convection-cooled core	No reactor coolant pumps	Eliminates reactor coolant pump accidents, shaft breaks, pump seizure, missile generation and pump leaks
High containment design pressure	Containment peak pressure for worst case design-basis accident remains below containment design pressure	Containment integrity assured, minimizing the potential for radioactive releases during postulated accidents.
RPV and NSSS inside the CNV	During an accident, any water lost from RPV stays within containment and is returned to the RPV by passive means	No postulated design-basis small-break LOCA capable of uncovering nuclear fuel
Evacuated containment	Subatmospheric pressure during normal operation  No insulation on RPV	Minimal amount of noncondensable gases increases the steam condensation rate for containment heat removal during postulated small-break LOCA. Amount of oxygen in containment during normal operations is minimized.  Eliminates potential sump screen blockage and permits cooling of the exterior of the vessel during an accident
Low power core (160 MWt)	Reduces decay heat removal requirements	Enhances in-vessel retention; maintains low accident consequences; reduces fission product source term; simplifies emergency planning
Reactor pool with partially (approximately 90%) immersed NPM	CNV partially immersed in reactor pool	Provides passive long-term cooling
Passive safety systems	Safety systems cool and depressurize the RPV/CNV even in the event of loss of external power	Active safety systems are not required

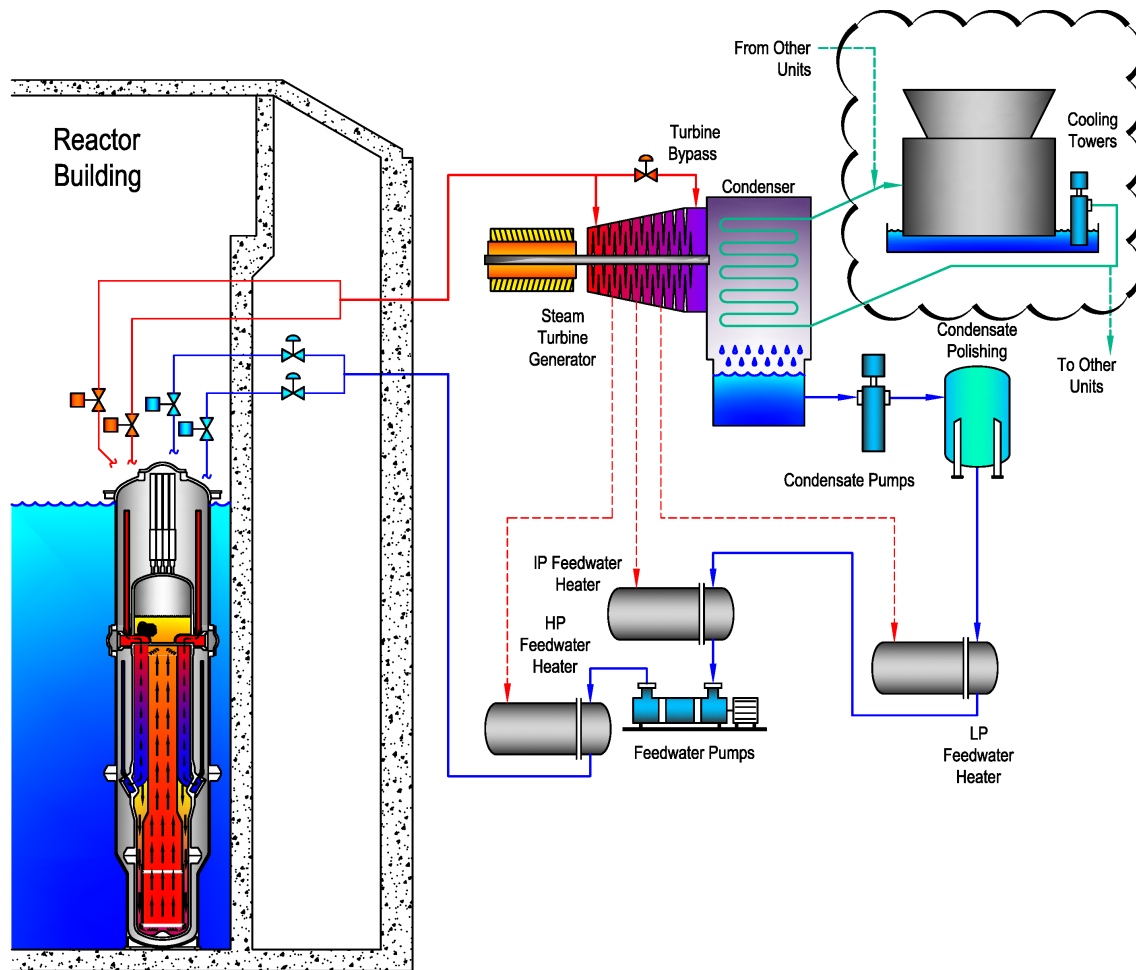
Figure 1.2-1: Conceptual Site Layout



**Figure 1.2-2: NuScale Functional Boundaries**

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Figure 1.2-3: Schematic of a Single NuScale Power Module and Associated Secondary Equipment



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**Figure 1.2-4: Layout of a Multi-Unit NuScale Power Plant**

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**Figure 1.2-5: Cutaway Illustration of 12 Module Configuration**

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Figure 1.2-6: Cutaway View of NuScale Power Module

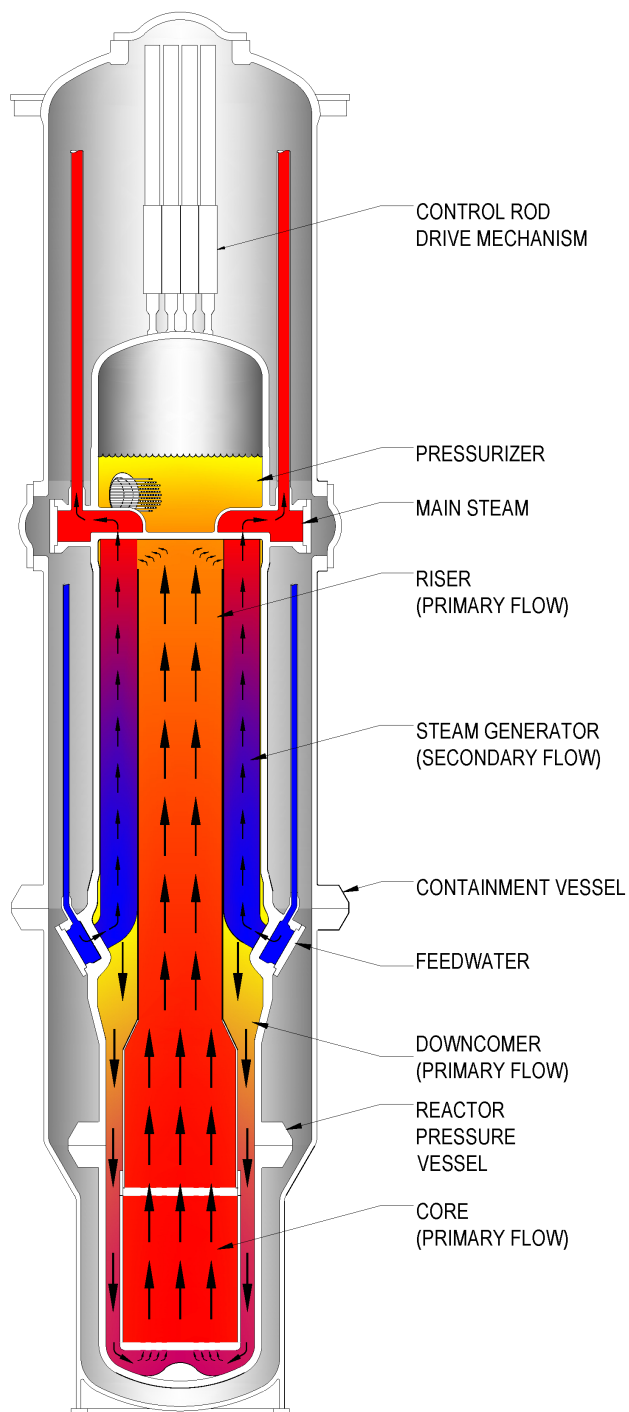


Figure 1.2-7: Steam Generator and Reactor Flow

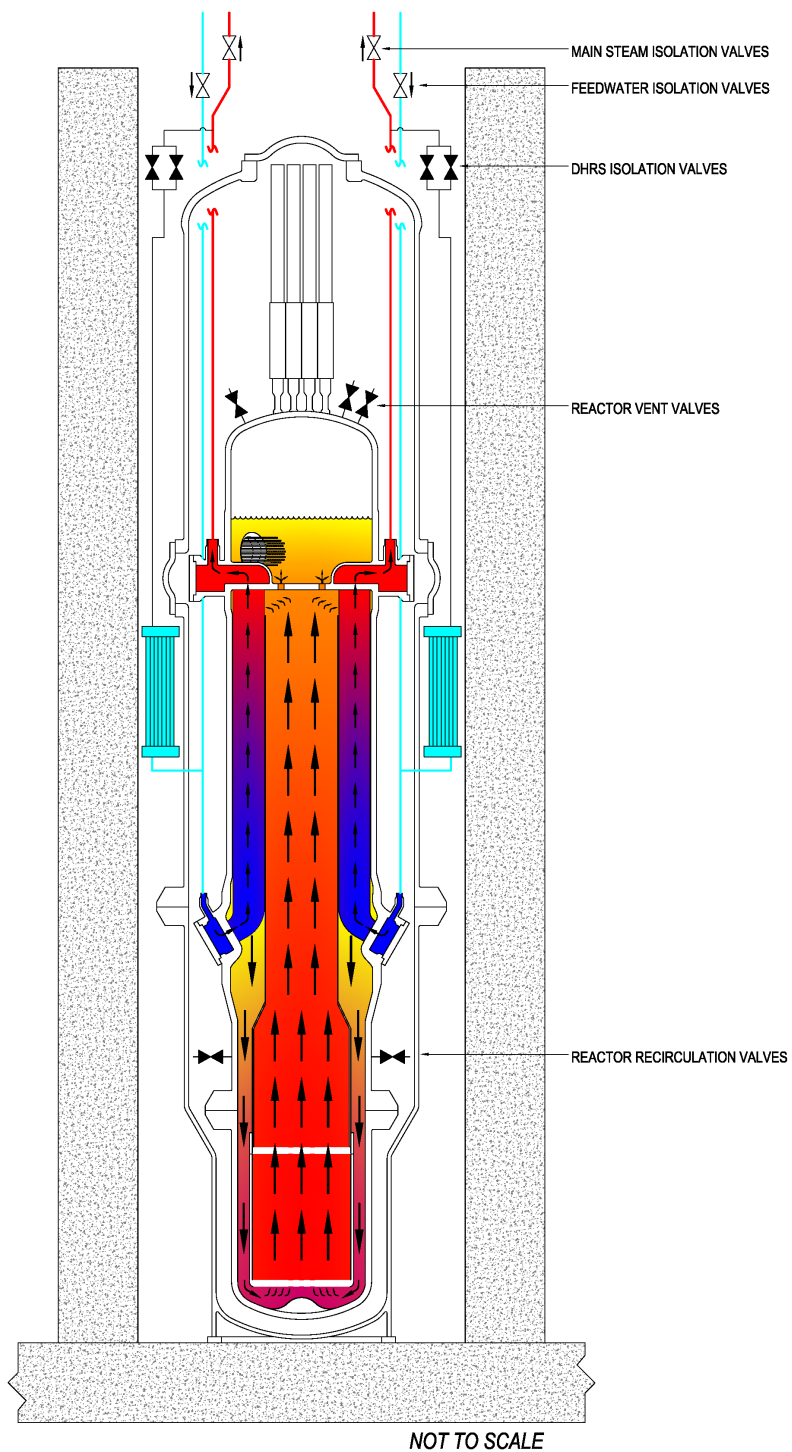
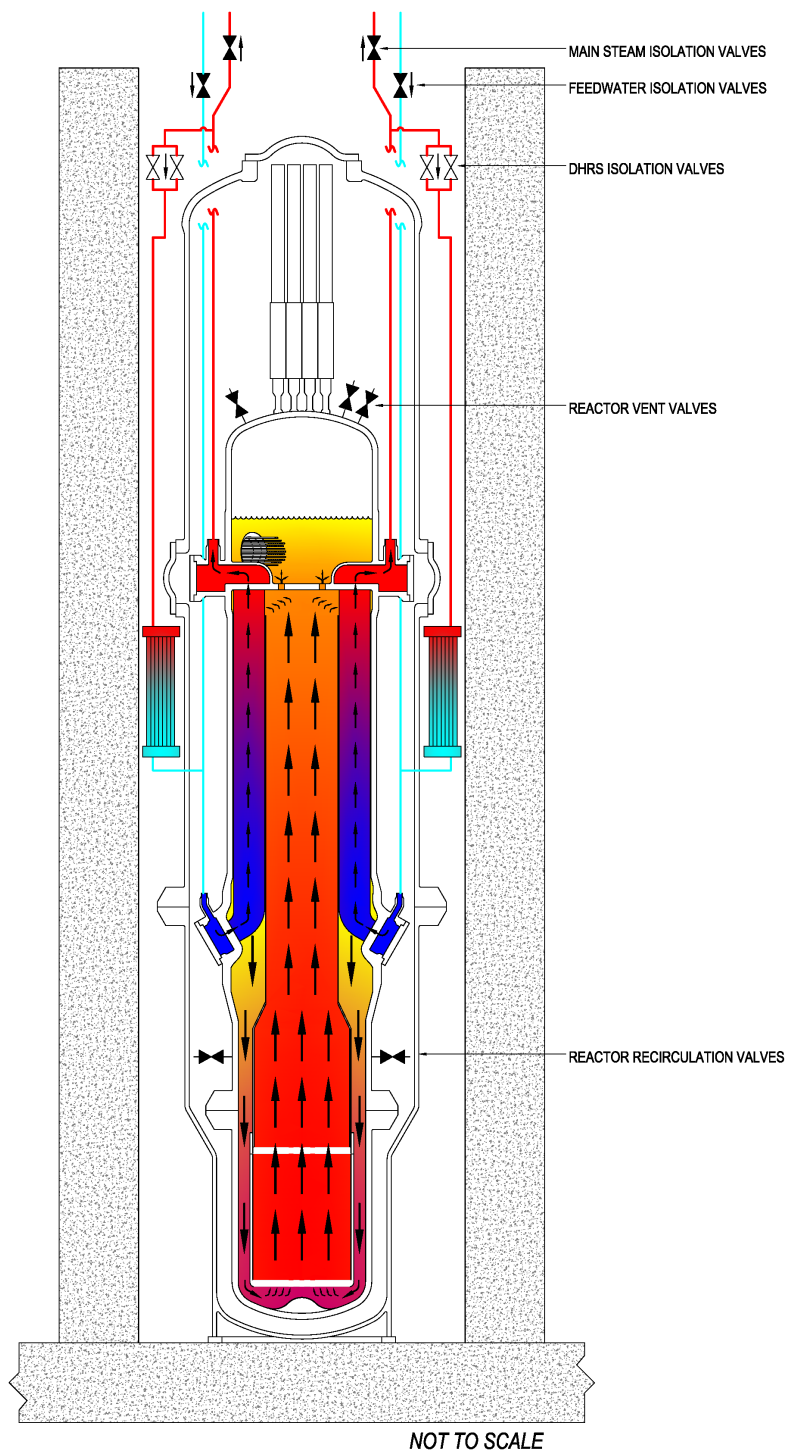




Figure 1.2-8: Decay Heat Removal System



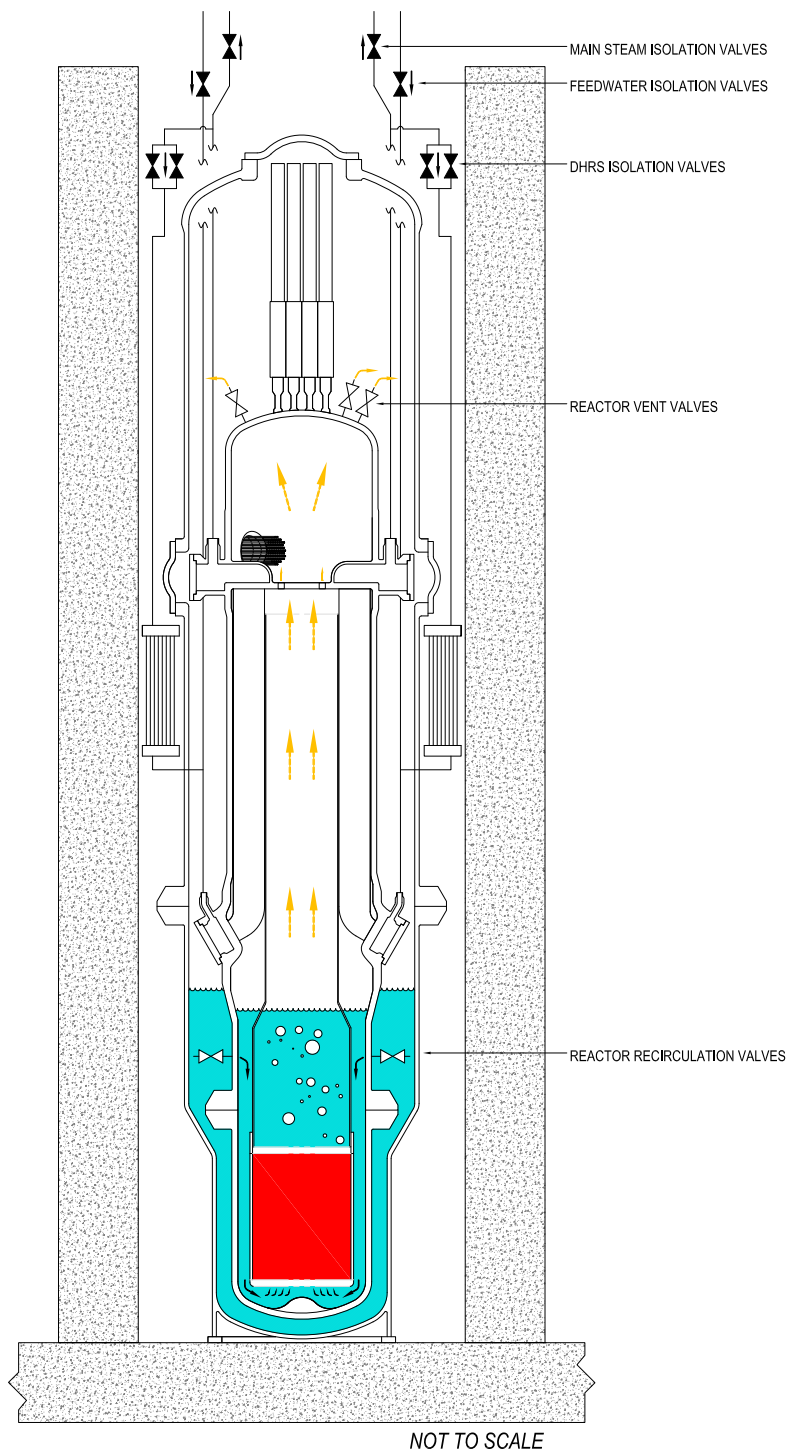
**Figure 1.2-9: Emergency Core Cooling System**

Figure 1.2-10: Reactor Building 24’-0” Elevation

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Figure 1.2-11: Reactor Building 35'-8" Elevation

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Figure 1.2-12: Reactor Building 50'-0" Elevation

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Figure 1.2-13: Reactor Building 62'-0" Elevation

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Figure 1.2-14: Reactor Building 75'-0" Elevation

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Figure 1.2-15: Reactor Building 86'-0" Elevation

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Figure 1.2-16: Reactor Building 100'-0" Elevation

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Figure 1.2-17: Reactor Building 126'-0" Elevation

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Figure 1.2-18: Reactor Building 145'-6" Elevation

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Figure 1.2-19: Reactor Building East and West Section View

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Figure 1.2-20: Reactor Building South Section View

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Figure 1.2-21: Control Building 50'-0" Elevation

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Figure 1.2-22: Control Building 63'-3" Elevation

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Figure 1.2-23: Control Building 76'-6" Elevation

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Figure 1.2-24: Control Building 100'-0" Elevation

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Figure 1.2-25: Control Building 120'-0" Elevation

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Figure 1.2-26: Control Building North Section View

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Figure 1.2-27: Control Building West Section View

Withheld - See Part 9

Figure 1.2-28: Radioactive Waste Building 71'-0" Elevation

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Figure 1.2-29: Radioactive Waste Building 82'-0" Elevation

Withheld - See Part 9

Figure 1.2-30: Radioactive Waste Building 100'-0" Elevation

Withheld - See Part 9

Figure 1.2-31: Radioactive Waste Building 120’-0” Elevation

Withheld - See Part 9



Figure 1.2-32: Radioactive Waste Building North and South Section Views

Withheld - See Part 9

Figure 1.2-33: Radioactive Waste Building West Section View

Withheld - See Part 9

**1.3 Comparison with Other Facilities**

The major NuScale Power Plant design features and nominal parameters are provided in Table 1.3-1 and discussed further in the associated final safety analysis report (FSAR) section(s). These NuScale features and values are shown in comparison with a typical pressurized water reactor (PWR) plant design. All values are nominal and provided for comparison only. The typical PWR values presented are representative of the Standardized Nuclear Unit Power Plant System design.

Table 1.3-2 provides a comparison of safety systems and components required to protect the reactor core for the NuScale Power Plant versus a typical PWR plant.

Table 1.3-1: NuScale Plant Comparison with Other Facilities

NuScale Plant Parameter or Feature (per NPM)	Typical PWR	NuScale
Nominal gross electrical output (MWe)	1,186	50
Core thermal output (MWt)	3,411	160
Number of fuel assemblies	193	37
Fuel assembly lattice	17 x 17	17 x 17
Effective fuel length (ft)	12	6.56
Fuel rods per fuel assembly	264	264
Average linear heat rate (kW/ft)	5.4	2.5
Number of Control Rod Assemblies	53	16
Design life (years)	40	60
Reactor Coolant System		
Number of heat transfer loops	4	No External Loops
Reactor Coolant Pipes (in)	27.5-31	None
Operating pressure (psia)	2,250	1,850
Hot leg temperature (°F)	618	590
Reactor Vessel		
Vessel inner diameter (in)	173	107.5
Thermal shielding/reflector design	Neutron pad design	Stacked stainless steel reflector blocks
In-core instrumentation	Bottom mounted	Top mounted
Steam Generator		
Number	4	2
Type	Vertical U-tube	Helical coil
Heat transfer area (ft <sup>2</sup> )	55,000	Approximately 18,000
Number of tubes	5,626	1,380
Reactor Coolant Pumps	4	0
Pressurizer		
Internal volume (ft <sup>3</sup> )	1,800	568
Surge nozzle nominal diameter (in)	14	None
Residual Heat Removal Pumps	2	None
Containment		
Type	PCCV	Steel Pressure Vessel
Inner diameter (ft-in)	140-0	14-2
Height (ft-in)	205-0 (inner)	75-8.5 (outer)
Containment Spray Pumps	2	None
High Pressure Safety Injection Pumps	2	None
Charging / Safety Injection Pumps	2	None
Low Pressure Safety Injection Pumps	2	None
Accumulators	4	None
I&C System type	Analog	Digital
Emergency Diesel Generators	2	None
Turbine Type	1800 rpm, Tandem Compound Six Flow	3,600 rpm, 10 stage with Superheat
Emergency Feedwater Pumps	3	None
Charging Pumps (CVCS pumps)	2	2
Used for Safety Injection	Yes	No
Volume Control Tank	1	0
Reactor Component Cooling Water Pumps	4	6 total for 12 NPMs

**Table 1.3-2: Safety Systems and Components Required to Protect the Reactor Core -  
NuScale Comparison with Other Facilities**

<b>Safety System or Component</b>	<b>Typical PWR</b>	<b>NuScale</b>
Reactor Pressure Vessel	X	X
Containment Vessel	X	X
Reactor Coolant System	X	X
Decay Heat Removal System	X	X
Emergency Core Cooling System	X	X
Control Rod Drive System	X	X
Containment Isolation System	X	X
Ultimate Heat Sink	X	X
Residual Heat Removal System	X	
Safety Injection System	X	
Refueling Water Storage Tank	X	
Condensate Storage Tank	X	
Auxiliary Feedwater System	X	
Emergency Service Water System	X	
Hydrogen Recombiner or Ignition System	X	
Containment Spray System	X	
Reactor Coolant Pumps	X	
Safety-Related Electrical Distribution System	X	
Alternative Off-Site Power	X	
Emergency Diesel Generators	X	
Safety-Related Class 1E Battery System	X	
Anticipated Transient Without Scram (ATWS) System	X	

**1.4 Identification of Agents and Contractors****1.4.1 Applicant and Program Manager**

NuScale Power, LLC (NuScale) was founded in 2007. The NuScale Power Plant design takes advantage of existing design tools and available nuclear fuel options while leveraging the wealth of knowledge developed through more than 50 years of practical application of light-water-cooled pressurized water reactor (PWR) technology.

**1.4.2 Division of Responsibility**

NuScale has the overall design responsibility for the NuScale certified design including reactor, containment, primary reactor systems and structures (e.g., Reactor Building, Control Building, and Radioactive Waste Building). NuScale maintains headquarters in Portland, Oregon with engineering design offices in Corvallis, Oregon and Charlotte, North Carolina. In addition, NuScale utilizes major testing facilities in the United States, Canada, Italy, France, and Germany.

**1.4.3 Principal Consultants and Other Participants**

Fluor Corporation (Fluor) provides the balance of plant design from its Greenville, South Carolina office. Fluor, a major stakeholder of NuScale, has extensive architectural-engineering experience with over 40 years' experience with commercial nuclear projects, providing operating plant support services to 50 United States and international units at 29 locations. Fluor and its nearly 40,000 employees work with governments and clients in diverse industries around the world to design, construct, and maintain complex and challenging capital projects.

COL Item 1.4-1: A COL applicant that references the NuScale Power Plant design certification will identify the prime agents or contractors for the construction and operation of the nuclear power plant.

## 1.5 Requirements for Additional Technical Information

This section describes the verification and confirmation tests of unique design features of the NuScale Power Plant. The testing program was developed to provide data in support of the final safety analyses, to verify safety analysis models, and to demonstrate the performance of unique design features and components.

### 1.5.1 NuScale Testing Programs

The following testing programs have been completed or are currently in progress. The tests focus on design features of the NuScale Power Module (NPM) design for which applicable data or operational experience did not previously exist. Tests specific to the NuScale fuel design are given in Section 1.5.1.1, Section 1.5.1.2, and Section 1.5.1.10; tests specific to the steam generator (SG) are given in Section 1.5.1.3, Section 1.5.1.4, Section 1.5.1.5, Section 1.5.1.8, and Section 1.5.1.9; tests specific to the control rod assemblies are given in Section 1.5.1.7, Section 1.5.1.11, and Section 1.5.1.12; finally, tests involving integrated system phenomena are given in Section 1.5.1.6.

#### 1.5.1.1 Critical Heat Flux Testing - Preliminary Fuel Design

The NPM employs a fuel design for heat generation that is similar to a standard pressurized water reactor (PWR), with the exception of the fuel assembly height and the reactor coolant driving force. The NuScale fuel is approximately half the height of standard PWR fuel and features low-flow natural circulation of primary coolant rather than pump-driven primary coolant flow. In order to meet fuel licensing requirements, two critical heat flux (CHF) test programs were conducted: (1) a test program described in this section for the preliminary fuel design, and (2) a second test program described in Section 1.5.1.2 for the final fuel design. The preliminary fuel design test program supported code development and safety analysis efforts, and provided data for development of NuScale's NSP-2 CHF correlation for the final fuel design.

The NPM reactor core design employs 37 nuclear fuel assemblies. Each assembly is composed of a 17x17 square lattice of fuel rods assembled according to a given rod-to-rod pitch. Each fuel rod is approximately 2 meters in length. Fuel rods are assembled using spacer grids placed at specified locations along the length of the fuel rods such that fuel rods are evenly spaced and adequately supported. Primary coolant enters the NPM reactor core from the bottom through the core inlet plenum and heat transfer to the coolant occurs as coolant travels upward along the length of the fuel assemblies.

In off-normal conditions, such as anticipated operational occurrences and postulated accidents, it must be known how close the heat transfer mode is to transitioning to a state where a continuous steam layer covers the fuel rods or portions of the fuel rods. The point at which this transition occurs is referred to as the CHF point. In order to determine the CHF point for the reduced-length fuel under appropriate flow conditions, a CHF testing program was conducted over a wide range of operating conditions. In these tests, several sensors and associated instrumentation were used to measure key test parameters, including: thermocouples embedded in heater rods, pressure transducers, mass flow rate instruments, and electrical voltage and current meters. These sensors were used to measure heater rod temperatures and fluid flow conditions at various points of the fluid loop, and the electrical power supplied to

heater rods when CHF occurred. The tests allowed NuScale to obtain fuel bundle subchannel exit temperatures to determine mixing coefficients and to obtain single-phase and two-phase pressure drop characteristics of the test assembly for a range of bundle powers and hydraulic conditions. All information necessary for CHF correlation development and evaluation was collected.

Facilities for the preliminary fuel design CHF test series were provided by Stern Laboratories in Ontario, Canada using their existing high-pressure flow loop, electrical power supplies, heat rejection devices, water conditioning system, and data acquisition system. The test section and fuel simulators were designed and manufactured to represent the preliminary NuScale fuel design. Fuel assembly simulators with uniform and cosine-axial power shapes were manufactured along with two sets of spacer grids that allowed testing of a 5x5 fuel assembly simulation bundle with or without the center fuel rod replaced by a guide tube. The approach of using a 5x5 representation of the larger 17x17 fuel assembly is an industry-accepted practice for CHF testing of PWR fuel bundles. The fuel assembly simulation bundles were mounted within a square flow channel and installed in the instrumented test section. This allowed testing of three separate configurations of fuel assembly simulation bundles:

- uniform-power profile without a guide tube in the center position
- uniform-power profile with a guide tube at the center position
- cosine-power profile without a guide tube in the center position

Data were required to span a range of bounding conditions and axial power shapes for vertical 5x5 fuel assembly configurations in accordance with the test specification and program documentation, which provided detailed test matrices for steady-state and transient CHF testing, pressure drop, and thermal mixing. The vertical 5x5 fuel assembly configurations were tested using industry-accepted test and acceptance methodology.

The CHF testing was conducted by flowing water over the test sections at discrete test points covering a range of hydraulic conditions sufficient to develop a CHF correlation that spanned the NPM operational envelope. At each test point, the loop was configured at the specified flow, inlet temperature, and exit pressure conditions. The bundle power was increased until CHF was detected, which was indicated by an excursion of the fuel simulator surface thermocouples. Loop flow conditions (temperature, pressure, and flow), bundle power, rod power, and fuel simulator temperatures were recorded for each run. As-built data for the test section and test article, such as flow channel width, fuel simulator diameters, and spacer grid dimensions, were also recorded.

In conclusion, tests were performed for a variety of thermal conditions using representative 5x5 fuel assembly simulations with a 2-meter heated length, uniform and symmetrical cosine axial power profiles, with and without a simulated guide tube. The experiments investigated the effects of shorter fuel length and low-flow natural circulation of the primary coolant, and provided data that were used to develop NuScale's NSP-2 CHF correlation in support of the NuScale small modular reactor



technology. This test program was inspected by the NRC in accordance with Inspection Procedures IP 35034, 35017, and 36100.

#### **1.5.1.2 Critical Heat Flux Testing - NuFuel HTP2 Fuel Design**

The primary objective for this test program was to obtain CHF data for the NuScale fuel design that employs AREVA HMP/HTP spacer grid technology (designated as NuFuel HTP2) to augment the existing database that was previously obtained for NuScale's preliminary fuel design (described in Section 1.5.1.1). The new data was used to validate NuScale's NSP-2 CHF correlation developed using the preliminary fuel design tests for the NPM application. In addition, this test allowed NuScale to obtain bundle subchannel exit temperatures to determine mixing coefficients, and to collect single-phase and two-phase pressure drop characteristics of the assembly for a range of bundle powers and hydraulic conditions.

The CHF test employed an electrically-heated test section that consisted of a 5x5 simulated fuel bundle built to prototypic geometry and employing AREVA HTP/HMP grid technology. Like the preliminary fuel tests, fuel assembly simulators with uniform and cosine-axial power shapes were tested using a 5x5 fuel bundle with or without the center fuel rod replaced by a guide tube. The testing was conducted by flowing water through the test section at specified flow rates over a range of hydraulic conditions of the NPM. At each test point, the loop was configured for a specified flow rate, inlet temperature, and exit pressure conditions, and the bundle power was increased until CHF was detected over a range of operating conditions and axial power shapes for vertical 5X5 fuel assembly configurations. The occurrence of CHF was indicated by an excursion of the fuel simulator surface temperatures.

The prototypic fuel design tests were conducted at the AREVA Karlstein Thermal Hydraulics (KATHY) facility located in Karlstein, Germany. The test data was used to validate the applicability of NuScale's NSP-2 CHF correlation for the NuFuel HTP2 fuel design.

#### **1.5.1.3 Steam Generator Tube Inspection Evaluation and Demonstration**

The NuScale helical coil SG is designed to provide access for required inservice inspection and repair. Access to the tube sheet in the NuScale design requires traversal of the steam access flanges through the containment vessel and the SG steam and feed plena.

Due to the unique inspection configuration of the helical SG, a design evaluation and proof-of-concept demonstration by an SG non-destructive examination inspection supplier was conducted to inform the SG design and provide confidence in the ability to inspect the helical SG. Information and experience gained in the proof-of-concept evaluation and demonstration test helped to inform the design development of the SG and associated plena.

Tests were conducted at the Oregon industrial facilities in Albany, OR and included representative, full-scale helical coils in a vertical orientation. Two coils were used that corresponded to a curvature similar to the innermost and outermost column geometries to envelop the largest and smallest transition radii within the SG. Bend radii

were included at the inlet and outlet of each tube to model the geometry of the tubes where they enter the tubesheet. An inspection delivery system was used to insert and retract a probe through the full length of the prototypic tubing. This system consisted of a simulated eddy current probe, probe conduit, and probe delivery system that included probe pusher hardware and software, guide tube, and delivery assist system.

The prototype dummy probe used in this proof-of-concept testing was representative of an actual eddy current inspection probe typical of those routinely used for the inspection of PWR steam generators. An important aspect of the demonstration was to ensure representative flow path bends would not add a limiting amount of friction.

The demonstrations were performed using different eddy current probe shaft materials and multiple probe dummy assemblies to determine a recommended shaft sheathing for the helical SG tubes. Various combinations of available inspection system components were evaluated. Inspection feasibility was successfully demonstrated.

#### **1.5.1.4 Steam Generator Thermal-Hydraulic Performance Testing - Electrically Heated Facility**

The NPM incorporates two collocated SGs housed within the reactor pressure vessel. The SGs provide heat transfer to and from the primary system for both normal and off-normal conditions. Through natural circulation, the reactor coolant system transfers the core power to the SG converting feed water into steam. Unlike current PWR designs, the reactor coolant flows around the outside of the SG tubes (primary side) and the feedwater and main steam flow through the inside of the tubes (secondary side). Because these design aspects of the helical SGs are different from those used in the nuclear fleet, operational experience is not available and large-scale experimental data were needed for validation of NuScale thermal-hydraulic systems and design computer codes, as well as determination of SG performance characteristics.

The objective of this testing was to determine the secondary side (inside tube) thermal-hydraulic performance of individual helical tubes representative of those used in the NPM steam generator design. This required the evaluation of operation over a range of conditions representative of the operational envelope. Measurement data were required to evaluate the distribution of temperature and pressure on the inside of the tubes.

The electrically-heated test focused on secondary-side performance and consisted of three isolated tubes that were instrumented with well-controlled boundary conditions. Heating was accomplished using Joule heating, wherein a known electrical current is passed through the tube walls to produce a constant heat flux boundary condition on the inside of the tubes. Three distinctive heating zones were employed to provide different heat fluxes for the subcooled, boiling, and superheat regions. Within each zone the heat flux was constant, which represents a simplification from the heat flux profile that results when fluid heating is employed, as would occur in an operating NuScale SG. This approach enabled tube wall heat flux to be controlled during testing and permitted better access to instrumentation on the outside of the tubing.

The testing was performed at the Societ Informazioni Esperienze Termoidrauliche (SIET) test facility in Piacenza, Italy. Types of testing carried out included adiabatic

testing, diabatic testing, transient testing, and density wave oscillation testing. In addition, dynamic pressure measurements were recorded during all test runs which supported development of power spectral density spectra that may be used to support evaluation of the potential for internal two-phase (boiling) pressure fluctuations to contribute to flow induced vibration of SG tubes. These data also were used to inform sizing of the SG inlet flow restrictors for stable secondary-side SG operation, provide benchmarking for NuScale thermal-hydraulic design and systems computer codes, and define steam outlet conditions as a function of primary heat generation and secondary side conditions. This test program was inspected by the NRC in accordance with Inspection Procedures IP 35034, 35017, and 36100.

#### **1.5.1.5 Steam Generator Thermal-Hydraulic Performance Testing - Fluid-Heated Facility**

Subsequent to the SG tests described in Section 1.5.1.4 that used three electrically heated SG tubes, a second set of SG tests was conducted using a 252-tube bundle array that was fluid heated on the exterior of the tubes to more accurately represent primary side SG conditions.

The test facility included a large pressure vessel, which was able to accommodate the tube bundle test section and allowed for testing at elevated pressures and temperatures. The test facility included heaters and pumps that provided a span of flow rates at a wide range of thermal-hydraulic conditions. The fluid-heated test focused on overall primary and secondary side performance, and consisted of a bank of 252 helical tubes, modeling five of the 21 helical coil columns, operated at near-prototypic primary- and secondary-flow conditions.

Testing activities were conducted at SIET in Piacenza, Italy using their fluid-heated hydraulic loop. Types of testing carried out included: adiabatic, diabatic, transient, density wave oscillation, and fluid-elastic instability tests. Each type of test consisted of multiple test points covering a range of conditions to characterize the phenomena of interest at various combinations of primary-side and secondary-side pressures, temperatures, and flow rates. In these tests, thermocouples, pressure transducers, mass flow rate instruments, and strain gauges were used to collect temperature, pressure, flow rate, and vibration data at several locations on the primary and secondary sides of the SG. These data have been used to benchmark NuScale thermal-hydraulic design and systems computer codes, and to define steam outlet conditions as a function of primary-fluid heating and secondary-side conditions.

#### **1.5.1.6 NuScale Integral System Test Program**

The purpose of the NuScale integral system test program was to generate thermal-hydraulic data for system characterization and safety code validation using a scaled representation of the NPM design. Tests have also supported the development of NPM operational procedures, and control logic strategies, and to inform safety methodology development.

The NuScale Integral System Test Facility (NIST-1) allows NuScale to replicate the integrated thermal-hydraulic phenomenon occurring in the reactor coolant system, containment, safety systems, and reactor pool. Data collected provide system characterization data required for validation of safety-related software, NRELAP5 and

PIM. The NRELAP5 code, which is based on a commercial version of RELAP5-3D, is a thermal-hydraulic analysis code developed at NuScale for use in the thermal-hydraulic design, safety analysis, and licensing of the NPM. The code incorporates models and correlations specific to the unique design features of the NPM. The PIM code is a NuScale-developed proprietary code used to assess the stability characteristics of the NPM during operation.

The NIST-1 is a scaled representation of the NPM reactor, containment, and reactor pool systems. It is constructed of stainless steel and has a maximum operating pressure of 1650 psia (11.4 MPa) and temperature of 630 degrees F (605 degrees K). Being a scaled facility, the volumes, lengths, and areas for the NIST-1 geometry are obtained by multiplying the respective NPM design volumes, lengths, active heat transfer areas, and flow areas by the applicable scale factors determined through a detailed scaling analysis. This process ensures the NIST-1 properly captures the important thermal-hydraulic phenomena and processes that would occur in the plant.

A series of tests have been completed at the NIST-1, located on the Oregon State University campus in Corvallis, OR, in support of NuScale's Design Certification Application. These tests include:

- facility characterization tests used to develop the NRELAP5 model of the NIST-1.
- loss-of-coolant accident (LOCA) tests used to validate NRELAP5 for LOCA analyses.
- flow-stability tests used to validate PIM for reactor stability analyses.
- non-LOCA (anticipated operational occurrence) tests used to validate NRELAP5 for non-LOCA analyses.
- long-term cooling tests used to validate NRELAP5 for long term cooling analyses.

Data obtained from the NIST-1 tests identified above have been used to successfully validate the NRELAP5 and PIM codes for LOCA, non-LOCA, flow stability, and long term cooling applications. This test program was inspected by the NRC in accordance with Inspection Procedures IP 35034, 35017, and 36100.

#### **1.5.1.7 Control Rod Drive Mechanism Proof Test**

The control rod drive mechanism for the NPM contains two new features that are not common in conventional control rod drive mechanisms: a remote disconnect mechanism and rod position indication. A proof-of-concept testing program was conducted to demonstrate the validity of these new designs with regard to performance, reliability, and repeatability of each system.

Testing was completed at the Curtiss Wright facilities in Cheswick, PA, for both normal and remote disconnect operation of the coils. The test setup included a functional drive rod assembly, a prototypic remote disconnect gripper coil, a prototypic remote disconnect gripper latch, a prototypic lift coil, and weights to simulate the control rod assembly (CRA) with a prototypic CRA hub socket.

The remote disconnect mechanism was found to provide a reliable and repeatable method to engage and disengage the CRA within the reactor pressure vessel. This is

consistent with the results of the remote operation, lift verification, and manual disengagement testing that was performed.

The tests provided a demonstration of hardware performance, which has been extended to aid in the design of drive rod position detection circuitry. Information gained from this testing has been used as a development tool to improve the design and does not create a design basis for the final control rod drive mechanism.

#### **1.5.1.8 Steam Generator Flow Restrictor Hydraulic Test**

To maintain secondary-side (within-tube) flow stability during operation of the NPM helical SG, a flow restriction device is used at the inlet to each SG tube. Performance of the flow restriction device varies as a function of power level (flow rate), and manufacturing and installation tolerances. For some potential flow restrictor designs, an uncertainty exists in the analytically-determined performance characteristics. Therefore, testing of potential flow restriction device designs, including variants to account for installation and manufacturing tolerances, was conducted to confirm the actual performance is acceptable in comparison to the expected design performance. Specifically, testing was needed to confirm flow-dependent pressure drop performance and vibration behavior.

The three objectives of the flow restrictor hydraulic test were to: (1) experimentally determine performance (K-factor) for flow-restrictor design concepts to verify design performance analyses, (2) assess the vibration behavior of the flow restrictors under different operating conditions, and (3) determine sensitivity of flow restrictor design to manufacturing and installation tolerances. Flow testing was performed for three different flow restriction devices (configured in a near-prototypical SG tube inlet) using water at a range of flow conditions representative of Reynolds numbers prototypic of NPM operating conditions between 1 percent and 110 percent of full flow.

Three types of flow restrictors (annular, threaded, and center flow) were fabricated and tested at the Alden Research Laboratory hydraulic test facilities in Holden, MA. Multiple-instrumented prototypical SG tubes with shortened lengths were used. Water temperature, pressures, flow rates, and vibration were recorded for each flow restrictor under varying conditions. All restrictors were able to be tested at conditions such that excessive vibration did not occur and no tube contact was observed. The tests provided a demonstration of hardware performance, which has informed the design of the actual flow restrictor device.

#### **1.5.1.9 Steam Generator Flow-Induced Vibration**

In accordance with Regulatory Guide 1.20, a comprehensive vibration assessment program (CVAP) for reactor internals must be completed prior to commercial operation. The CVAP must verify the structural integrity of the reactor internals for flow-induced vibrations by analytical, testing, in-service measurement, and inspection programs. As part of the CVAP for the NPM, reactor internal components including the SG were screened to determine which were potentially susceptible to various flow-induced vibration phenomena. The screening identified that the SG helical tubing and support bars are potentially susceptible to turbulent buffeting, fluid-elastic instability, and vortex shedding phenomena.

The objectives of this test program are to characterize the flow-induced excitation and response that is the result of single-phase primary-water flow (i.e., flow on the outside of the helical tubes). Specifically, tests are intended to determine the natural frequency of oscillation and the waveform of the tubes and prototypical support bars in both air and water, and to obtain data to characterize the dynamic pressure fluctuations and the stress and strain of tubes and supports for mass flow rates typical of the reactor primary side.

The test is performed using a test article that includes a 252-tube bundle modeling five of the 21 helical coil columns in the NPM SG. This number of tubes is sufficient to capture the effects of interactions between the tube coils while allowing for a simplified test specimen. The tube supports are prototypical and full height in order to accurately represent the frequency response of the helical SG, which is significantly affected by contact and gapped conditions at the supports and the free span length of the tubes between supports.

The test specimen is fabricated at General Vessels in Cremona, Italy, and assembled at the SIET facility in Piacenza, Italy. Tests are conducted at the AREVA facility in Erlangen, Germany, at ambient temperature with corrections for thermal expansion and fluid properties determined analytically. The tests are run at low pressure because external pressure has been demonstrated to have a relatively insignificant effect on tube frequencies and vibration response. Tests are being run with air inside tubes with corrections to frequencies due to added mass and flow of water and steam within the tubes predicted analytically.

Testing is conducted under flow conditions that are appropriately scaled to simulate reactor coolant flow rates from 75 percent to 120 percent power, or greater.

This testing program is currently underway. Data validate methods used for determination of mode frequency and mode shape for NPM SG helical coils, including modeling of tube supports. Data also characterize primary side dynamic pressure fluctuations and SG tube vibration response under turbulent flow conditions representative of normal operations.

#### **1.5.1.10 Fuel Mechanical and Hydraulic Testing**

The hydraulic and mechanical properties of the fuel assemblies must be known in order to computationally predict the response of the reactor to normal and off-normal conditions. This information is typically collected by physically testing the fuel assemblies in hydraulic and mechanical test facilities specially designed to simulate the operating conditions. The NuFuel HTP2 design uses fuel assembly components that have differences from those that have been previously characterized and that are currently deployed in operating reactors. Differences include reduced flow rates associated with natural circulation operation, and a shorter fuel assembly length. These differences directly affect the hydraulic and mechanical characteristics of the fuel assemblies; therefore, new hydraulic and mechanical testing was performed to measure various mechanical characteristics of a prototypical fuel assembly in response to normal and off-normal conditions under NPM-specific operating conditions.

The prototypical fuel assembly was identical to an actual NuFuel HTP2 fuel assembly except it used depleted uranium fuel pellets instead of enriched uranium fuel pellets. Because the physical density and weights of enriched and depleted uranium are almost identical, this difference did not affect testing results. The testing included several subsets of test series in order to evaluate the following:

- axial-flow pressure drop along the assembly
- required flow rate for assembly lift off from support plate
- fuel-rod wear at support grids during nominal flow conditions
- lateral-load deflection (stiffness)
- free-vibration modes frequencies and amplitudes
- baffle-impact stiffness (lateral impact)
- lateral-forced vibration modes frequencies and amplitudes
- mode evolution
- axial stiffness
- drop response (dynamic axial response)

The test program was conducted at AREVA's Richland Testing Facility in Richland, WA to characterize and verify mechanical performance of the NuScale fuel. The testing program was completed and provided data needed to support analysis of the fuel assembly during normal operations, shipping and handling, and seismic events.

#### **1.5.1.11 Control Rod Assembly Guide Tube Flow Test**

In a traditional PWR plant, the CRA guide tube structures above the core and the control rod end caps are susceptible to wear or fatigue due to large crossflow velocities and leakage flow through the fuel assembly guide tubes at the exit of the fuel assemblies. In the NPM design, core flow velocity is low as a result of the plant's natural circulation design, and flow exits the CRA guide tube structure vertically into the riser assembly. It is expected that crossflow rates passing through the CRA guide tube structures and leakage flow rate through the fuel assembly guide tubes are less significant than in a typical PWR. To confirm this assumption, flow-induced vibration tests in the lower riser region have been developed.

Testing is being performed at the AREVA Technical Center in Erlangen, Germany. The test specimen includes a prototypical CRA guide tube assembly, a prototypical CRA, and a prototypical fuel assembly. The test is being performed under a static system pressure and temperature, with a flow velocity varying up to 250 percent of the nominal NPM flow rate.

This testing program is currently underway. Test results will characterize the response of the CRA fingers and rodlets due to single-phase primary flow-induced vibration (e.g., turbulent flow excitation) and will characterize the flow patterns and velocities at planes through CRA guide tube structure openings.

### 1.5.1.12 Control Rod Assembly Drop and Control Rod Drive Shaft Alignment Test

The NPM is designed with control rod drive shafts that are longer than conventional PWR designs and have the capability to be remotely disconnected. These shafts are aligned using the following multiple-support features:

- control rod drive mechanism nozzles in the reactor vessel head
- integrated steam plenum pressurizer baffle plate
- five upper control rod drive shaft supports in the upper riser section
- a control rod drive shaft alignment cone located at the top of the CRA guide tube

The design uses a CRA and fuel-assembly design similar to, but shorter than, traditional operating reactors. The arrangement of a shorter fuel assembly and CRA coupled to a longer drive shaft creates a unique configuration of these components with no operational experience or testing experience. The CRA-drop and control rod drive shaft-alignment test program was developed to confirm the operability of this unique design.

Testing is being performed at the AREVA Technical Center in Erlangen, Germany, and is divided into two main test configurations: an ambient pressure and temperature test, and an elevated temperature condition test. The ambient test configuration is composed of a full-length drive shaft coupled with the NPM control rod assembly and fuel assembly, as well as the control rod drive shaft support structures and a CRA guide tube assembly. The CRA guide tube assembly and fuel assembly are joined in the water under ambient conditions with no coolant flow. During the test, the coupled CRA and control rod drive shaft assembly drops with variations in the alignment of the control rod drive shaft supports and CRA guide tube assembly and a mid-span deflection of the fuel assembly.

In the elevated-condition test configuration, the test specimen is composed of a partial-length control rod drive shaft, with additional weight added to simulate the total weight of the full-length control rod drive shaft. The CRDA is coupled with a CRA, fuel assembly, and a CRA guide tube assembly contained in a pressure vessel. During the test, the CRA drops with variations of CRA guide tube assembly misalignment configurations under nominal operation pressure and flow conditions.

This testing program is currently underway. Test results are used to confirm the operability of the control rod drive shafts for a range of potential component conditions and distortions.

## 1.5.2 NuScale Test and Inspection Plans

Information on NuScale test and inspection plans related to plant startup testing is provided in Section 14.2.



**1.6 Material Referenced**

Topical reports and technical reports that are incorporated by reference as part of the NuScale Power Plant Design Certification Application are listed in Table 1.6-1 and Table 1.6-2, respectively.

**Table 1.6-1: NuScale Referenced Topical Reports**

<b>Topical Report Number</b>	<b>Topical Report Title</b>	<b>Submittal Date</b>	<b>FSAR Section</b>
TR-1010-859-A, Rev 3	NuScale Topical Report: Quality Assurance Program Description for the NuScale Power Plant	May 2015	17
TR-0515-13952-A, Rev 0	Risk Significance Determination	July 2015	17, 19
TR-0815-16497, Rev 0	Safety Classification of Passive Nuclear Power Plant Electrical Systems	October 2015	8
TR-1015-18653, Rev 1	Design of the Highly Integrated Protection System Platform Topical Report	November 2016	7
TR-0915-17565, Rev 1	Accident Source Term Methodology	April 2016	15
TR-0116-20825, Rev 0	Applicability of AREVA Fuel Methodology for the NuScale Design	July 2016	4
TR-0616-48793, Rev 0	Nuclear Analysis Codes and Methods Qualification	August 2016	4
TR-0516-49417, Rev 0	Evaluation Methodology for Stability Analysis of the NuScale Power Module	July 2016	4
TR-0516-49422, Rev 0	LOCA Evaluation Model	December 2016	15
TR-0915-17564, Rev 0	Subchannel Analysis Methodology	October 2016	4
TR-0516-49416, Rev 0	Non-LOCA Methodologies		15
TR-0116-21012, Rev 0	NuScale Power Critical Heat Flux Correlation NSP2	October 2016	4
TR-0716-50350, Rev 0	Rod Ejection Analysis Methodology	December 2016	15
TR-0716-50351, Rev 0	NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces	December 2015	4
TR-0915-17772, Rev 0	Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Site	December 2016	15

**Table 1.6-2: NuScale Referenced Technical Reports**

Report Number	Title	FSAR Section
TR-0416-48929	NuScale Design of Physical Security Systems	13.6
TR-0516-49084	Containment Analysis Methodology	6.2
TR-0716-50439	Comprehensive Vibration Assessment Program (CVAP) Technical Report TR-0716-50439	14.2
TR-0816-50796	Loss of Large Areas Due to Explosions and Fires Assessment	20.2
TR-0816-51127	NuFuel HTP2 Fuel and Control Rod Assembly Designs	4.2
TR-0916-51299	Long-Term Cooling Methodology	15
TR-1116-52065	Effluent Release Methodology Technical Report	11.1, 11.2, 11.3
RP-0215-10815	Concept of Operations	18.7
RP-0316-17614	Human Factors Engineering Operating Experience Review Results Summary Report	18.2
RP-0316-17615	Human Factors Engineering Functional Requirements Analysis and Function Allocation Results Summary Report	18.3
RP-0316-17616	Human Factors Engineering Task Analysis Results Summary Report	18.4
RP-0316-17617	Human Factors Engineering Staffing and Qualifications Results Summary Report	18.5
RP-0316-17618	Human Factors Engineering Treatment of Important Human Actions Results Summary Report	18.6
RP-0316-17619	Human Factors Engineering Human-System Interface Design Results Summary Report	18.7
RP-0516-49116	Control Room Staffing Plan Validation Results	18.5
RP-0914-8534	Human Factors Engineering Program management Plan	18.1
RP-0914-8543	Human Factors Verification and Validation Implementation Plan	18.1
RP-0914-8544	Human Factors Engineering Design Implementation Implementation Plan	18.11
RP-1215-20253	Control Room Staffing Plan Validation Methodology	18.5

## 1.7 Drawings and Other Detailed Information

Where appropriate, simplified instrumentation and controls (I&C), electrical, or mechanical drawings are provided as figures. These figures are used in conjunction with the written text in the associated section to provide visual clarification of design information. Component position indications shown on these figures do not represent a specific operational state unless noted.

### 1.7.1 Electrical and Instrumentation and Control Drawings

Table 1.7-1 provides a list of I&C functional diagrams and electrical one-line diagrams used in the FSAR.

See Figure 1.7-1a, Figure 1.7-1b, and Figure 1.7-2 for the legends of the symbols and characters used in electrical and I&C diagrams.

COL Item 1.7-1: A COL applicant that references the NuScale Power Plant design certification will provide site-specific diagrams and legends, as applicable.

### 1.7.2 Piping and Instrumentation Diagrams

Table 1.7-2 provides a list of system drawings used in the FSAR.

See Figure 1.7-3a through Figure 1.7-3f for a legend of the symbols and characters used in P&IDs.

COL Item 1.7-2: A COL applicant that references the NuScale Power Plant design certification will list additional site-specific P&IDs and legends as applicable.

**Table 1.7-1: Instrumentation and Controls Functional and Electrical One-Line Diagrams**

<b>Figure</b>	<b>Title</b>
Figure 7.0-1	Overall Instrumentation and Controls System Architecture Diagram
Figure 7.0-3	Module Protection System Safety Architecture Overview
Figure 7.0-4	Separation Group A Communication Architecture
Figure 7.0-5	Separation Group-A and Division I Reactor Trip System and Engineered Safety Features Actuation System Communication Architecture
Figure 7.0-6	Reactor Trip Breaker Arrangement
Figure 7.0-7	Equipment Interface Module Configuration
Figure 7.0-8	Equipment Interface Module Output
Figure 7.0-9	Pressurizer Trip Breaker Arrangement
Figure 7.0-10	Module Protection System Gateway Diagram
Figures 7.0-11a and 7.0-11b	Module Protection System Power Distribution
Figure 7.0-12	Neutron Monitoring System Ex-Core Block Diagram
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Figure 7.0-14	Safety Display and Indication System Boundary
Figure 7.0-15	Safety Display and Indication Hub
Figure 7.0-16	Display Interface Module
Figure 7.0-17	Module Control System Internal Functions and External Interfaces
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Figure 8.3-1	Station Single Line Diagram
Figures 8.3-2a and 8.3-2b	13.8kV and Switchyard System
Figures 8.3-3a and 8.3-3b	Medium Voltage Electrical System
Figures 8.3-4a through 8.3-4z	Low Voltage Electrical System
Figures 8.3-5a and 8.3-5b	Backup Power Supply System
Figure 8.3-6	Highly Reliable DC Power System (Common)
Figures 8.3-7a and 8.3-7b	Highly Reliable DC Power System (Module Specific)
Figures 8.3-8a through 8.3-8f	Normal DC Power System
Figure 11.5-2	Process and Effluent Radiation Monitoring System I&C Configuration

**Table 1.7-2: System Drawings**

<b>Figure</b>	<b>Title</b>
Figure 5.1-2	Reactor Coolant System Simplified Diagram
Figure 5.4-9	Decay Heat Removal System Simplified Diagram
Figure 6.3-1	Emergency Core Cooling System
Figure 6.4-1	Control Room Habitability System Simplified Diagram
Figure 9.1.3-1	Spent Fuel Pool Cooling System Diagram
Figures 9.1.3-2a and 9.1.3-2b	Reactor Pool Cooling System Diagram
Figure 9.1.3-3	Pool Cleanup System Diagram
Figure 9.1.3-4	Pool Surge Cooling System Diagram
Figure 9.2.2-1	Schematic of Reactor Component Cooling Water System
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Figure 9.2.5-2	Ultimate Heat Sink Qualified Makeup Line
Figure 9.2.7-1	Site Cooling Water System Schematic
Figure 9.2.8-1	Chilled Water System Piping and Instrumentation Drawing
Figure 9.2.9-1	Utility Water System
Figure 9.3.1-1	Instrument Air and Service Air
Figure 9.3.1-2	Nitrogen Distribution System
Figure 9.3.3-1	Radioactive Waste Drain System Simplified Configuration
Figure 9.3.3-2	Balance-of-Plant Drain System Simplified Configuration
Figure 9.3.4-1	Chemical and Volume Control System Simplified Diagram (Chemical and Volume Control System Module 1 with Module Heatup System Subsystem 6A Shown)
Figure 9.3.4-2	Boron Addition System Simplified Diagram
Figure 9.3.6-1	Containment Evacuation System
Figure 9.3.6-2	Containment Flooding and Drain System
Figure 9.4.1-1	Control Room Ventilation System Simplified Diagram
Figure 9.4.2-1	Reactor Building HVAC System Simplified Diagram
Figure 9.4.3-1	Radioactive Waste Building HVAC System Simplified System Diagram
Figure 9.4.4-1	Turbine Building HVAC System Simplified Diagram
Figure 9.5.1-1	Fire Protection System Water Supplies and Fire Pumps
Figure 9.5.1-2	Fire Protection System Yard Fire Main Loop
Figure 10.1-1	Power Conversion System Block Flow Diagram
Figure 10.1-2	Flow Diagram and Heat Balance Diagram at Rated Power for Steam and Power Conversion System Cycle
Figure 10.2-1	Turbine Generator System Piping and Instrumentation Diagram
Figure 10.4-1	Condenser Piping and Instrumentation Diagram
Figure 10.4-2	Condenser Air Removal System Piping and Instrumentation Diagram
Figure 10.4-3	Circulating Water System Piping and Instrumentation Diagram (Typical of 2)
Figures 10.4-4a and 10.4-4b	Auxiliary Boiler System Piping and Instrumentation Diagram
Figures 11.2-1a through 11.2-1j	Liquid Radioactive Waste System Diagram
Figures 11.3-1a and 11.3-1b	Gaseous Radioactive Waste System Diagram
Figure 11.4-1	Block Diagram of the Solid Radioactive Waste System
Figure 11.4-2a	Process Flow Diagram for SRW Wet Solid Waste
Figure 11.4-2b	Solid Radioactive Waste System

### E-LINE DIAGRAM LEGEND

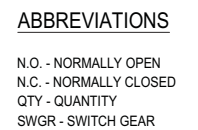


Figure 1.7-1b: Electrical Symbols

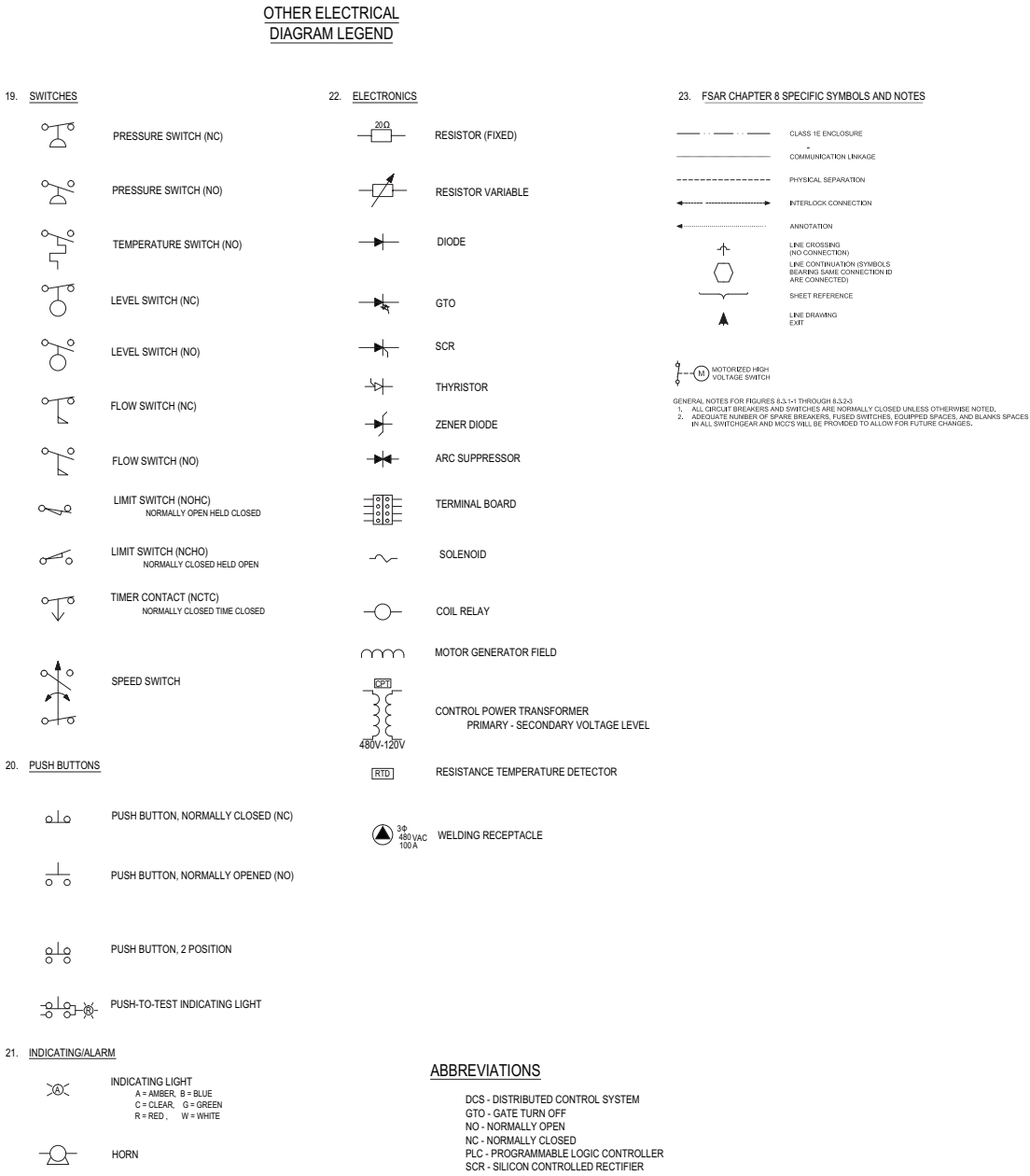




Figure 1.7-2: Instrumentation and Controls Symbol Legend

SYMBOL	LOGIC FUNCTION	DESCRIPTION	SYMBOL	LOGIC FUNCTION	DESCRIPTION
	OR	THE OR GATE OUTPUT IS TRUE IF ANY INPUT IS TRUE.		UNSPECIFIED FUNCTION	THE OUTPUT VALUE IS A NONLINEAR OR UNSPECIFIED FUNCTION OF THE INPUT. THE FUNCTION IS DEFINED IN A NOTE OR OTHER TEXT.
	AND	THE AND GATE OUTPUT IS TRUE ONLY IF ALL INPUTS ARE TRUE.		SUMMATION	THE OUTPUT VALUE IS THE ALGEBRAIC SUM OF THE INPUTS.
	COINCIDENT LOGIC	THE AND GATE OUTPUT IS TRUE ONLY IF 2 OUT OF 4 INPUTS, OR MORE, ARE TRUE.			
	NOT	THE NOT GATE OUTPUT IS FALSE IF THE INPUT IS TRUE. THE OUTPUT IS TRUE IF THE INPUT IS FALSE.			
	TIME DELAY	A FUNCTION WHICH PRODUCES AN OUTPUT FOLLOWING A PRESET TIME DELAY AFTER RECEIVING AN INPUT. IF THE INPUT CHANGES FROM TRUE TO FALSE BEFORE THE PRESET TIME DELAY ELAPSES, THEN THE OUTPUT REMAINS FALSE			
	ELECTRONIC BISTABLE OUTPUT INDICATOR	BISTABLE OUTPUT IS A LOGIC "1" WHEN THE MEASURED VARIABLE IS GREATER THAN THE SETPOINT VALUE.			
PS LS TS NS FS VS	PRESSURE BISTABLE LEVEL BISTABLE TEMPERATURE BISTABLE NEUTRON FLUX BISTABLE FLOW BISTABLE VOLTAGE BISTABLE	BISTABLE OUTPUT IS A LOGIC "1" WHEN THE MEASURED VARIABLE IS LESS THAN THE SETPOINT VALUE.			
A	SEPARATION GROUP (SG)	ONE OF FOUR REDUNDANT SEPARATION GROUPS (SGS) SGS ARE IDENTIFIED AS A, B, C, OR D.			
	MOMENTARY HAND SWITCH	MOMENTARY HAND SWITCH LOCATED IN THE MAIN CONTROL ROOM .			
	MAINTAINED-POSITION HAND SWITCH	MAINTAINED-POSITION HAND SWITCH LOCATED IN THE MAIN CONTROL ROOM .			
	STATUS INDICATION	MAIN CONTROL ROOM INDICATION OF SYSTEM TRIP/ACTUATION STATUS.			
	STATUS INDICATION	MAIN CONTROL ROOM INDICATION OF OPERATIONAL BYPASS PERMISSIVE STATUS.			
	ANNUNCIATOR	MAIN CONTROL ROOM ANNUNCIATOR FOR SYSTEM TRIP/ACTUATION STATUS.			
	ANNUNCIATOR	MAIN CONTROL ROOM FIRST OUT ANNUNCIATOR FOR SYSTEM TRIP/ACTUATION STATUS.			
	MAINTAINENCE/TEST SWITCH	LOCALLY MOUNTED SWITCH FOR MAINTENANCE AND TESTING.			
	MCS MANUAL CONTROL INTERFACE	MCS MOMENTARY MANUAL CONTROL INTERFACE LOCATED IN THE MAIN CONTROL ROOM FOR NONSAFETY CONTROL OF MPS ACTUATED EQUIPMENT.			

Figure 1.7-3a: Piping and Instrumentation Diagram Legends

VALVES		FIRE & SAFETY		SPECIALTY ITEMS		FITTINGS/MISC	DRAINS
	BALL VALVE		NEEDLE VALVE		FLAME ARRESTOR		ALL VENTS & DRAINS ON THE P&ID REPRESENT A 0.75" NORMALLY CLOSED VALVE. EXCEPTIONS MUST BE NOTED ON P&ID.  CLOSED VENT OR DRAIN XXXX = SYSTEM ABBREVIATION
	THREE-WAY BALL VALVE (NOTE 1)		KNIFE GATE VALVE		HAMMER ARRESTOR		
	FOUR-WAY, FOUR-PORTED BALL VALVE (NOTE 1)		SLIDE VALVE		EXPANSION JOINT		 OPEN VENT OR DRAIN XXXX = SYSTEM ABBREVIATION
	STRAIGHT GLOBE VALVE		THREE-WAY SLIDE VALVE		HINGED EXPANSION JOINT		
	ANGLED GLOBE VALVE		FLOAT VALVE		SWIVEL JOINT		 FLOOR DRAIN
	THREE-WAY GLOBE VALVE (NOTE 1)		Y BLOWDOWN VALVE		SINGLE BASKET STRAINER		
	SAFETY ANGLE VALVE / GENERIC TWO-WAY ANGLE VALVE		ANGLE BLOWDOWN VALVE		DUPLEX BASKET STRAINER		 DRAIN FUNNEL
	THREE-WAY GATE VALVE / GENERIC THREE-WAY VALVE (NOTE 1)		AUTOMATIC RECIRCULATION VALVE		SIMPLEX BASKET STRAINER		
	FOUR-WAY, FOUR-PORTED GATE VALVE / GENERIC FOUR-WAY VALVE (NOTE 1)		VALVE STEM EXTENDED THROUGH SHIELDED WALL		SUMP STRAINER		 CLEAN OUT
	GATE VALVE / GENERIC TWO-WAY VALVE		BACKFLOW PREVENTER		T-STRAINER		
	FOUR-WAY, FIVE-PORTED VALVE (NOTE 1)		FOOT VALVE		STARTUP STRAINER		 DRAIN
	BUTTERFLY VALVE		EXTENDED BODY GATE VALVE		SCREEN STRAINER		
	PLUG VALVE		HOSE VALVE		CONE STRAINER		 GATE VALVE, FLANGED
	THREE-WAY PLUG VALVE (NOTE 1)		ANGLED HOSE VALVE		TEMPORARY STRAINER		
	FOUR-WAY PLUG VALVE (NOTE 1)				VENT		 GATE VALVE, PLUGGED
	ECCENTRIC ROTARY DISC VALVE				VACUUM BREAKER		
	DIAPHRAGM VALVE				Y-STRAINER		 BALL VALVE, FLANGED
	PINCH VALVE				FILTER		
	BELLOWS SEALED VALVE				DISTRIBUTOR		 BALL VALVE, PLUGGED
	CHECK VALVE				INLET AIR FILTER		
	CHECK VALVE WITH 3/32 ORIFICE IN CLAPPER				REMOVABLE SPOOL		 WALL PENETRATION
	STOP CHECK VALVE				MECHANICAL COUPLING		
	WAFER CHECK VALVE				DRIP LEG		
	TILTING DISC CHECK VALVE				EXHAUST VENT		
	ANGLE CHECK VALVE				FREE VENT WITH SCREEN		
	LIFT CHECK VALVE				FREE VENT WITHOUT SCREEN		
	EXCESS FLOW CHECK VALVE				SAMPLE COOLER		
	STEM LEAK-OFF VALVE				SPRAY DESUPERHEATER		
	TRIPLE DUTY VALVE				DESUPERHEATER		
					PACKED BED		

NOTES:

1. ARROW INDICATES FAILURE OR UNACTUATED FLOW PATH.

### Figure 1.7-3b: Piping and Instrumentation Diagram Legends


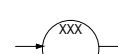











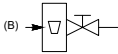



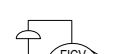








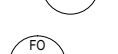











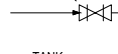


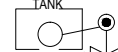


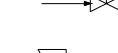


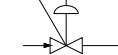



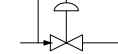








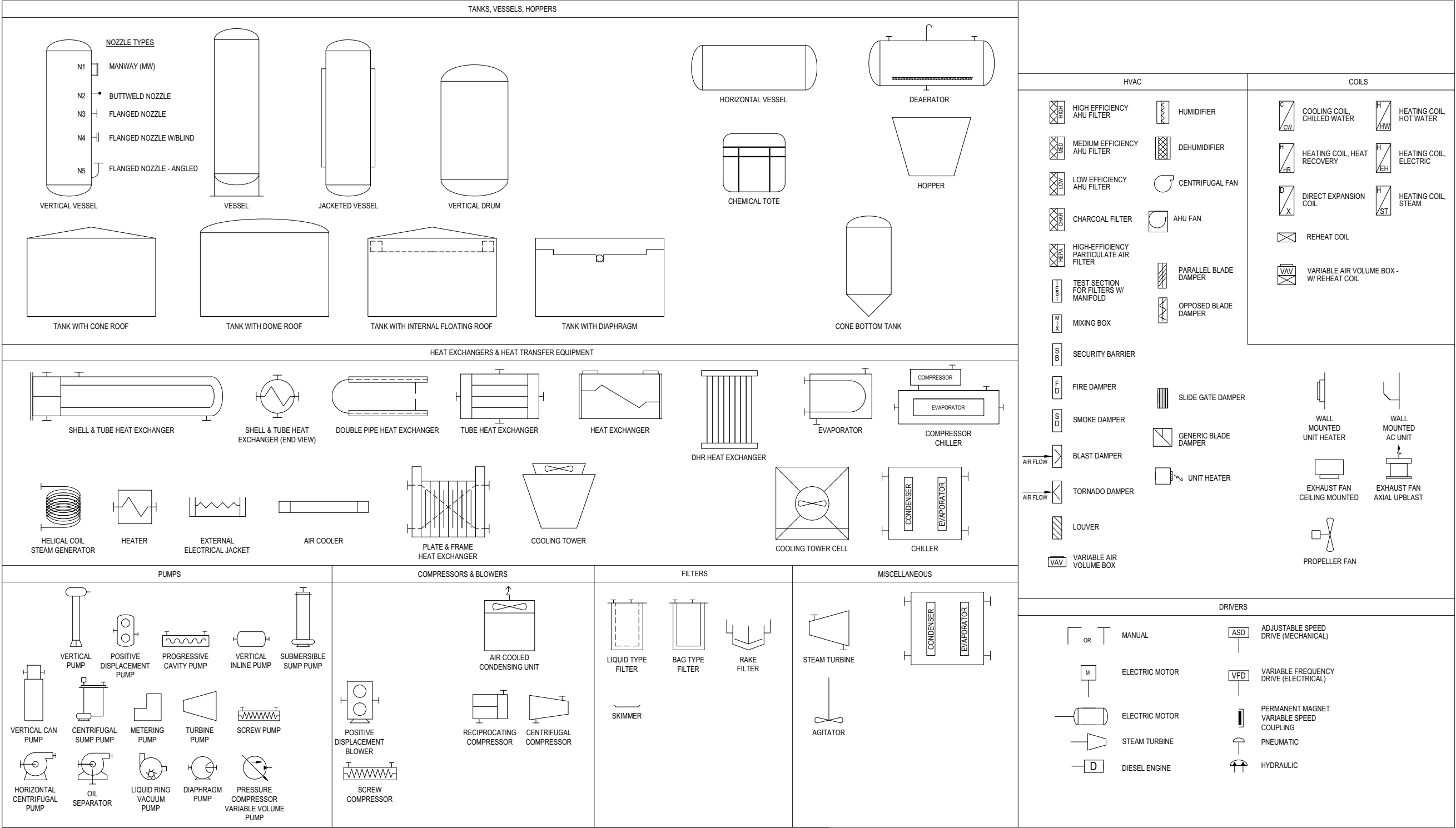
ACTUATORS		SELF-ACTUATING FINAL CONTROL ELEMENTS		ADDITIONAL ACTUATOR TYPES			
	PNEUMATIC ACTUATOR (GENERIC ACTUATOR / SPRING-DIAPHRAGM ACTUATOR)		AUTOMATIC FLOW REGULATOR XXX=FCV WITHOUT INDICATOR XXX=FICV WITH INTEGRAL INDICATOR		PRESSURE SETTING PSET = 100 psig		BALL FLOAT
	PNEUMATIC ACTUATOR (SPRING-DIAPHRAGM ACTUATOR W/ POSITIONER)		VARIABLE AREA FLOWMETER W/ INTEGRAL MANUAL ADJUSTING VALVE (INSTRUMENT TAG BUBBLE REQUIRED WITH 'B')		PRESSURE-REDUCING REGULATOR W/ INTEGRAL OUTLET PRESSURE RELIEF AND PRESSURE GAUGE.		CAPACITANCE SENSOR
	PNEUMATIC ACTUATOR (PRESSURE-BALANCED DIAPHRAGM ACTUATOR)		CONSTANT FLOW REGULATOR		ANGLE PRESSURE RELIEF VALVE / GENERIC PRESSURE SAFETY VALVE		DISPLACEMENT FLOAT
	LINEAR PISTON ACTUATOR (SINGLE-ACTING, SPRING-OPOSED OR DOUBLE-ACTING)		FLOW SIGHT GLASS (TYPE SHALL BE NOTED IF MORE THAN ONE TYPE IS USED)		ANGLE VACUUM RELIEF VALVE / GENERIC VACUUM SAFETY VALVE		PADDLE WHEEL
	LINEAR PISTON ACTUATOR WITH POSITIONER		GENERIC FLOW RESTRICTION / SINGLE STAGE ORIFICE PLATE (NOTE REQUIRED FOR MULTI-STAGE OR CAPILLARY TUBE TYPES)		STRAIGHT-THRU PRESSURE RELIEF VALVE		
	ROTARY PISTON ACTUATOR (SINGLE-ACTING, SPRING-OPOSED OR DOUBLE-ACTING)		RESTRICTION ORIFICE DRILLED IN VALVE PLUG (TAG NUMBER SHALL BE OMITTED IS VALVE IS OTHERWISE IDENTIFIED)		PRESSURE-VACUUM RELIEF VALVE		
	ROTARY PISTON ACTUATOR WITH POSITIONER		LEVEL REGULATOR W/ BALL FLOAT AND MECHANICAL LINKAGE		PRESSURE SAFETY ELEMENT / PRESSURE RUPTURE DISK		
	BELLOWS SPRING OPOSED ACTUATOR		BACKPRESSURE REGULATOR, INTERNAL PRESSURE TAP		VACUUM SAFETY ELEMENT / VACUUM RUPTURE DISK		
	ROTARY MOTOR-OPERATED ACTUATOR		BACKPRESSURE REGULATOR, EXTERNAL PRESSURE TAP		TEMPERATURE REGULATOR W/ FILLED THERMAL SYSTEM		
	SOLENOID ACTUATOR (OPEN-CLOSE OR MODULATING)		PRESSURE REDUCING REGULATOR, INTERNAL PRESSURE TAP		THREE-WAY TEMPERATURE REGULATOR W/ FILLED THERMAL SYSTEM		
	ACTUATOR WITH SIDE-MOUNTED HANDWHEEL		PRESSURE REDUCING REGULATOR, EXTERNAL PRESSURE TAP		ANGLED TEMPERATURE REGULATOR W/ FILLED THERMAL SYSTEM		
	ACTUATOR WITH TOP-MOUNTED HANDWHEEL		DIFFERENTIAL PRESSURE REGULATOR, INTERNAL PRESSURE TAPS		THERMAL SAFETY ELEMENT, FUSIBLE PLUG OR DISK		
	MANUAL ACTUATOR		DIFFERENTIAL PRESSURE REGULATOR, EXTERNAL PRESSURE TAPS		STEAM TRAP / GENERIC MOISTURE TRAP (NOTE REQUIRED FOR OTHER TRAP TYPES)		
	ACTUATOR WITH MANUAL ACTUATED PARTIAL STROKE TEST DEVICE		DIFFERENTIAL PRESSURE REGULATOR W/ INTERNAL & EXTERNAL PRESSURE TAPS		MOISTURE TRAP WITH EQUALIZATION LINE		
	ACTUATOR WITH REMOTE ACTUATED PARTIAL STROKE TEST DEVICE		PRESSURE REDUCING REGULATOR, INTERNAL PRESSURE TAP				
	ON-OFF SOLENOID ACTUATOR, NON-LATCHING / AUTOMATIC RESET		PRESSURE REDUCING REGULATOR, INTERNAL PRESSURE TAP				
	ON-OFF SOLENOID ACTUATOR, LATCHING / ON-OFF SOLENOID ACTUATOR, MANUAL OR REMOTE RESET		PRESSURE REDUCING REGULATOR, INTERNAL PRESSURE TAP				
	ON-OFF SOLENOID ACTUATOR, LATCHING / ON-OFF SOLENOID ACTUATOR, MANUAL AND REMOTE RESET		PRESSURE REDUCING REGULATOR, INTERNAL PRESSURE TAP				
	SPRING OR WEIGHT ACTUATED RELIEF OR SAFETY VALVE ACTUATOR						
	PILOT ACTUATED RELIEF OR SAFETY VALVE ACTUATOR W/ PRESSURE SENSING LINE (PILOT PRESSURE SENSING LINE DELETED IF SENSING IS INTERNAL)						
	ELECTRIC TO PNEUMATIC CONTROL SIGNAL CONVERTER						
	ELECTRIC TO HYDRAULIC CONTROL SIGNAL CONVERTER						
	VOLTAGE TO CURRENT CONTROL SIGNAL CONVERTER						
	ELECTROHYDRAULIC LINEAR OR ROTARY ACTUATOR						

Figure 1.7-3c: Piping and Instrumentation Diagram Legends



VERTICAL VESSEL

VESSEL

JACKETED VESSEL

VERTICAL DRUM

TANK WITH CONE ROOF

TANK WITH DOME ROOF

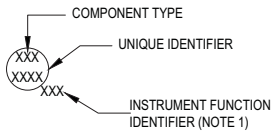
TANK WITH INTERNAL FLOATING ROOF

TANK WITH DIAPHRAGM

Figure 1.7-3d: Piping and Instrumentation Diagram Legends

INSTRUMENTATION DEVICE AND FUNCTION SYMBOLS (NOTE 2)				
SHARED DISPLAY, SHARED CONTROL				
PRIMARY CHOICE OR BASIC PROCESS CONTROL SYSTEM	ALTERNATE CHOICE OR SAFETY INSTRUMENTED SYSTEM	COMPUTER SYSTEMS AND SOFTWARE	DISCRETE	LOCATION AND ACCESSIBILITY
				<ul style="list-style-type: none"><li>LOCATED IN FIELD</li><li>NOT PANEL, CABINET OR CONSOLE MOUNTED</li><li>VISIBLE AT FIELD LOCATION</li><li>NORMALLY OPERATOR ACCESSIBLE</li></ul>
				<ul style="list-style-type: none"><li>LOCATED IN OR ON FRONT OF CONTROL OR MAIN PANEL OR CONSOLE</li><li>VISIBLE ON FRONT OF PANEL OR ON VIDEO DISPLAY</li><li>NORMALLY OPERATOR ACCESSIBLE AT PANEL FRONT OR CONSOLE</li></ul>
				<ul style="list-style-type: none"><li>LOCATED IN REAR OF CENTRAL OR MAIN PANEL</li><li>LOCATED IN CABINET BEHIND PANEL</li><li>NOT VISIBLE ON FRONT OF PANEL OR ON VIDEO DISPLAY</li><li>NOT NORMALLY OPERATOR ACCESSIBLE AT PANEL OR CONSOLE</li></ul>
				<ul style="list-style-type: none"><li>LOCATED IN OR ON FRONT OF SECONDARY OR LOCAL PANEL OR CONSOLE</li><li>VISIBLE ON FRONT OF PANEL OR ON VIDEO DISPLAY</li><li>NORMALLY OPERATOR ACCESSIBLE AT PANEL FRONT OR CONSOLE</li></ul>
				<ul style="list-style-type: none"><li>LOCATED IN REAR OF SECONDARY OR LOCAL PANEL</li><li>LOCATED IN FIELD CABINET</li><li>NOT VISIBLE ON FRONT OF PANEL OR ON VIDEO DISPLAY</li><li>NOT NORMALLY OPERATOR ACCESSIBLE AT PANEL OR CONSOLE</li></ul>

INSTRUMENT TAGGING



NOTES:

- INSTRUMENT FUNCTION IDENTIFIER: USED ONLY WHEN THE COMPONENT TYPE REQUIRES FURTHER CLARIFICATION. SEE FUNCTION IDENTIFIERS TABLE ON FIGURE 1.7-3e.
- FOR INSTRUMENTATION TYPE AND FUNCTION, SEE INSTRUMENTATION IDENTIFICATION LETTERS TABLE ON THIS SHEET AND FUNCTION IDENTIFIERS TABLE ON FIGURE 1.7-3e.
- TEMPERATURE ELEMENTS ARE THERMOCOUPLES UNLESS NOTED OTHERWISE.
- FOR GUIDANCE ON THE USE OF THE INSTRUMENTATION IDENTIFICATION LETTERS TABLE, REFER TO ANSI/ISA 5.1-2009.

TYPICAL INSTRUMENT COMPONENT COMBINATIONS																	
		CONTROLLERS			READOUT DEVICES	SWITCHES AND ALARM DEVICES*			TRANSMITTER		SOLENOIDS, RELAYS, COMPUTING DEVICES						
FIRST LETTERS	INDICATING OR MEASURABLE VARIABLE	INDICATING	BLIND	CONTROL VALVES	INDICATING	HIGH**	LOW**	COMB**	INDICATING	BLIND		PRIMARY ELEMENT	TEST POINT	WELL OR PROBE	VIEWING DEVICE, GLASS	SAFETY DEVICE	FINAL ELEMENT
A	ANALYSIS	AIC	AC		AI	ASH	ASL	ASHL	AIT	AT	AY	AE	AP	AW			AV
B	BURNER/COMBUSTION	BIC	BC		BI	BSH	BSL	BSHL	BIT	BT	BY	BE		BW	BG		BZ
C	USER'S CHOICE																
D	USER'S CHOICE																
E	VOLTAGE	EIC	EC		EI	ESH	ESL	ESHL	EIT	ET	EY	EE					EZ
F	FLOW RATE	FIC	FC	FCV	FI	FSH	FSL	FSJ	FIT	FT	FY	FE	FP		FG		FV
FQ	FLOW QUANTITY	FQIC			FQI	FQSH	FQSL		FQI/T	FQT	FQY	FQE					FQV
FF	FLOW RATIO	FFIC	FFC		FFI	FFSH	FFSL										
G	USER'S CHOICE																
H	HAND	HIC	HC					HS									HV
I	CURRENT	IIC			II	ISH	ISL	ISHL	IIT	IT	IY	IE					IZ
J	POWER	JIC			JI	JSH	JSL	JSHL	JIT	JT	JY	JE					JZ
K	TIME	KIC	KC	KCV	KI	KSH	KSL	KSHL	KIT	KT	KY	KE					KZ
L	LEVEL	LIC	LC	LCV	LI	LSH	LSL	LSHL	LIT	LT	LY	LE		LW	LG		LV
M	USER'S CHOICE																
N	USER'S CHOICE																
O	USER'S CHOICE																
P	PRESSURE/VACUUM	PIC	PC	PCV	PI	PSH	PSL	PSHL	PIT	PT	PY	PE	PP			PSV	PV
PD	PRESSURE, DIFFERENTIAL	PDIC	PDC	PDCV	PDI	PDSH	PDSL		PDIT	PDT	PDY	PDE	PDP			PSE	PDV
Q	QUANTITY	QIC			QI	QSH	QSL	QSHL	QIT	QT	QY	QE					QZ
R	RADIATION	RIC	RC		RI	RSH	RSL	RSHL	RIT	RT	RY	RE		RW			RZ
S	SPEED/FREQUENCY	SIC	SC	SCV	SI	SSH	SSL	SSHL	SIT	ST	SY	SE					SV
T	TEMPERATURE (NOTE 2)	TIC	TC	TCV	TI	TSH	TSL	TSHL	TIT	TT	TY	TE	TP	TW		TSE	TV
TD	TEMPERATURE, DIFFERENTIAL	TDIC	TDC	TDCV	TDI	TDSH	TDSL		TDIT	TDT	TDY	TDE	TDP	TDW			TCV
U	MULTI VARIABLE				UI												UV
V	VIBRATION/MACHINERY ANALYSIS				VI	VSH	VSL	VSHL	VIT	VT	VY	VE					
W	WEIGHT/FORCE	WIC	WC	WCV	WI	WSH	WSL	WSHL	WIT	WT	WY	WE					WZ
WD	WEIGHT/FORCE,DIFFERENTIAL	WDIC	WDC	WDCV	WDI	WDSH	WDSL		WDIT	WDT	WDY	WDE					WDZ
X	UNCLASSIFIED																XZ
Y	EVENT/STATE/PRESENCE	YIC	YC		YI	YSH	YSL			YT	YY	YE					YZ
Z	POSITION	ZIC	ZC	ZCV	ZI	ZSH	ZSL	ZSHL	ZIT	ZT	ZY	ZE					ZV
ZD	GAUGING/DEVIATION	ZDIC	ZDC	ZDCV	ZDI	ZDSH	ZDSL		ZDIT	ZDT	ZDY	ZDE					ZDV

NOTE: THIS TABLE IS NOT ALL-INCLUSIVE

\*A, ALARM, THE ANNUNCIATION DEVICE, MAY BE USED IN THE SAME FASHION AS S, SWITCH, THE ACTUATION DEVICE

\*\* THE LETTERS "H" AND "L" MAY BE OMITTED IF NOT DEFINED. IF APPROPRIATE, "C" (CLOSED) AND

"O" (OPEN) MAY BE USED IN PLACE OF "H" AND "L."


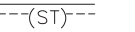



OTHER POSSIBLE COMBINATIONS:




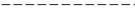
















FO (RESTRICTION ORIFICE) QQI (INDICATING COUNTER)  
PFR (PRESSURE RATIO RECORD) HCV (HAND CONTROL VALVE)  
KOI (TIME TOTALIZING INDICATOR)

INSTRUMENTATION IDENTIFICATION LETTERS (NOTE 3)				
FIRST LETTERS		SUCCEEDING LETTERS		
COLUMN 1	COLUMN 2	COLUMN 3	COLUMN 4	COLUMN 5
MEASURED/INITIATING VARIABLE	VARIABLE MODIFIER	READOUT/PASSIVE FUNCTION	OUTPUT/ACTIVE FUNCTION	FUNCTION MODIFIER
A ANALYSIS		ALARM		
B BURNER, COMBUSTION		USER'S CHOICE	USER'S CHOICE	USER'S CHOICE
C USER'S CHOICE			CONTROL	CLOSE
D USER'S CHOICE	DIFFERENCE, DIFFERENTIAL			DEVIATION
E VOLTAGE		SENSOR, PRIMARY ELEMENT		
F FLOW, FLOW RATE	RATIO			
G USER'S CHOICE		GLASS, GAUGE, VIEWING DEVICE		
H HAND				HIGH
I CURRENT		INDICATE		
J POWER		SCAN		
K TIME, SCHEDULE	TIME RATE OF CHANGE		CONTROL STATION	
L LEVEL		LIGHT		LOW
M USER'S CHOICE				MIDDLE, INTERMEDIATE
N USER'S CHOICE		USER'S CHOICE	USER'S CHOICE	USER'S CHOICE
O USER'S CHOICE		ORIFICE, RESTRICTION		OPEN
P PRESSURE		POINT (TEST CONNECTION)		
Q QUANTITY	INTEGRATE, TOTALIZE	INTEGRATE, TOTALIZE		
R RADIATION		RECORD		RUN
S SPEED, FREQUENCY	SAFETY		SWITCH	STOP
T TEMPERATURE			TRANSMIT	
U MULTIVARIABLE		MULTIFUNCTION	MULTIFUNCTION	
V VIBRATION, MECHANICAL ANALYSIS			VALVE, DAMPER, LOUVER	
W WEIGHT, FORCE		WELL, PROBE		
X UNCLASSIFIED	X-AXIS	ACCESSORY DEVICES, UNCLASSIFIED	UNCLASSIFIED	UNCLASSIFIED
Y EVENT, STATE, PRESENCE	Y-AXIS		AUXILIARY DEVICES	
Z POSITION, DIMENSION	Z-AXIS, SAFETY INSTRUMENTED SYSTEM		DRIVER, ACTUATOR, UNCLASSIFIED FINAL CONTROL ELEMENT	

Figure 1.7-3e: Piping and Instrumentation Diagram Legends

FUNCTION IDENTIFIERS									
ANALYSIS									
CALOR CO CO2 COL COMB COND DEN DEWPT DO	CALORIMETER CARBON MONOXIDE CARBON DIOXIDE COLOR COMBUSTION ELECTRICAL CONDUCTIVITY DENSITY DEW POINT DISSOLVED OXYGEN	GC H2 HC H2O H2S HUM %RH IR LC	GAS CHROMATOGRAPH HYDROGEN/HYDROGEN ANALYSIS HYDROCARBON WATER HYDROGEN SULFIDE HUMIDITY RELATIVE HUMIDITY INFRARED LIQUID CHROMATOGRAPH	MeOH MOIST MS NIR N2 NH3 NOx O2 OP	METHYL ALCOHOL MOISTURE MASS SPECTROMETER NEAR INFRARED NITROGEN AMMONIA OXIDES OF NITROGEN OXYGEN OPACITY	ORP PHASE pH REF RI SOx SP GR TC TDL	OXIDATION REDUCTION PHASE HYDROGEN ION REFRACTOMETER REFRACTIVE INDEX OXIDES OF SULFUR SPECIFIC GRAVITY THERMAL CONDUCTIVITY TUNABLE DIODE LASER	TDS THC TOC TURB UV VIS VISC	TOTAL DISSOLVED SOLIDS TOTAL HYDROCARBON TOTAL ORGANIC CARBON TURBIDITY ULTRAVIOLET VISIBLE LIGHT VISCOSITY
FLOW									
CFR CONE COR DOP DSO FLN	CONSTANT FLOW REGULATOR CONE CORIOLLIS DOPPLER DOPPLER SONIC FLOW NOZZLE	FLT LAM MAG OP OP-CT OP-CQ	FLOW RATE LAMINAR MAGNETIC ORIFICE PLATE CORNER TAPS CIRCLE QUADRANT	OP-E OP-FT OP-MH OP-P OP-VC PD PT	ECCENTRIC FLANGE TAPS MULTI-HOLE PIPE TAPS VENA CONTRACTA TAPS POSITIVE DISPLACEMENT PITOT TUBE	PV SNR SON TAR THER TTS	PITOT VENTURI SONAR SONIC TARGET THERMAL TRANSIT TIME SONIC	TUR US VENT VOR WDG	TURBINE ULTRASONIC VENTURI TUBE VORTEX SHEDDING WEDGE
LEVEL									
CAP d/p DI DISP	CAPACITANCE DIFFERENTIAL PRESSURE DIELECTRIC CONSTANT DISPLACER	DP GWR LSR MAG	DIFFERENTIAL PRESSURE GUIDED WAVE RADAR LASER MAGNETIC	MS NUC RADAR RES	MAGNETOSTRICTIVE NUCLEAR RADAR RESISTANCE	SON US	SONIC ULTRASONIC		
PRESSURE									
ABS AVG DRF	ABSOLUTE AVERAGE DRAFT	MAN P-V SG	MANOMETER PRESSURE-VACUUM STRAIN GAUGE	VAC	VACUUM				
TEMPERATURE									
BM IR RAD RP	BI-METAL INFRARED RADIATION RADIATION PYROMETER	RTD TC TCE	RESISTANCE TEMP. DETECTOR THERMOCOUPLE THERMOCOUPLE, TYPE E	TCJ TCK TCT	THERMOCOUPLE, TYPE J THERMOCOUPLE, TYPE K THERMOCOUPLE, TYPE T	THRM TMP TRAN	THERMISTOR THERMOPILE TRANSISTOR		
MISCELLANEOUS									
ANNUNCIATION		BURNER, COMBUSTION		OTHER			POSITION		
ALM ANN	ALARM ANNUNCIATOR	FR IGN IR UV	FLAME ROD IGNITER TELEVISION ULTRA VIOLET	CONC HOA L/R MOS MULTI OIL OX NR	CONCENTRIC HAND-OFF-AUTO LOCAL/REMOTE MAINTENANCE OVERRIDE SWITCH MULTIVARIABLE OVERLOAD OVERRIDE SWITCH NARROW RANGE	PB PC SMOKE SYNC TDR TEST VIBR WR	PUSHBUTTON PHOTOCELL SMOKE SYNCHRONIZATION TIME DELAY RELAY TEST VIBRATION WIDE RANGE	CAP EC IND LAS MAG MECH OPT RADAR	CAPACITANCE EDDY CURRENT INDUCTIVE LASER MAGNETIC MECHANICAL OPTICAL RADAR
QUANTITY		RADIATION		SPEED		WEIGHT, FORCE			
PE TOG	PHOTOELECTRIC TOGGLE	α β Y n RAD REM	ALPHA RADIATION BETA RADIATION GAMMA RADIATION NEUTRON RADIATION RADIATION ADSORBED DOSE ROENTGEN EQUIVALENT MAN	ACC EC PROX VEL	ACCELERATION EDDY CURRENT PROXIMITY VELOCITY	LC SG WS	LOAD CELL STRAIN GAUGE WEIGH SCALE		

INSTRUMENTATION LINE SYMBOLS: INSTRUMENT TO PROCESS AND EQUIPMENT CONNECTIONS	
SYMBOL	DESCRIPTION
	<ul style="list-style-type: none"><li>• INSTRUMENT CONNECTION TO PROCESS AND EQUIPMENT</li><li>• PROCESS IMPULSE LINES</li><li>• ANALYZER SAMPLE LINES</li></ul>
	<ul style="list-style-type: none"><li>• HEAT (COOL) TRACED IMPULSE OR SAMPLE LINE FROM PROCESS</li><li>• TYPE OF TRACING INDICATED BY (ET) ELECTRICAL, (ST) STEAM, (CW) CHILLED WATER, ETC</li></ul>
	<ul style="list-style-type: none"><li>• GENERIC INSTRUMENT CONNECTION TO PROCESS FLOW</li><li>• GENERIC INSTRUMENT CONNECTION TO EQUIPMENT</li></ul>
	<ul style="list-style-type: none"><li>• HEAT (COOL) TRACED GENERIC INSTRUMENT IMPULSE LINE</li><li>• PROCESS LINE OR EQUIPMENT MAY OR MAY NOT BE TRACED</li></ul>
	<ul style="list-style-type: none"><li>• HEAT(COOL) TRACED INSTRUMENT</li><li>• INSTRUMENT IMPULSE LINE MAY OR MAY NOT BE TRACED</li></ul>

LINE SYMBOLS	
LINE TYPE/SYMBOL	DESCRIPTION
IA 	<ul style="list-style-type: none"><li>• IA MAY BE REPLACED BY PA (PLANT AIR), NS (NITROGEN), OR GS (ANY GAS SUPPLY).</li><li>• INDICATE SUPPLY PRESSURE AS REQUIRED, E.G. PA-70 KPA, NS-150 PSIG, ETC.</li></ul>
ES 	<ul style="list-style-type: none"><li>• INSTRUMENT ELECTRIC POWER SUPPLY.</li><li>• INDICATE VOLTAGE AND TYPE AS REQUIRED, E.G. ES-220 VAC</li><li>• ES MAY BE REPLACED BY 24 VDC, 120 VAC, ETC.</li></ul>
HS 	<ul style="list-style-type: none"><li>• INSTRUMENT HYDRAULIC POWER SUPPLY.</li><li>• INDICATE PRESSURE AS REQUIRED, E.G. HS-70 PSIG.</li></ul>
	<ul style="list-style-type: none"><li>• ELECTRONIC OR ELECTRICAL CONTINUOUSLY VARIABLE OR BINARY SIGNAL.</li><li>• FUNCTIONAL DIAGRAM BINARY SIGNAL.</li></ul>
	<ul style="list-style-type: none"><li>• FUNCTIONAL DIAGRAM CONTINUOUSLY VARIABLE SIGNAL.</li><li>• ELECTRICAL SCHEMATIC LADDER DIAGRAM SIGNAL AND POWER RAILS.</li></ul>
	<ul style="list-style-type: none"><li>• FILLED THERMAL ELEMENT CAPILLARY TUBE.</li><li>• FILLED SENSING LINE BETWEEN PRESSURE SEAL AND INSTRUMENT.</li></ul>
	<ul style="list-style-type: none"><li>• GUIDED ELECTROMAGNETIC SIGNAL.</li><li>• GUIDED SONIC SIGNAL.</li><li>• FIBER OPTIC SIGNAL.</li></ul>
	<ul style="list-style-type: none"><li>• COMMUNICATION LINK AND SYSTEM BUS, BETWEEN DEVICES AND FUNCTIONS OF A SHARED DISPLAY, SHARED CONTROL SYSTEM.</li><li>• DCS, PLC, OR PC COMMUNICATION LINK AND SYSTEM BUS.</li></ul>
	<ul style="list-style-type: none"><li>• COMMUNICATION LINK OR BUS CONNECTING TWO OR MORE INDEPENDENT MICROPROCESSORS OR COMPUTER-BASED SYSTEMS.</li><li>• DCS-TO-DCS, DCS-TO-PLC, PLC-TO-PC, DCS-TO-FIELDBUS, ETC, CONNECTIONS.</li></ul>
	<ul style="list-style-type: none"><li>• COMMUNICATION LINK AND SYSTEM BUS, BETWEEN DEVICES AND FUNCTIONS OF A FIELDBUS SYSTEM.</li><li>• LINK FROM AND TO "INTELLIGENT" DEVICES.</li></ul>
	<ul style="list-style-type: none"><li>• COMMUNICATION LINK BETWEEN A DEVICE AND A REMOTE CALIBRATION ADJUSTMENT DEVICE OR SYSTEM.</li><li>• LINK FROM AND TO "SMART" DEVICES</li></ul>
	<ul style="list-style-type: none"><li>• MECHANICAL LINK OR CONNECTION.</li></ul>
	<ul style="list-style-type: none"><li>• PRIMARY LINE</li></ul>
	<ul style="list-style-type: none"><li>• SECONDARY LINE</li></ul>
	<ul style="list-style-type: none"><li>• STEAM TRACE LINE</li></ul>
	<ul style="list-style-type: none"><li>• ELECTRICAL TRACE LINE</li></ul>
	<ul style="list-style-type: none"><li>• PNEUMATIC SIGNAL</li></ul>
	<ul style="list-style-type: none"><li>• HYDRAULIC SIGNAL.</li></ul>
R 	<ul style="list-style-type: none"><li>• REFRIGERANT</li></ul>
	<ul style="list-style-type: none"><li>• OTHER SYSTEMS</li></ul>



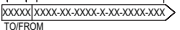

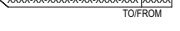






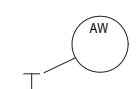

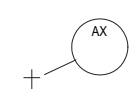


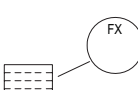




























EQUIPMENT DESCRIPTIONS (NOTE 1)			BOUNDARY IDENTIFICATION		MISCELLANEOUS IDENTIFICATIONS	
<u>AIR COOLER</u> TUBE DP/DT: PSIG/°F TUBE OP/OT : PSIG/°F DESIGN DUTY: BTU/HR CONFIGURATION: MOTOR NAME PLATE: HP	<u>HEAT EXCHANGER</u> DESIGN DUTY: BTU/HR DUTY CYCLE: % SHELL DP/DT: PSIG/°F SHELL OP/OT: PSIG/°F TUBE DP/DT: PSIG/°F TUBE OP/OT: PSIG/°F	<u>PRESSURE VESSEL</u> DP/DT: PSIG/°F OP/OT: PSIG/°F SIZE: ID X T-T CAPACITY: GAL	LIMITS OF UPSTREAM PIPE CLASS OR LINE NUMBER	LIMITS OF DOWNSTREAM PIPE CLASS OR LINE NUMBER		
<u>COMPRESSOR</u> DP/DT: PSIG/°F OP/OT: PSIG/°F CAPACITY: FT³/MIN DUTY CYCLE: % MOTOR NAME PLATE: HP	<u>TANK</u> DP/DT: PSIG/°F OP/OT: PSIG/°F CAPACITY: FT³/MIN SIZE: ID X HEIGHT: FT CAPACITY: GAL	<u>PUMP</u> CAPACITY: GPM DUTY CYCLE: % DESIGN HEAD: FT MOTOR NAME PLATE: HP DP/DT: PSIG/°F	AREA A			
<u>FILTERS</u> DP/DT: PSIG/°F OP/OT: PSIG/°F SIZE: ID X T-T PARTICLE SIZE: MICRON	<u>AHU/ ACU/ FCU</u> AIR FLOW: CFM OP/OT: PSIG/°F COOLING CAPACITY: BTU/ HR HEATING CAPACITY: BTU/ HR (KW) MOTOR NAMEPLATE: HP	<u>CHILLER</u> CAPACITY: TONS DUTY CYCLE: % EVAP FLOW: GPM EVAP EWT/ LWT: °F/ °F COND FLOW: GPM COND EWT/ LWT: °F/ °F POWER: kW	DRAWING CONNECTIONS		 VENDOR PACKAGE	
<u>FAN</u> AIR FLOW: CFM DUTY CYCLE: % MOTOR NAMEPLATE: HP	<u>UNIT HEATER</u> HEATING CAPACITY: kW		CONNECTOR I.D. NUMBER	DESTINATION PAGE NUMBER	 NON-VENDOR PACKAGE OR MODULE PACKAGE	
						
						
						
						

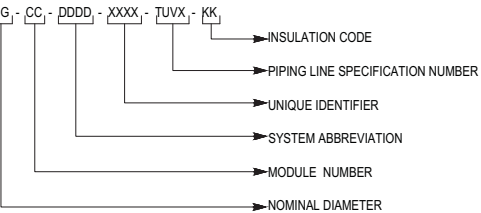
Figure 1.7-3f: Piping and Instrumentation Diagram Legends

PRIMARY ELEMENTS		AUXILIARY AND ACCESSORY DEVICES	
FLOW		ANALYSIS	
	GENERIC ORIFICE PLATE		SAMPLE INSERT PROBE, FLANGED / SAMPLE WELL, FLANGED
	QUICK-CHANGE ORIFICE PLATE		SAMPLE ANALYSIS ACCESSORY, FLANGED
	CONCENTRIC CIRCLE ORIFICE PLATE	FLOW	
	ECCENTRIC CIRCLE ORIFICE PLATE		FLOW STRAIGHTENING VANES / FLOW CONDITIONING ELEMENT
	CIRCLE QUADRANT ORIFICE PLATE		INSTRUMENT PURGE/FLUSHING FLUID OR DEVICES
	MULTI-HOLE ORIFICE PLATE	PRESSURE	
	GENERIC VENTURI TUBE, FLOW NOZZLE, OR FLOW TUBE (NOTE 1)		DIAPHRAGM PRESSURE SEAL, GENERIC CONNECTION / DIAPHRAGM CHEMICAL SEAL, GENERIC CONNECTION
	VENTURI TUBE		DIAPHRAGM PRESSURE SEAL, WELDED/ DIAPHRAGM CHEMICAL SEAL, WELDED
	FLOW NOZZLE	TEMPERATURE	
	FLOW TUBE		THERMOWELL, FLANGED (NOTE 2) (BUBBLE MAY BE OMITTED IF CONNECTED TO ANOTHER INSTRUMENT)
	INTEGRAL ORIFICE PLATE		
	STANDARD PITOT TUBE		
	AVERAGING PITOT TUBE		
ANALYSIS		LEVEL	
	SINGLE ELEMENT SENSING PROBE		INTERNALLY MOUNTED BALL FLOAT (MAY BE INSTALLED THROUGH TOP OF VESSEL)
	DUAL ELEMENT SENSING PROBE		INTERNALLY MOUNTED DISPLACER
	FIBEROPTIC SENSING PROBE		RADIATION SINGLE POINT
BURNER			RADIATION, MULTI-POINT OR CONTINUOUS
	ULTRAVIOLET FLAME DETECTOR / TELEVISION FLAME MONITOR		PRIMARY ELEMENT (E.G. DIP TUBE) AND STILLING WELL (MAY BE INSTALLED THROUGH SIDE OF VESSEL) (MAY BE INSTALLED WITHOUT STILLING WELL)
	FLAME ROD / FLAME DETECTOR		FLOAT WITH GUIDE WIRES (LOCATION READOUT SHOULD BE NOTED: AT GRADE, AT TOP, OR ACCESSIBLE FROM A LADDER) (GUIDE WIRES MAY BE OMITTED)
PRESSURE			INSERT PROBE (MAY BE INSTALLED THROUGH TOP OF VESSEL)
	STRAIN GAGE OR OTHER ELECTRONIC SENSOR (NOTE 2) (BUBBLE MAY BE OMITTED IF CONNECTED TO ANOTHER INSTRUMENT)		RADAR
TEMPERATURE			
	GENERIC ELEMENT WITHOUT THERMOWELL (NOTE 2) (BUBBLE MAY BE OMITTED IF CONNECTED TO ANOTHER INSTRUMENT)		

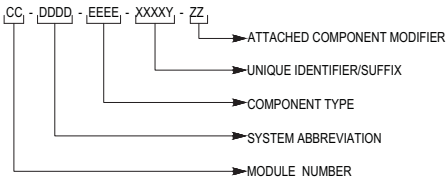
NOTES:

1. ABBREVIATIONS FROM FUNCTION IDENTIFIERS TABLE ON FIGURE 1.7-3e SHALL BE USED IF MORE THAN ONE ELEMENT TYPE APPEARS ON THE DRAWING.
2. AN ABBREVIATION FROM THE FUNCTION IDENTIFIERS TABLE ON FIGURE 1.7-3e SHOULD BE USED TO IDENTIFY THE ELEMENT TYPE.

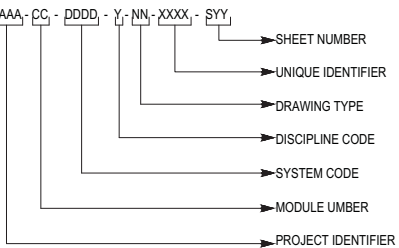
LINE NUMBERING FORMAT



COMPONENT IDENTIFICATION SYSTEM



ENGINEERING DRAWING NUMBERING



## 1.8 Interfaces with Certified Design

This section addresses interface requirements between the NuScale Power Plant certified design and the site-specific design provided in the combined license (COL) application. Section 1.2 identifies the structures, systems, and components that are included in the certified design. Figure 1.2-2 provides a representation of the overall facility and Figure 1.2-2 provides the general boundaries between the certified design and site-specific design.

Table 1.8-1 identifies the interfaces between the NuScale certified design and the site-specific design. There are two types of interface requirements described:

- CDI: Conceptual design information that is provided for the non-certified portion of the plant to facilitate review of the certified design and to confirm the adequacy of identified interface requirements.
- COL: NuScale design assumptions related to site-specific design elements that are the responsibility of the COL applicant. This type of interface is identified as a COL information item.

### 1.8.1 Combined License Information Items

Information that must be provided in order to license and operate a site-specific NuScale Power Plant, but is not included in the certified design, is identified throughout the Final Safety Analysis Report as COL information items. Table 1.8-2 lists the COL information items, includes the COL information item text and identifies the section where the information item is located. The COL applicant addresses each COL information item in the section where it is located.

### 1.8.2 Departures

COL Item 1.8-1: A COL applicant that references the NuScale Power Plant design certification will provide a list of departures from the certified design.



**Table 1.8-1: Summary of NuScale Certified Design Interfaces with Remainder of Plant**

System, Structure, or Component	Interface Type	FSAR Section
Turbine Generator Buildings	CDI	1.2.2
Annex Building	CDI	1.2.2
Cooling towers, pump houses, and associated structures, systems, and components (e.g., cooling tower basin, circulating water pumps, cooling tower fans, chemical treatment building, etc.)	CDI	1.2.2, 10.4.5
Security Buildings	CDI	1.2.2
Central Utility Building	CDI	1.2.2
Diesel Generator Buildings	CDI	1.2.2
Offsite power transmission system, main switchyard, and transformer area	CDI	8.2
Auxiliary AC power system	CDI	8.3.1
Site cooling water system	CDI	9.2.7
Circulating water system	CDI	10.4.5
Grounding and lightning protection system	CDI	8.3.1
Potable and sanitary water systems	COL	9.2.4
Resin tanks for the condensate polishing system	COL	10.4
Site drainage system	COL	N/A
Raw water system	COL	9.2.7
Site-specific design parameters, geographic and demographic characteristics, meteorological characteristics, nearby industrial, transportation, and military facilities, hydrologic characteristics, geology, seismology, and geotechnical characteristics, weather conditions and site topography, flooding	COL	Table 2.0-1, 2.1, 2.2, 2.3, 2.4, 2.5, 3.3, 3.4
Site-specific communications	COL	9.5.2
Turbine generators	COL	3.5-1
Diesel generators	COL	3.5-1
Operational Support Center	COL	13.3

**Table 1.8-2: Combined License Information Items**

<b>Item No.</b>	<b>Description of COL Information Item</b>	<b>Section</b>
COL Item 1.1-1:	A COL Applicant that references the NuScale Power Plant design certification will identify the site-specific plant location.	1.1
COL Item 1.1-2:	A COL Applicant that references the NuScale Power Plant design certification will provide the schedules for completion of construction and commercial operation of each power module.	1.1
COL Item 1.4-1:	A COL Applicant that references the NuScale Power Plant design certification will identify the prime agents or contractors for the construction and operation of the nuclear power plant.	1.4
COL Item 1.7-1:	A COL Applicant that references the NuScale Power Plant design certification will provide site-specific diagrams and legends, as applicable.	1.7
COL Item 1.7-2:	A COL Applicant that references the NuScale Power Plant design certification will list additional site-specific P&IDs and legends as applicable.	1.7
COL Item 1.8-1:	A COL Applicant that references the NuScale Power Plant design certification will provide a list of departures from the certified design.	1.8
COL Item 1.9-1:	A COL Applicant that references the NuScale Power Plant design certification will review and address the conformance with regulatory criteria in effect six months before the docket date of the COL application for the site-specific portions and operational aspects of the facility design.	1.9
COL Item 1.10-1:	A COL Applicant that references the NuScale Power Plant design certification will evaluate the potential hazards resulting from construction activities of the new NuScale facility to the safety-related and risk significant structures, systems, and components of existing operating unit(s) and newly constructed operating unit(s) at the co-located site per 10 CFR 52.79(a)(31). The evaluation will include identification of any management and administrative controls necessary to eliminate or mitigate the consequences of potential hazards and demonstration that the limiting conditions for operation of an operating unit would not be exceeded. This COL item is not applicable for construction activities (build-out of the facility) at an individual NuScale Power Plant with operating NuScale Power Modules.	1.10
COL Item 2.0-1:	A COL Applicant that references the NuScale Power Plant design certification will demonstrate that site-specific characteristics are bounded by the design parameters specified in Table 2.0-1. If site-specific values are not bounded by the values in Table 2.0-1, the COL applicant will demonstrate the acceptability of the site-specific values in the appropriate sections of its combined license application.	2.0
COL Item 2.1-1:	A COL Applicant that references the NuScale Power Plant design certification will describe the site geographic and demographic characteristics.	2.1
COL Item 2.2-1:	A COL Applicant that references the NuScale Power Plant design certification will describe nearby industrial, transportation, and military facilities. The COL applicant will demonstrate that the design is acceptable for each potential accident, or provide site-specific design alternatives.	2.2
COL Item 2.3-1:	A COL Applicant that references the NuScale Power Plant design certification will describe the site-specific meteorological characteristics for Section 2.3.1 through Section 2.3.5, as applicable.	2.3
COL Item 2.4-1:	A COL Applicant that references the NuScale Power Plant design certification will investigate and describe the site-specific hydrologic characteristics for Section 2.4.1 through Section 2.4.14, as applicable.	2.4
COL Item 2.5-1:	A COL Applicant that references the NuScale Power Plant design certification will describe the site-specific geology, seismology, and geotechnical characteristics for Section 2.5.1 through Section 2.5.5, below.	2.5
COL Item 3.2-1:	A COL Applicant that references the NuScale Power Plant design certification will update Table 3.2-1 to identify the classification of site-specific SSC.	3.2
COL Item 3.3-1:	A COL Applicant that references the NuScale Power Plant design will confirm that nearby structures exposed to severe and extreme (tornado and hurricane) wind loads will not collapse and adversely affect the RXB or Seismic Category I portion of the CRB.	3.3
COL Item 3.4-1:	A COL Applicant that references the NuScale Power plant design certification will confirm the final location of structures, systems, and components subject to flood protection and final routing of piping.	3.4

**Table 1.8-2: Combined License Information Items (Continued)**

<b>Item No.</b>	<b>Description of COL Information Item</b>	<b>Section</b>
COL Item 3.4-2:	A COL Applicant that references the NuScale Power plant design certification will identify the selected mitigation strategy for each room containing structures, systems, and components subject to flood protection.	3.4
COL Item 3.4-3:	A COL Applicant that references the NuScale Power plant design certification will develop an inspection and maintenance program to ensure that each water-tight door, penetration seal, or other “degradable” measure remains capable of performing its intended function.	3.4
COL Item 3.4-4:	A COL Applicant that references the NuScale Power plant design certification will confirm that site-specific tanks or water sources are placed in locations where they cannot cause flooding in the Reactor Building or Control Building.	3.4
COL Item 3.4-5:	A COL Applicant that references the NuScale Power Plant design certification will determine the extent of waterproofing and dampproofing needed for the underground portion of the Reactor Building and Control Building based on site-specific conditions.	3.4
COL Item 3.4-6:	A COL Applicant that references the NuScale Power Plant design certification will confirm that nearby structures exposed to external flooding will not collapse and adversely affect the RXB or Seismic Category I portion of the CRB.	3.4
COL Item 3.5-1:	A COL Applicant that references the NuScale Power Plant certified design will provide a missile analysis for the turbine generator which demonstrates that the probability of a turbine generator producing a low trajectory turbine missile is less than 10 <sup>-5</sup> .	3.5
COL Item 3.5-2:	A COL Applicant that references the NuScale Power Plant certified design will address the effect of turbine missiles from nearby or co-located facilities.	3.5
COL Item 3.6-1:	A COL Applicant that references the NuScale Power Plant design certification will determine if a high-energy line break or moderate energy line break outside of the Reactor Building, Control Building, or Radioactive Waste Building could affect site-specific essential equipment (or result in a transient or other off-normal event in a second module), and install protection as necessary.	3.6
COL Item 3.6-2:	A COL Applicant that references the NuScale Power Plant design certification will finalize the stress analysis of the high-energy lines in the Reactor Pool Bay and update Table 3.6-2, Figure 3.6-12, Figure 3.6-13, Figure 3.6-14 and Figure 3.6-15 as appropriate.	3.6
COL Item 3.6-3:	A COL Applicant that references the NuScale Power Plant design certification will finalize the stress analysis and the environmental analysis of the high-energy lines outside the reactor pool bay. This includes the identification of any new detection and auto-isolation functions for mitigating an auxiliary boiler high-energy line break. The COL Applicant will update Table 3.6-2, Figure 3.6-16 and Figure 3.6-17 as appropriate.	3.6
COL Item 3.6-4:	A COL Applicant that references the NuScale Power Plant design certification will perform stress analysis for high energy lines outside the NPM if needed to identify and mitigate the consequences of potential breaks.	3.6
COL Item 3.7-1:	A COL Applicant that references the NuScale Power Plant design certification will describe the site-specific SSE.	3.7
COL Item 3.7-2:	A COL Applicant that references the NuScale Power Plant design certification will provide site-specific time histories. In addition to the above criteria for cross correlation coefficients, time step and earthquake duration, strong motion durations, comparison to response spectra and power spectra density, the applicant will also confirm that site-specific ratios V/A and AD/V <sup>2</sup> (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.	3.7
COL Item 3.7-3:	A COL Applicant that references the NuScale Power Plant Standard design will	3.7
COL Item 3.7-4:	A COL Applicant that references the NuScale Power Plant design certification will confirm that nearby structures exposed to a site-specific safe shutdown earthquake (SSE) will not collapse and adversely affect the RXB or Seismic Category I portion of the CRB.	3.7
COL Item 3.7-5:	A COL Applicant that references the NuScale Power Plant design certification will perform a SSI analysis of the RXB and the CRB using the NuScale SASSI2010 models for those structures.	3.7
COL Item 3.7-6:	A COL Applicant that references the NuScale Power Plant design certification will perform a SSSI analysis that includes the RXB, CRB, RWB and both Turbine Generator Buildings.	3.7

**Table 1.8-2: Combined License Information Items (Continued)**

<b>Item No.</b>	<b>Description of COL Information Item</b>	<b>Section</b>
COL Item 3.7-7:	A COL Applicant that references the NuScale Power Plant design certification will provide a seismic monitoring system and a seismic monitoring program that satisfies RG 1.12 "Nuclear Power Plant Instrumentation for Earthquakes," Rev. 2 (or later) and RG 1.166 "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-earthquake Actions," Rev. 0 (or later). This information is to be provided as noted below.	3.7
COL Item 3.7-8:	A COL Applicant that references the NuScale Power Plant design certification will identify the implementation milestone for the seismic monitoring program.	3.7
COL Item 3.8-1:	A COL Applicant that references the NuScale Power Plant design certification will describe the site-specific program for monitoring and maintenance of the Seismic Category I structures in accordance with the requirements of 10 CFR 50.65 as discussed in RG 1.160. Monitoring is to include below grade walls, groundwater chemistry if needed, base settlements and differential displacements.	3.8
COL Item 3.8-2:	A COL Applicant that references the NuScale Power Plant design certification will confirm that the site independent RXB and CRB are acceptable for use at the designated site.	3.8
COL Item 3.9-1:	A COL Applicant that references the NuScale Power Plant design certification will submit the results from the comprehensive vibration assessment program for the NuScale Power Module, in accordance with Regulatory Guide 1.20.	3.9
COL Item 3.9-2:	A COL Applicant that references the NuScale Power Plant design certification will develop design specifications and design reports in accordance with the requirements outlined under American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III (Reference 3.9-1). A COL applicant will address any known issues through the reactor vessel internals reliability programs (i.e. Comprehensive Vibration Assessment Program, steam generator programs, etc.) in regards to known aging degradation mechanisms such as those addressed in Section 4.5.2.1.	3.9
COL Item 3.9-3:	A COL Applicant that references the NuScale Power Plant design certification will provide a summary of reactor core support structure maximum total stress, deformation, and cumulative usage factor values for each component and each operating condition in conformance with ASME Boiler and Pressure Vessel Code Section III Subsection NG.	3.9
COL Item 3.9-4:	A COL Applicant that references the NuScale Power Plant design certification will submit a Preservice Testing program for valves as required by 10 CFR 50.55a.	3.9
COL Item 3.9-5:	A COL Applicant that references the NuScale Power Plant design certification will establish an Inservice Testing program in accordance with ASME OM Code and 10 CFR 50.55a.	3.9
COL Item 3.9-6:	A COL Applicant that references the NuScale Power Plant design certification will identify any site-specific valves and provide inservice testing in accordance with the latest endorsed ASME Code with addenda of the ASME OM Code incorporated by reference by 10 CFR 50.55a 18 months prior to the date for initial fuel load.	3.9
COL Item 3.9-7:	Where the NuScale definition of Modes of Operation differ from those defined in the ASME OM Code, an Alternative may be provided to reconcile the terminology. A COL Applicant that reference the NuScale Power Plant design certification will generate any relief request(s) needed as part of the Inservice Testing Program Document.	3.9
COL Item 3.10-1:	A COL Applicant that references the NuScale Power Plant design certification will develop and maintain a site-specific seismic and dynamic qualification program.	3.10
COL Item 3.10-2:	A COL Applicant that references the NuScale Power Plant design certification will develop the equipment qualification database and ensure equipment qualification record files are created for the structures, systems, and components that require seismic qualification.	3.10
COL Item 3.10-3:	A COL Applicant that references the NuScale Power Plant design certification will submit an implementation program for Nuclear Regulatory Commission approval prior to the installation of the equipment that requires seismic qualification.	3.10
COL Item 3.11-1:	A COL Applicant that references the NuScale Power Plant design certification will submit a full description of the Environmental Qualification Program and milestones and completion dates for program implementation.	3.11

**Table 1.8-2: Combined License Information Items (Continued)**

<b>Item No.</b>	<b>Description of COL Information Item</b>	<b>Section</b>
COL Item 3.11-2:	A COL Applicant that references the NuScale Power Plant design certification will develop the equipment qualification database and ensure equipment qualification record files are created for the structures, systems, and components that require environmental qualification.	3.11
COL Item 3.12-1:	A COL Applicant that references the NuScale Power Plant design certification may use a piping analysis program other than the programs listed in Section 3.12.4.1; however, the applicant will implement a benchmark program using the models for NuScale Power Plant standard design.	3.12
COL Item 3.12-2:	A COL Applicant that references the NuScale Power Plant design certification will confirm that the site-specific seismic response is within the parameters specified in Section 3.7. A COL applicant may perform a site-specific piping stress analysis in accordance with the methodologies described in this section, as appropriate.	3.12
COL Item 3.13-1:	A COL Applicant that references the NuScale Power Plant design certification will provide an inservice inspection program for ASME Class 1, 2 and 3 threaded fasteners or describe the implementation program, including milestones, completion dates and expected conclusions. The program will identify the applicable edition and addenda of ASME BPVC, Section XI and ensure compliance with 10 CFR 50.55a.	3.13
COL Item 5.2-1:	A COL Applicant that references the NuScale Power Plant design certification and uses a later Code edition or addenda other than American Society of Mechanical Engineers Boiler and Pressure Vessel Code 2013 will perform and document with a code reconciliation an American Society of Mechanical Engineers Design Report as required by American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Paragraph NCA-3554, "Modification of Documents and Reconciliation With Design Report."	5.2
COL Item 5.2-2:	A COL Applicant that references the NuScale Power Plant design certification will provide a certified Overpressure Protection Report in compliance with American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Subarticles NB-7200 and NC-7200 to demonstrate the reactor coolant pressure boundary and secondary system are designed with adequate overpressure protection features.	5.2
COL Item 5.2-3:	A COL Applicant that references the NuScale Power Plant design certification will establish measures to control the on-site cleaning of RPV and pressure retaining components associated with the RCPB during construction.	5.2
COL Item 5.2-4:	A COL Applicant that references the NuScale Power Plant design certification will develop and implement a Strategic Water Chemistry Plan. The Strategic Water Chemistry Plan will provide the optimization strategy for maintaining primary coolant chemistry and provide the basis for requirements for sampling and analysis frequencies, and corrective actions for control of primary water chemistry consistent with the Electric Power Research Institute Pressurized Water Reactor Primary Water Chemistry Guidelines.	5.2
COL Item 5.2-5:	A COL Applicant that references the NuScale Power Plant design certification will develop and implement a Boric Acid Control Program that includes: inspection elements to ensure the integrity of the reactor coolant pressure boundary components for subsequent service, the type of visual or other nondestructive inspections to be performed, and the required inspection frequency.	5.2
COL Item 5.2-6:	A COL Applicant that references the NuScale Power Plant design certification will develop site-specific preservice examination, inservice inspection, and inservice testing program plans in accordance with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and will establish implementation milestones. If applicable, a COL Applicant that references the NuScale Power Plant design certification will identify the implementation milestone for the augmented inservice inspection program. The COL applicant will identify the applicable edition of the American Society of Mechanical Engineers Code utilized in the program plans consistent with the requirements of 10 CFR 50.55a	5.2

**Table 1.8-2: Combined License Information Items (Continued)**

<b>Item No.</b>	<b>Description of COL Information Item</b>	<b>Section</b>
COL Item 5.2-7:	A COL Applicant that references the NuScale Power Plant design certification will establish plant-specific procedures that specify operator actions for identifying, monitoring, trending, and locating reactor coolant system leakage in response to prolonged low leakage conditions that exist above normal leakage rates and below the technical specification limits. The objective of the methods of detecting and locating the reactor coolant pressure boundary leak will be to provide the operator sufficient time to take actions before the plant technical specification limits are reached.	5.2
COL Item 5.3-1:	A COL Applicant that references the NuScale Power Plant design certification will establish measures to control the onsite cleaning of the RPV during construction in accordance with RG 1.28.	5.3
COL Item 5.3-2:	A COL Applicant that references the NuScale Power Plant design certification will develop operating procedures to ensure that transients will not be more severe than those for which the reactor design adequacy had been demonstrated.	5.3
COL Item 5.4-1:	A COL Applicant that references the NuScale Power Plant design certification will develop and implement a Steam Generator Program for periodic monitoring of the degradation of steam generator components to ensure that steam generator tube integrity is maintained. The Steam Generator Program will be based on NEI 97-06, "Steam Generator Program Guidelines," Revision 3 and applicable EPRI steam generator guidelines. The elements of the program will include: assessment of degradation, tube inspection requirements, tube integrity assessment, tube plugging, primary-to-secondary leakage monitoring, primary and secondary side water chemistry control, foreign material exclusion, loose parts management, contractor oversight, self-assessment, and reporting.	5.4
COL Item 6.2-1:	A COL Applicant that references the NuScale Power Plant design certification will develop a "Containment Leakage Rate Testing Program" which will identify which Option is to be implemented under 10 CFR 50, Appendix J. Option A defines a prescriptive-based testing approach whereas Option B defines a performance-based testing program.	6.2
COL Item 6.3-1:	A COL Applicant that references the NuScale Power Plant design certification will describe a containment cleanliness program that limits debris within containment. The program should contain the following elements:	6.3
COL Item 6.4-1:	A COL Applicant that references the NuScale Power Plant design certification will comply with RG 1.78 Revision 1, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."	6.4
COL Item 6.4-2:	A COL Applicant that references the NuScale Power Plant design certification will specify operator training and qualification in the use of self-contained portable breathing apparatus.	6.4
COL Item 6.4-3:	A COL Applicant that references the NuScale Power Plant design certification will specify the technical resources to be stored within the CRE.	6.4
COL Item 6.4-4:	A COL Applicant that references the NuScale Power Plant design certification will specify food, water, and medical supplies to be stored within the CRE.	6.4
COL Item 6.4-5:	A COL Applicant that references the NuScale Power Plant design certification will specify testing and inspection requirements for the CRHS, including CRE integrity testing.	6.4
COL Item 6.6-1:	A COL Applicant that references the NuScale Power Plant design certification will implement an Inservice Inspection/Testing Program in accordance with 10 CFR 50.55a(f).	6.6
COL Item 6.6-2:	A COL Applicant that references the NuScale Power Plant design certification will develop a preservice inspection and Inservice Inspection Program plans in accordance with Section XI of the ASME Code, and will establish the implementation milestones for the program. The COL applicant will identify the applicable edition of the ASME Code utilized in the program plan consistent with the requirements of 10 CFR 50.55a. The COL applicant will, if needed, address the use of a single ISI Program for multiple NPMs, including any Alternative to the Code that may be necessary to implement such an ISI Program.	6.6
COL Item 7.2-1:	A COL Applicant that references the NuScale Power Plant design certification is responsible for the implementation of the life cycle processes for the operation phase for the instrumentation and controls systems, as defined in IEEE Standard 1074-2006 and IEEE Standard 1012-2004.	7.2

**Table 1.8-2: Combined License Information Items (Continued)**

<b>Item No.</b>	<b>Description of COL Information Item</b>	<b>Section</b>
COL Item 7.2-2:	A COL Applicant that references the NuScale Power Plant design certification is responsible for the implementation of the life cycle processes for the maintenance phase for the instrumentation and controls systems, as defined in IEEE Standard 1074-2006 and IEEE Standard 1012-2004.	7.2
COL Item 7.2-3:	The NuScale Digital I&C Software Configuration Management Plan provides guidance for the retirement and removal of a software product from use. A COL Applicant that references the NuScale Power Plant design certification is responsible for the implementation of the life cycle processes for the retirement phase for the instrumentation and controls systems, as defined in IEEE Standard 1074-2006 and IEEE Standard 1012-2004. The NuScale Digital I&C Software Configuration Management Plan provides guidance for the retirement and removal of a software product from use.	7.2
COL Item 8.2-1:	The design of the switchyard and the connections to an offsite power system are site-specific and are the responsibility of the combined license (COL) applicant. A COL Applicant that references the NuScale Power Plant design certification will describe the site-specific switchyard layout and design, including offsite power connections, control and indication, characteristics of circuit breakers and buses, protective relaying, power supplies, lightning and grounding protection equipment, and conformance with GDC 5.	8.2
COL Item 8.2-2:	A COL Applicant that references the NuScale Power Plant design certification will describe the site-specific offsite power connection and grid stability studies, including the effects of grid contingencies such as the loss of the largest operating unit on the grid, the loss of one NuScale Power Module, and the loss of the full complement of NuScale Power Modules (up to 12). The study will be performed in accordance with the applicable Federal Energy Regulatory Commission, North American Electric Reliability Corporation, and transmission system operator requirements, including communication agreements and protocols.	8.2
COL Item 8.2-3:	A COL Applicant that references the NuScale Power Plant design certification will describe the testing of the switchyard and the connections to an offsite power system, if provided, consistent with Regulatory Guide 1.68, Revision 3.	8.2
COL Item 8.3-1:	A COL Applicant that references the NuScale Power Plant design certification will describe the site-specific location, type, and design of the power source to be used as the auxiliary alternating current power system.	8.3
COL Item 8.3-2:	A COL Applicant that references the NuScale Power Plant design certification will describe the design of the site-specific electrical heat tracing system.	8.3
COL Item 8.3-3:	A COL Applicant that references the NuScale Power Plant design certification will describe the design of the site-specific plant grounding grid and lightning protection network.	8.3
COL Item 9.1-1:	A COL Applicant that references the NuScale Power Plant design certification will develop plant programs and procedures for safe operations during handling and storage of new and spent fuel assemblies, including criticality control.	9.1
COL Item 9.1-2:	A COL Applicant that references the NuScale Power Plant design certification will demonstrate that a NRC-licensed cask can be lowered into the dry dock and used to remove spent fuel assemblies from the plant.	9.1
COL Item 9.1-3:	A COL Applicant that references the NuScale Power Plant design certification will develop procedures related to the transfer of spent fuel to a transfer cask.	9.1
COL Item 9.1-4:	A COL Applicant that references the NuScale Power Plant design certification will provide the periodic testing plan for fuel handling equipment.	9.1
COL Item 9.1-5:	COL Item - The COL Applicant that references the NuScale Power Plant design certification will describe the process for handling and receipt of new NuScale Power Modules. <ul style="list-style-type: none"> <li>• operating and maintenance procedures</li> <li>• inspection and test plans</li> <li>• personnel qualification and operator training</li> </ul>	9.1
COL Item 9.1-6:	The COL Applicant that references the NuScale Power Plant design certification will provide a design for a spent fuel cask.	9.1
COL Item 9.1-7:	The COL Applicant that references the NuScale Power Plant design certification will provide a description of the program governing heavy loads handling. The program should address	9.1

**Table 1.8-2: Combined License Information Items (Continued)**

<b>Item No.</b>	<b>Description of COL Information Item</b>	<b>Section</b>
COL Item 9.2-1:	A COL Applicant that references the NuScale Power Plant design certification will select the appropriate chemicals for the reactor component cooling water system based on site-specific water quality and materials requirements.	9.2
COL Item 9.2-2:	A COL Applicant that references the NuScale Power Plant design certification will describe the source and pre-treatment methods of potable water for the site, including the use of associated pumps and storage tanks.	9.2
COL Item 9.2-3:	A COL Applicant that references the NuScale Power Plant design certification will describe the method for sanitary waste storage and disposal, including associated treatment facilities.	9.2
COL Item 9.2-4:	A COL Applicant that references the NuScale Power Plant design certification will provide details on the prevention of long-term corrosion and organic fouling in the site cooling water system.	9.2
COL Item 9.2-5:	A COL Applicant that references the NuScale Power Plant design certification will identify the site specific water source and provide a water treatment system that is capable of producing water that meets the plant water chemistry requirements.	9.2
COL Item 9.3-1:	A COL Applicant that references the NuScale Power Plant design certification will submit a leakage control program, including an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems.	9.3
COL Item 9.3-2:	A COL Applicant that references the NuScale Power Plant design certification will develop the post-accident sampling contingency plans for using the process sampling system and the CES off-line radiation monitor to obtain reactor coolant and containment atmosphere samples.	9.3
COL Item 9.4-1:	A COL Applicant that references the NuScale Power Plant design certification will specify a periodic testing and inspection program for the normal control room HVAC system.	9.4
COL Item 9.4-2:	A COL Applicant that references the NuScale Power Plant design certification will specify periodic testing and inspection requirements for the Reactor Building heating ventilation and air conditioning system in accordance with Regulatory Guide 1.140.	9.4
COL Item 9.4-3:	A COL Applicant that references the NuScale Power Plant design certification will specify periodic testing and inspection requirements for the Radioactive Waste Building HVAC system.	9.4
COL Item 9.4-4:	A COL Applicant that references the NuScale Power Plant design certification will specify periodic testing and inspection requirements for the TBVS.	9.4
COL Item 9.5-1:	A COL Applicant that references the NuScale Power Plant design certification will provide a description of the offsite communication system, how that system interfaces with the onsite communications system, as well as how continuous communications capability is maintained to ensure effective command and control with onsite and offsite resources during both normal and emergency situations.	9.5
COL Item 9.5-2:	A COL Applicant that references the NuScale Power Plant design certification will determine the location for the security power equipment within a vital area in accordance with 10 CFR 73.55(e)(9)(vi)(B).	9.5
COL Item 10.2-1:	A COL Applicant that references the NuScale Power Plant design certification will identify the specific turbine vendor and provide a description of the turbine and control system design.	10.2
COL Item 10.2-2:	A COL Applicant that references the NuScale Power Plant design certification will describe how the functional requirements for turbine overspeed are met for the vendor specific turbine design and will provide a schematic of the turbine control system and protection systems.	10.2



**Table 1.8-2: Combined License Information Items (Continued)**

Item No.	Description of COL Information Item	Section
COL Item 10.2-3:	<p>A COL Applicant that references the NuScale Power Plant design certification will perform an evaluation of the probability of turbine missile generation. The report provides a calculation of the probability of turbine missile generation using established methods and industry guidance applicable to the fabrication technology employed. The analysis is a comprehensive report containing a description of turbine fabrication methods, material quality and properties, and required maintenance and inspections that addresses:</p> <ul style="list-style-type: none"> <li>a) the calculated probability of turbine missile generation from material and overspeed related failures based on as-built rotor and blade designs and asbuilt material properties (as determined in certified testing and nondestructive examination).</li> <li>b) maximum anticipated speed resulting from a loss of load, assuming normal control system function without trip.</li> <li>c) overspeed basis and overspeed protection trip setpoints.</li> <li>d) discussion of the design and structural integrity of turbine rotors.</li> <li>e) an analysis of potential degradation mechanisms (e.g., stress corrosion cracking, pitting, low-cycle fatigue, corrosion fatigue, erosion and erosioncorrosion), and any specific maintenance or operating requirements necessary for mitigation.</li> <li>f) material properties (e.g., yield strength, stress-rupture properties, fracture toughness, minimum operating temperature of the high-pressure turbine rotor) and the method of determining those properties.</li> <li>g) required preservice test and inspection procedures and acceptance criteria to support calculated turbine missile probability.</li> <li>h) actual maximum tangential and radial stresses and their locations in the turbine rotor.</li> <li>i) rotor and blade design analyses, including loading combinations, assumptions and warmup time, that demonstrate sufficient safety margin to withstand loadings from postulated overspeed events up to 120 percent of rated speed.</li> <li>j) description of the required inservice inspection and testing program for valves essential to overspeed protection and any inservice tests, inspections, and maintenance activities for the turbine and valve assemblies that are required to support the calculated missile probability, including inspection and test frequencies with technical bases, type of inspection, techniques, areas to be inspected, acceptance criteria, disposition of reportable indications, and corrective actions.</li> </ul>	10.2
COL Item 10.3-1:	A COL Applicant that references the NuScale Power Plant design certification will provide a site-specific chemistry control program based on the EPRI PWR Secondary Water Chemistry Guidelines and NEI 97-06.	10.3
COL Item 10.3-2:	A COL Applicant that references the NuScale Power Plant design certification will provide a description of the flow-accelerated corrosion monitoring program for carbon steel portions of the steam and power conversion systems that contain water or wet steam and are susceptible to erosion and corrosion damage.	10.3
COL Item 10.4-1:	A COL Applicant that references the NuScale Power Plant design certification will determine the size and number of new and spent resin tanks in the condensate polishing system.	10.4
COL Item 10.4-2:	A COL Applicant that references the NuScale Power Plant design certification will describe the site-specific auxiliary boiler system, the chemistry requirements, chemistry maintenance program, and how the system meets the design requirements.	10.4
COL Item 10.4-3:	A COL Applicant that references the NuScale Power Plant design certification will provide a secondary water chemistry analysis. This analysis will show that the size, materials, and capacity of the feedwater treatment system equipment and components satisfies the water quality requirements of the secondary water chemistry program described in Section 10.3.5, and that it is compatible with the chemicals used.	10.4

**Table 1.8-2: Combined License Information Items (Continued)**

<b>Item No.</b>	<b>Description of COL Information Item</b>	<b>Section</b>
COL Item 11.2-1:	A COL Applicant that references the NuScale Power Plant design certification will ensure mobile equipment used and connected to plant systems is in accordance with ANSI/ANS-40.37, Regulatory Guide (RG) 1.143, 10 CFR 20.1406, NRC IE Bulletin 80-10 and 10 CFR 50.34a.	11.2
COL Item 11.2-2:	A COL Applicant that references the NuScale Power Plant design certification will calculate doses to members of the public using the site-specific parameters, compare those liquid effluent doses to the numerical design objectives of 10 CFR 50, Appendix I, and comply with the requirements of 10 CFR 20.1302 and 40 CFR 190.	11.2
COL Item 11.2-3:	A COL Applicant that references the NuScale Power Plant design certification will perform a site-specific evaluation in accordance with NRC Branch Technical Position 11-6.	11.2
COL Item 11.2-4:	A COL Applicant that references the NuScale Power Plant design certification will perform a site-specific evaluation using the site-specific dilution flow.	11.2
COL Item 11.2-5:	A COL Applicant that references the NuScale Power Plant design certification will perform a cost-benefit analysis as required by 10 CFR 50.34a and 10 CFR 50, Appendix I, to demonstrate conformance with regulatory requirements. This cost-benefit analysis is to be performed using the guidance of Regulatory Guide 1.110.	11.2
COL Item 11.3-1:	A COL Applicant that references the NuScale Power Plant design certification will perform a site-specific cost-benefit analysis.	11.3
COL Item 11.3-2:	A COL Applicant that references the NuScale Power Plant design certification will calculate doses to members of the public using the site-specific parameters, compare those gaseous effluent doses to the numerical design objectives of 10 CFR 50, Appendix I, and comply with the requirements of 10 CFR 20.1302 and 40 CFR 190.	11.3
COL Item 11.3-3:	A COL Applicant that references the NuScale Power Plant design certification will perform an analysis in accordance with Branch Technical Position 11-5 using the site-specific parameters.	11.3
COL Item 11.4-1:	A COL Applicant that references the NuScale Power Plant design certification will describe any mobile equipment used and connected to plant systems in accordance with ANSI/ANS 40.37, Regulatory Guide 1.143, 10 CFR 20.1406, IE Bulletin 80-10, and 10 CFR 50.34a.	11.4
COL Item 11.4-2:	A COL Applicant that references the NuScale Power Plant design certification will develop a site-specific Process Control Program following the guidance of Nuclear Energy Institute (NEI) 07-10A (Reference 11.4-3).	11.4
COL Item 11.5-1:	A COL Applicant that references the NuScale Power Plant design certification will describe site-specific process and effluent monitoring and sampling system components and address the guidance provided in ANSI N13.1-2011, ANSI N42.18-2004 and RGs 1.21, 1.33 and 4.15.	11.5
COL Item 11.5-2:	A COL Applicant that references the NuScale Power design certification will develop an offsite dose calculation manual (ODCM) that contains a description of the methodology and parameters used for calculation of offsite doses for gaseous and liquid effluents, using the guidance of Nuclear Energy Institute (NEI) 07-09A (Reference 11.5-8).	11.5
COL Item 11.5-3:	A COL Applicant that references the NuScale Power design certification will develop a radiological environmental monitoring program (REMP), consistent with the guidance in NUREG-1301 and NUREG-0133, that considers local land use census data for the identification of potential radiation pathways radioactive materials present in liquid and gaseous effluents, and direct external radiation from systems, structures, and components.	11.5
COL Item 12.1-1:	A COL Applicant that references the NuScale Power Plant design certification will describe the operational ALARA program.	12.1
COL Item 12.2-1:	A COL Applicant that references the NuScale Power Plant design certification will describe any additional site-specific contained radiation sources that exceed 100 mCi (including sources for instrumentation and radiography) not identified in Section 12.2.1.	12.2
COL Item 12.3-1:	A COL Applicant that references the NuScale Power Plant design certification will develop the administrative controls regarding access to High Radiation Areas per the guidance of Regulatory Guide 8.38.	12.3
COL Item 12.3-2:	A COL Applicant that references the NuScale Power Plant design certification will develop the administrative controls regarding access to very high radiation areas per the guidance of Regulatory Guide 8.38.	12.3

**Table 1.8-2: Combined License Information Items (Continued)**

<b>Item No.</b>	<b>Description of COL Information Item</b>	<b>Section</b>
COL Item 12.3-3:	A COL Applicant that references the NuScale Power Plant design certification will specify personnel exposure monitoring hardware, specify contamination identification and removal hardware, and establish administrative controls and procedures to control access into and exiting the radiologically controlled area.	12.3
COL Item 12.3-4:	A COL Applicant that references the NuScale Power Plant design certification will develop the processes and programs necessary for the implementation of 10 CFR 20.1501 related to conducting radiological surveys, maintaining proper records, calibration of equipment, and personnel dosimetry.	12.3
COL Item 12.3-5:	A COL Applicant that references the NuScale Power Plant design certification will describe design criteria for locating additional fixed area radiation monitors.	12.3
COL Item 12.3-6:	A COL Applicant that references the NuScale Power Plant design certification will develop the processes and programs necessary for the use of portable airborne monitoring instrumentation, including accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.	12.3
COL Item 12.3-7:	A COL Applicant that references the NuScale Power Plant design certification will develop the processes and programs associated with Objectives 5 and 6, to work in conjunction with design features, necessary to demonstrate compliance with 10 CFR 20.1406, and the guidance of Regulatory Guide 4.21.	12.3
COL Item 12.4-1:	A COL Applicant that references the NuScale Power Plant design certification will estimate doses to construction personnel from a co-located existing operating nuclear power plant that is not a NuScale Power Plant.	12.4
COL Item 12.5-1:	A COL Applicant that references the NuScale Power Plant design certification will describe elements of the operational radiation protection program to ensure that occupational and public radiation exposures are as low as reasonably achievable in accordance with 10 CFR 20.1101.	12.5
COL Item 13.1-1:	A COL Applicant that references the NuScale Power Plant design certification will provide a description of the corporate or home office management and technical support organization.	13.1
COL Item 13.1-2:	A COL Applicant that references the NuScale Power Plant design certification will provide a description of the proposed structure, functions, and responsibilities of the onsite organization necessary to operate and maintain the plant. The proposed operating staff shall be consistent with the minimum licensed operator staffing requirements in Section 18.5.	13.1
COL Item 13.1-3:	A COL Applicant that references the NuScale Power Plant design certification will provide a description of the qualification requirements for each management, operating, technical, and maintenance position described in the operating organization.	13.1
COL Item 13.2-1:	A COL Applicant that references the NuScale Power Plant design certification will provide a description and schedule of the initial training and qualification as well as requalification programs for reactor operators, senior reactor operators, and non-licensed plant staff (plant management, supervisory personnel, technicians and general employees).	13.2
COL Item 13.3-1:	A COL Applicant that references the NuScale Power Plant design certification will provide a description of the onsite operational support center (OSC) including the direct communication system or systems between the OSC and the control room.	13.3
COL Item 13.3-2:	A COL Applicant that references the NuScale Power Plant design certification will provide a description of a near-site emergency operations facility for management of overall licensee emergency response and which complies with the guidance in NUREG-0696, "Functional Criteria for Emergency Response Facilities."	13.3
COL Item 13.3-3:	A COL Applicant that references the NuScale Power Plant design certification will provide a comprehensive emergency plan in accordance with 10 CFR 50.47, 10 CFR 50, Appendix E, 10 CFR 52.48, and 10 CFR 52.79(a)(21).	13.3

**Table 1.8-2: Combined License Information Items (Continued)**

<b>Item No.</b>	<b>Description of COL Information Item</b>	<b>Section</b>
COL Item 13.4-1:	<p>A COL Applicant that references the NuScale Power Plant design certification will provide site-specific information, including implementation schedule, for operational programs:</p> <ul style="list-style-type: none"> <li>• Inservice inspection programs (refer to Section 5.2 and Section 6.6)</li> <li>• Inservice testing programs (refer to Section 3.9 and Section 5.2)</li> <li>• Environmental qualification program (refer to Section 3.11)</li> <li>• Pre-service inspection program (refer to Section 5.2)</li> <li>• Reactor vessel material surveillance program (refer to Section 5.3)</li> <li>• Pre-service testing program (refer to Section 3.9.6, Section 5.2, and Section 6.6)</li> <li>• Containment leakage rate testing program (refer to Section 6.2)</li> <li>• Fire protection program (refer to Section 9.5)</li> <li>• Process and effluent monitoring and sampling program (refer to Section 11.5)</li> <li>• Radiation protection program (refer to Section 12.5)</li> <li>• Non-licensed plant staff training program (refer to Section 13.2)</li> <li>• Reactor operator training program (refer to Section 13.2)</li> <li>• Reactor operator requalification program (refer to Section 13.2)</li> <li>• Emergency planning (refer to Section 13.3)</li> <li>• Process control program (PCP) (refer to Section 11.4)</li> <li>• Security (refer to Section 13.6)</li> <li>• Quality assurance program (refer to Section 17.5)</li> <li>• Maintenance rule (refer to Section 17.6)</li> <li>• Motor-operated valve testing (refer to Section 3.9)</li> <li>• Initial test program (refer to Section 14.2)</li> </ul>	13.4
COL Item 13.5-1:	A COL Applicant that references the NuScale Power Plant design certification will describe the site-specific procedures that provide administrative control for activities that are important for the safe operation of the facility consistent with the guidance provided in RG 1.33, Revision 3.	13.5
COL Item 13.5-2:	A COL Applicant that references the NuScale Power Plant design certification will describe the site-specific procedures that licensed operators perform in the control room including normal operating procedures, abnormal operating procedures, and emergency operating procedures (EOPs), and describe the classification system for these types of procedures and general format and content for each classification.	13.5
COL Item 13.5-3:	<p>A COL Applicant that references the NuScale Power Plant design certification will describe the site-specific program for developing maintenance and other operating procedures, including the classification system, objectives, and organizational responsibility for the following types of procedures:</p> <ul style="list-style-type: none"> <li>• plant radiation protection procedures</li> <li>• emergency preparedness procedures</li> <li>• instrument calibration and test procedures</li> <li>• chemical-radiochemical control procedures</li> <li>• radioactive waste management procedures</li> <li>• maintenance and modification procedures</li> <li>• material control procedures</li> <li>• plant security procedures</li> </ul>	13.5
COL Item 13.6-1:	<p>A COL Applicant that references the NuScale Power Plant design certification will provide the following:</p> <ul style="list-style-type: none"> <li>• Security Plans (Physical Security, Security Training and Qualification, and Safeguards Contingency Plans)</li> <li>• proposed site security provisions to be implemented during construction and as modules are completed and become operational of a new plant</li> <li>• portions of the physical security system not located within the nuclear island and structures</li> </ul>	13.6

**Table 1.8-2: Combined License Information Items (Continued)**

<b>Item No.</b>	<b>Description of COL Information Item</b>	<b>Section</b>
COL Item 13.6-2:	A COL Applicant that references the NuScale Power Plant design certification will be responsible for the requirements described in Table 5-1 of TR-0416-48929, Rev 0 NuScale Design of Physical Security Systems.	13.6
COL Item 13.6-3:	A COL Applicant that references the NuScale Power Plant design certification will provide a secondary alarm station that is equal and redundant to the central alarm station.	13.6
COL Item 13.6-4:	A COL Applicant that references the NuScale Power Plant design certification will provide Inspections, Tests, Analyses, and Acceptance Criteria for site specific physical security SSCs.	13.6
COL Item 13.6-5:	A COL Applicant that references the NuScale Power Plant design certification will provide a description of the Access Authorization Program.	13.6
COL Item 13.6-6:	A COL Applicant that references the NuScale Power Plant design certification will provide a Cyber Security Plan.	13.6
COL Item 13.7-1:	A COL Applicant that references the NuScale Power Plant design certification will provide a description of the applicant's 10 CFR 26 compliant fitness-for-duty (FFD) program for plant operations.	13.7
COL Item 13.7-2:	A COL Applicant that references the NuScale Power Plant design certification will provide a description of the applicant's 10 CFR 26 compliant fitness-for-duty (FFD) program for construction.	13.7
COL Item 14.2-1:	A COL Applicant that references the NuScale Power Plant design certification will describe the site-specific organizations that manage, supervise, or execute the Initial Test Program, including the associated training requirements.	14.2
COL Item 14.2-2:	A COL Applicant that references the NuScale Power Plant design certification is responsible for the development of the Startup Administration Manual that will contain the administrative procedures and requirements that control the activities associated with the Initial Test Program. The COL applicant will provide a milestone for completing the Startup Administrative Manual and making it available for NRC inspection.	14.2
COL Item 14.2-3:	A COL Applicant that references the NuScale Power Plant design certification will identify the specific operator training to be conducted during low-power testing related to the resolution of TMI Action Plan Item I.G.1, as described in NUREG-0660, NUREG-0694, and NUREG-0737.	14.2
COL Item 14.2-4:	A COL Applicant that references the NuScale Power Plant design certification will provide a schedule for the Initial Test Program.	14.2
COL Item 14.2-5:	A COL Applicant that references the NuScale Power Plant design certification will provide a test abstract for the potable water system pre-operational testing.	14.2
COL Item 14.2-6:	A COL Applicant that references the NuScale Power Plant design certification will provide a test abstract for the SMS pre-operational testing.	14.2
COL Item 14.2-7:	A COL Applicant that references the NuScale Power Plant design certification will select the plant configuration to perform the Island Mode Test (number of NPMs in service).	14.2
COL Item 14.3-1:	A COL Applicant that references the NuScale Power Plant design certification will provide the site-specific selection methodology and Inspections, Tests, Analyses, and Acceptance Criteria for emergency planning.	14.3
COL Item 14.3-2:	A COL Applicant that references the NuScale Power Plant design certification will provide the site-specific selection methodology and ITAAC for SSC within their scope.	14.3
COL Item 16.1-1:	A COL Applicant that references the NuScale Power Plant design certification will provide the final plant-specific information identified by [ ] in the generic Technical Specifications.	16.1
COL Item 17.4-1:	A combined license (COL) applicant that references the NuScale Power Plant design certification will describe the reliability assurance program conducted during the operations phases of the plant's life.	17.4
COL Item 17.4-2:	A COL Applicant that references the NuScale Power Plant design certification will identify any site-specific SSC within the scope of the RAP.	17.4
COL Item 17.4-3:	A COL Applicant that references the NuScale Power Plant design certification will identify the QA controls for the RAP SSC during site-specific design, procurement, fabrication, construction, and preoperational testing activities.	17.4

**Table 1.8-2: Combined License Information Items (Continued)**

<b>Item No.</b>	<b>Description of COL Information Item</b>	<b>Section</b>
COL Item 17.5-1:	A COL Applicant that references the NuScale Power Plant design certification will describe the quality assurance program applicable to site-specific design activities and to the construction and operations phases.	17.5
COL Item 17.6-1:	A COL Applicant that references the NuScale Power Plant design certification will describe the program for monitoring the effectiveness of maintenance required by 10 CFR 50.65.	17.6
COL Item 18.5-1:	A COL Applicant that references the NuScale Power Plant design certification will address the staffing and qualifications of non-licensed operators.	18.5
COL Item 18.12-1:	A COL Applicant that references the NuScale Power Plant design certification will provide a description of the human performance monitoring program in accordance with applicable NUREG-0711 or equivalent criteria.	18.12
COL Item 19.1-1:	A COL Applicant that references the NuScale Power Plant design certification will identify and describe the use of the probabilistic risk assessment in support of licensee programs being implemented during the COL application phase.	19.1
COL Item 19.1-2:	A COL Applicant that references the NuScale Power Plant design certification will identify and describe specific risk-informed applications being implemented during the COL application phase.	19.1
COL Item 19.1-3:	A COL Applicant that references the NuScale Power Plant design certification will specify and describe the use of the probabilistic risk assessment in support of Licensee programs during the construction phase (from issuance of the COL up to initial fuel loading).	19.1
COL Item 19.1-4:	A COL Applicant that references the NuScale Power Plant design certification will specify and describe risk-informed applications during the construction phase (from issuance of the COL up to initial fuel loading).	19.1
COL Item 19.1-5:	A COL Applicant that references the NuScale Power Plant design certification will specify and describe the use of the probabilistic risk assessment in support of licensee programs during the operational phase (from initial fuel loading through commercial operation).	19.1
COL Item 19.1-6:	A COL Applicant that references the NuScale Power Plant design certification will specify and describe risk-informed applications during the operational phase (from initial fuel loading through commercial operation).	19.1
COL Item 19.1-7:	A COL Applicant that references the NuScale Power Plant design certification will evaluate site-specific external event hazards, screen those for risk-significance, and evaluate the risk associated with external hazards that are not bounded by the design certification.	19.1
COL Item 19.1-8:	A COL Applicant that references the NuScale Power Plant design certification will confirm the applicability of assumptions and data and modify as necessary for the to the as-built/as-operated probabilistic risk assessment.	19.1
COL Item 19.2-1:	A COL Applicant that references the NuScale Power Plant design certification will develop severe accident management guidelines and other administrative controls to define the response to beyond design basis events.	19.2
COL Item 19.2-2:	A COL Applicant that references the NuScale Power Plant design certification will use the site-specific probabilistic risk assessment to evaluate and identify improvements in the reliability of core and containment heat removal systems as specified by 10 CFR 50.34(f)(1)(i).	19.2
COL Item 19.2-3:	A COL Applicant that references the NuScale Power Plant design certification will evaluate all severe accident mitigation design alternatives screened as "not required for design certification application."	19.2
COL Item 19.3-1:	A COL Applicant that references the NuScale Power Plant design certification will identify any site-specific RTNSS SSC and applicable RTNSS process controls.	19.3
COL Item 20.1-1:	A COL Applicant that references the NuScale Power Plant design certification will ensure equipment and structures credited for FLEX strategies are designed to be available following a site-specific seismic hazard.	20.1
COL Item 20.1-2:	A COL Applicant that references the NuScale Power Plant design certification will determine if a flood hazard is applicable at the site location. If a flood hazard is applicable, then the COL applicant will ensure equipment and structures credited for FLEX strategies are designed to be available following a site-specific flood (including wave action) hazard.	20.1

**Table 1.8-2: Combined License Information Items (Continued)**

<b>Item No.</b>	<b>Description of COL Information Item</b>	<b>Section</b>
COL Item 20.1-3:	A COL Applicant that references the NuScale Power Plant design certification will determine if high wind and applicable missile hazards are applicable at the site location. If high wind and applicable missile hazards are applicable, then the COL applicant will ensure equipment and structures credited for FLEX strategies are designed to be available following a site-specific high wind and applicable missile hazards.	20.1
COL Item 20.1-4:	A COL Applicant that references the NuScale Power Plant design certification will determine if snow, ice and extreme cold temperature hazards are applicable at the site location. If snow, ice and extreme cold hazards are applicable, the COL applicant will ensure equipment and structures credited for FLEX strategies are designed to be available following a site-specific snow, ice or extreme cold temperature hazard.	20.1
COL Item 20.1-5:	A COL Applicant that references the NuScale Power Plant design certification will determine if extreme high temperature hazard is applicable at the site location. If extreme high temperature hazard is applicable, the COL applicant will ensure equipment and structures credited for FLEX strategies are designed to be available following a site-specific extreme high temperature hazard.	20.1
COL Item 20.1-6:	A COL Applicant that references the NuScale Power Plant design certification will develop and implement the strategies and guidance for makeup to the UHS after an ELAP event utilizing supplemental FLEX equipment.	20.1
COL Item 20.1-7:	A COL Applicant that references the NuScale Power Plant design certification will develop a training and qualification program using the systems approach to training process. The training will ensure personnel will be able to perform activities in accordance with the FLEX strategies and guidelines.	20.1
COL Item 20.1-8:	A COL Applicant that references the NuScale Power Plant design certification will develop procedures, training and qualification program for operations, maintenance, testing, and calibration of UHS level instrumentation to ensure the level instruments will be available when needed and personnel are knowledgeable in interpreting the information as addressed in NEI 12-02.	20.1
COL Item 20.2-1:	A COL Applicant that references the NuScale Power design certification will develop enhanced firefighting capabilities by implementing the guidance in NRC guidance document "Developing Mitigating Strategies/Guidance for Nuclear Power Plants to Respond to Loss of Large Areas of the Plant in Accordance with B.5.b of the February 25, 2002, Order" dated February 25, 2005 (Reference 20.2-3). The enhanced firefighting capabilities should address the expectation elements listed in Section 4.1.3 of the Technical Report TR-0816-50796 (Reference 20.2-1).	20.2
COL Item 20.2-2:	A COL Applicant that references the NuScale Power design certification will provide a means for water spray scrubbing using fog nozzles and the availability of water sources, and address runoff water containment issues (sandbags, portable dikes, etc.) as an attenuation measure for mitigating radiation releases outside containment.	20.2
COL Item 20.3-1:	A COL Applicant that references the NuScale Power Plant design certification will ensure that the severe accident management guidelines, FLEX support guidelines, and extensive damage mitigation guidelines are integrated with the emergency operating procedures consistent with Recommendation 8.1 of SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan."	20.3
COL Item 20.4-1:	A COL Applicant that references the NuScale Power Plant design certification will perform an analysis that demonstrates the Emergency Response Organization staff has the ability to implement the strategies of the emergency operating procedures, severe accident mitigation guidelines, FLEX support guidelines, and extensive damage mitigation guidelines. The analysis will be performed with the off-site response organization access to on-site being impeded. The event shall be a loss of all on-site and off-site alternating current power and loss of normal access to the ultimate heat sink.	20.4
COL Item 20.4-2:	A COL Applicant that references the NuScale Power Plant design certification will develop a supporting Emergency Response Organization structure with defined roles and responsibilities to implement the strategies of the emergency operating procedures, severe accident mitigation guidelines, FLEX support guidelines, and extensive damage mitigation guidelines.	20.4

**Table 1.8-2: Combined License Information Items (Continued)**

<b>Item No.</b>	<b>Description of COL Information Item</b>	<b>Section</b>
COL Item 20.4-3:	A COL Applicant that references the NuScale Power Plant design certification will develop and describe at least one onsite and one offsite communications system capable of remaining functional during an extended loss of alternating current power including the effects of the loss of the local communications infrastructure.	20.4
COL Item 20.4-4:	A COL Applicant that references the NuScale Power Plant design certification will develop, implement, and maintain the training and qualification of personnel that perform activities in accordance with FLEX support guidelines, severe accident mitigation guidelines, and extensive damage mitigation guidelines. The training and qualification on these activities will be developed using the systems approach to training as defined in 10 CFR 55.4 except for elements already covered under other NRC regulations.	20.4
COL Item 20.4-5:	A COL Applicant that references the NuScale Power Plant design certification will develop drills or exercises that demonstrate the ability to transition to one or more of the strategies and guidelines of the emergency operating procedures, FLEX support guidelines, extensive damage mitigation guidelines, and severe accident mitigation guidelines using only the station communication equipment designed to be available following an extended loss of alternating current including effects of the loss of the local communications infrastructure.	20.4
COL Item 20.4-6:	A COL Applicant that references the NuScale Power Plant design certification will develop and describe the means to be used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials to the environment including releases from all reactor core and spent fuel pool sources.	20.4



## 1.9 Conformance with Regulatory Criteria

This section provides a guide to conformance with regulatory criteria in individual table format, as listed below. Conformance is assessed to regulatory criteria in effect six months before the anticipated docket date.

Table 1.9-1, "Conformance Status Legend," defines the codes used to indicate conformance in Table 1.9-2 through Table 1.9-8

Table 1.9-2, Conformance with Regulatory Guides

Table 1.9-3, Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)

Table 1.9-4, Conformance with Interim Staff Guidance (ISG)

Table 1.9-5, Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933)

Table 1.9-6, Evaluation of Operating Experience (Generic Letters and Bulletins)

Table 1.9-7, Conformance with Advanced and Evolutionary Light Water Reactor Design Issues (SECYs and associated SRMs)

Table 1.9-8, Conformance with SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs"

COL Item 1.9-1: A COL applicant that references the NuScale Power Plant design certification will review and address the conformance with regulatory criteria in effect six months before the submittal date of the COL application for the site-specific portions and operational aspects of the facility design.

### 1.9.1 Conformance with Regulatory Guides

Table 1.9-2 provides an evaluation of conformance with the guidance in NRC regulatory guides in effect 6 months before the submittal date of the Final Safety Analysis Report (FSAR). This evaluation also includes an identification and description of deviations from the guidance in the NRC Regulatory Guides as well as suitable justifications for any alternative approaches proposed.

The conformance evaluation was performed on the following groups of Regulatory Guides:

- Division 1, Power Reactors
- Division 4, Environmental and Siting (applies to the environmental report and should be discussed therein)
- Division 5, Materials and Plant Protection (applies to the security plan and should be discussed therein)
- Division 8, Occupational Health

### 1.9.2 Conformance with Standard Review Plan

NuScale performed a review of the SRP including Branch Technical Positions and guidance referenced within the SRP. A summary of this review was submitted to the NRC as NP-RT-0612-023, "Gap Analysis Summary Report," Revision 1, in July 2014 (Reference 1.9-1). The gap analysis review for applicability was directed towards the acceptance criteria of each SRP section. However, the review considered the relevance of sub-tier guidance whether referenced in the acceptance criteria or in other portions of the SRP section being reviewed.

Additionally, NuScale considered conformance with the DSRS developed by the NRC for the review of the NuScale Power small modular reactor design. This information has been incorporated into Table 1.9-3. Conformance with NRC Interim Staff Guidance is presented in Table 1.9-4.

### 1.9.3 Generic Issues

In accordance with 10 CFR 52.47(a)(8), conformance must be assessed against technically relevant Three Mile Island (TMI) requirements identified in 10 CFR 50.34(f), except for paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v). Plant characteristics and plant programs that address relevant TMI requirements are described in the appropriate FSAR sections.

In accordance with 10 CFR 52.47(a)(21), proposed resolutions must be identified for any technically relevant unresolved safety issues and medium-priority to high-priority generic safety issues (GSI) identified in the version of NUREG-0933 that is current six months prior to the application for design certification. Resolution and closure of generic issues is managed via the NRC Generic Issues Program. NRC SECY-07-0110, dated July 6, 2007 provides the most recent supplemental status report of the Generic Issues Program prior to the FSAR submittal. As such, Appendix B of NUREG-0933, Rev. 21 (including the Main Report and Supplements 1-34) and NRC letter SECY-07-0110, were used to identify those generic issues applicable to the NuScale Power Plant design certification.

Table 1.9-5 identifies the applicable TMI requirements and generic issues, along with an abbreviated summary description of the NRC position for each table entry. Table 1.9-5 also provides a brief conformance assessment notation, including annotation of any exceptions, and a reference to the FSAR section(s) addressing the issue. Those 10 CFR 50.34(f) TMI requirements specifically applicable to another nuclear power plant design concept, such as a boiling water reactor (BWR), nuclear power plant designs of another vendor or passive plant designs, are not included in Table 1.9-5, because they are not technically relevant to the NuScale design. Those NUREG-0933 generic issues determined as non-applicable were eliminated from consideration in Table 1.9-5 based on these:

- Resolved: Issue has been completely resolved and removed from the latest Generic Issues Program list of Active and Regulatory Office Implementation Generic Issues.
- BWR, Ice Condenser Containment or Other: Issue applies to another nuclear power plant design concept or to the design of a nuclear facility other than a nuclear power plant.

#### 1.9.4 Operational Experience (Generic Communications)

Per 10 CFR 52.47(a)(22) requirements, applicants for design certification of new plant designs include a description of how operational experience has been incorporated into the design process. Operational experience insights are incorporated into applicable SRP sections as they are updated. Operational experience from NRC Bulletins and Generic Letters not incorporated into the most recent applicable SRP six months before the application docket date are incorporated into the design unless stated otherwise. The design is an evolution of nuclear power plant designs that have been operated in the United States, as addressed by 10 CFR 52.41(b)(1); hence NRC guidance for technically relevant operational experience issues is addressed in the appropriate FSAR sections.

The conformance assessment relative to operational experience is provided in Table 1.9-6, "Evaluation of Operating Experience (Generic Letters and Bulletins)." Further, 10 CFR 21 notifications were reviewed for impact to the NuScale design as part of the supplier evaluation process. NuScale's QA supplier evaluation program includes a review of 10 CFR 21 notifications for every Nuclear Safety Related supplier prior to use as an approved supplier for safety related items/services. The evaluation for any 10 CFR 21 notifications is also performed as part of monitoring of supplier performance by periodic annual review. There have been no 10 CFR 21 notifications impacting nuclear safety related work performed by NuScale approved safety related suppliers for the development of the NuScale Design. Therefore, all applicable 10 CFR 21 notifications have been evaluated.

#### 1.9.5 Advanced and Evolutionary Light-Water Reactor Design Issues

Guidance in SRP Section 1.0 recommends that this section address the applicable licensing and policy issues developed by the NRC and documented in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," Secretary of the Commission, Office of the U.S. Nuclear Regulatory Commission, April, 2, 1993, as supplemented by the associated staff requirements memorandum (SRM) dated July 21, 1993.

Table 1.9-7 lists applicable design issues identified in SECYs and their associated SRMs. The table provides a conformance assessment notation, including annotation of any exceptions, for each issue. Table 1.9-7 also provides a cross-reference from the SECY issues to the FSAR sections that address them. Table 1.9-8 provides a separate assessment of SECY-93-087 line items pertaining to ALWR designs.

#### 1.9.6 References

1.9-1 NuScale Power, LLC, NP-RT-0612-023, Rev 1, Gap Analysis Summary Report.

**Table 1.9-1: Conformance Status Legend**

<b>Conformance Status Code</b>	<b>Description</b>
<b>Conforms</b>	The regulation or regulatory guidance is relevant and applicable, and can be applied “as-is.” The design fully conforms to the requirement or guidance described in the Section(s) identified. Where options are identified in the regulation or regulatory guidance, “Conforms” indicates that the design fully conforms to the option(s) selected.
<b>Partially Conforms</b>	<p>The design conforms to those portions of the requirement or guidance that can be appropriately applied as written.</p> <p>The underlying purpose or intent of the requirement or guidance is relevant to the design but cannot be appropriately applied as written, or some portion of the requirement or guidance is applicable while other portions are not applicable. The following are examples:</p> <ul style="list-style-type: none"> <li>• A portion of the regulatory requirement or guidance is literally applicable, but the specific language refers to a different type of light water reactor (LWR) design or structures, systems, and components (SSC) that are not part of the design.</li> <li>• The regulatory requirement or guidance is applicable except for aspects that are specific to combined license or early site permit applicants, or to BWR designs, etc.</li> <li>• The intent of a regulatory requirement or guidance is applicable, but the specific language refers to one of the following: <ul style="list-style-type: none"> <li>- a different type of LWR design</li> <li>- an SSC that is not part of the design, but for which a substantively equivalent function is served by other SSCs within the design</li> </ul> </li> </ul>
<b>Not Applicable</b>	<p>The regulation or guidance is not appropriate to apply and therefore conformance is not required. Applicability to the COL applicant is identified in the table. The following are examples:</p> <ul style="list-style-type: none"> <li>• The regulatory requirement or guidance is applicable only to BWR designs.</li> <li>• The regulatory requirement or guidance is applicable only to large pressurized water reactor (PWR) designs.</li> <li>• The regulatory requirement or guidance is applicable to the design, but is the responsibility of the COL applicant.</li> <li>• The regulatory requirement or guidance is applicable to SSCs that are not part of the design.</li> </ul>
<b>Departure</b>	<p>For items found within the Code of Federal Regulations (CFR): the regulation is literally applicable; however NuScale intends to depart from the regulation based on the design or safety basis of the NuScale design. That is to say that conformance to the regulation would have a minimal, or even negative, impact on safety of the NuScale design and hence a departure from the regulation (that was originally created for traditional LWRs) is warranted. The form of the departure may be through an exemption request under 10 CFR 52.7 or through a specific process available for a set of regulations. For example, the introduction to 10 CFR 50, Appendix A provides a departure from the General Design Criteria as explained in Section 3.1, and the TMI action items may be identified and justified as “not technically relevant” to the design consistent with 10 CFR 52.47(a)(8) and 10 CFR 50.34(f).<sup>1</sup></p> <p><sup>1</sup>Note that some TMI action items are categorized as “Partially Conforms” or “Not Applicable” rather than “Departure.” The difference is that those requirements are not applicable by their own terms, for example because they apply to BWRs or to an SSC that the NuScale design lacks. A departure from a TMI requirement is appropriate where the requirement is literally applicable but is inappropriate to apply to the NuScale design.</p>

**Table 1.9-2: Conformance with Regulatory Guides**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors	2	Not Applicable	Not Applicable	This guidance is only applicable to BWRs.	Not Applicable
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors	2	Not Applicable	Not Applicable	This RG pertains to existing reactors; RG 1.183 is specified in SRP Section 15.0.3 to be used for new reactors.	Not Applicable
1.5	Safety Guide 5 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors	-	Not Applicable	Not Applicable	This guidance is only applicable to BWRs.	Not Applicable
1.6	Safety Guide 6 - Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	-	Partially Conforms	Applicable	The onsite electrical AC power systems do not contain any Class 1E distribution systems. The EDSS design conforms to the guidance for independence of standby power sources and their distribution systems.	8.3
1.7	Control of Combustible Gas Concentrations in Containment	3	Not Applicable	Not Applicable	The containment vessel design is such that its integrity does not rely on combustible gas control systems.	6.2
1.8	Qualification and Training of Personnel for Nuclear Power Plants	3	Not Applicable	Applicable	Site-specific programmatic and operational activities are the responsibility of the COL applicant.	Not Applicable
1.9	Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants	4	Not Applicable	Not Applicable	Based on reduced reliance on AC power, the design does not require or include safety-related emergency diesel generators.	8.3
1.11	Instrument Lines Penetrating the Primary Reactor Containment	1	Not Applicable	Not Applicable	No lines penetrate the NPM containment.	6.2

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

<b>RG</b>	<b>Division Title</b>	<b>Rev.</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
1.12	Nuclear Power Plant Instrumentation for Earthquakes	2	Partially Conforms	Applicable	Selection of specific equipment is the responsibility of the COL applicant. In addition, seismic detectors cannot be installed inside the containment so Section 3.7.3 indicates they are installed in the RXB.	3.7
1.13	Spent Fuel Storage Facility Design Basis	2	Partially Conforms	Applicable	The design of the new and spent fuel storage facility complies with Regulatory Position C.8, Makeup Water, although not as in a typical PWR. Compliance with this Position is demonstrated by the large inventory of water within the Seismic Category I structures forming the ultimate heat sink (UHS) and by the separate Quality Group C, Seismic Category I makeup line. For Regulatory Position C.9, Pool Cooling, one exception is that the UHS pool structures containing the inventory of makeup water credited for spent fuel cooling during accident conditions meet Seismic Category I requirements, but as structures, they are not designed to Quality Group C requirements. The second exception is that the reactor building ventilation system is not credited with the capability to vent steam or moisture to the atmosphere to protect safety-related components from high temperatures and moisture levels because such protection is not required for the design.	9.1 9.2
1.14	Reactor Coolant Pump Flywheel Integrity	1	Not Applicable	Not Applicable	This guidance is applicable only to PWR designs that rely on reactor coolant pumps. The NuScale design differs from that of large PWRs in that the design does not require or include reactor coolant pumps. Rather, the design uses passive natural circulation of the primary coolant, eliminating the need for reactor coolant pumps.	Not Applicable

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Pre-operational and Initial Startup Testing	3	Partially Conforms	Applicable	The aspects of this RG that mandate start-up tests or inspections for non-prototype reactors are applicable as these tests must be accommodated in the start-up of each installed NuScale Power Module. The remainder is applicable to the COL applicant, and only specific COL applicants which are for prototype reactors.	3.9
1.21	Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste	2	Partially Conforms	Applicable	Site-specific, programmatic and operational aspects are the responsibility of the COL applicant.	11.5
1.22	Periodic Testing of Protection System Actuation Functions	0	Conforms	Applicable	None.	7.2
1.23	Meteorological Monitoring Programs for Nuclear Power Plant	1	Not Applicable	Applicable	This guidance is the responsibility of the COL applicant.	2.3
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure	0	Partially Conforms	Applicable	Site-specific guidance is the responsibility of the COL applicant.	11.3
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	0	Not Applicable	Not Applicable	This RG pertains to TID14844 source terms used by licensees of existing reactors that are not authorized to use the alternative source term under 10CFR50.67. RG 1.183 is specified to be used in lieu of RG 1.25 for new reactors (and existing reactors authorized to use the alternative source term under 10CFR50.67).	Not Applicable

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

<b>RG</b>	<b>Division Title</b>	<b>Rev.</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	4	Conforms	Applicable	The quality group classification from RG 1.26 applicable to a specific component is described throughout the FSAR.	3.2 5.2 5.4 6.2 9.1 9.3 10.3 10.4
1.27	Ultimate Heat Sink for Nuclear Power Plants (for Comment)	3	Not Applicable	Not Applicable	RG does not apply to plants that utilize a passive containment cooling system as their ultimate heat sink.	9.2.5
1.28	Quality Assurance Program Criteria (Design and Construction)	4	Conforms	Applicable	The NuScale design is based on NQA12008 and the NQA1a2009 addenda (rather than NQA11994), as endorsed in RG 1.28, Rev. 4. The design for threaded fasteners meet the cleaning criteria in RG 1.28.	3.13 4.5 6.1 7.2 10.3 14.2 17.1 17.5



**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.29	Seismic Design Classification for Nuclear Power Plants	5	Conforms	Applicable	Endorsed in Section 3.2.1. Each SSC described in Staff Regulatory Guidance C.1 is designated as Seismic Category I. The Seismic Category I dynamic analysis is extended at the interface between Seismic Category I and non-seismic Category I SSC in accordance with Staff Regulatory Guidance C.2. Pertinent Quality Assurance requirements of Appendix B to 10 CFR 50 are applied to all activities affecting the safety-related functions of Seismic Category I SSC in accordance with Staff Regulatory Guidance C.3.  The seismic classification from RG 1.29 applicable to a specific component is described throughout the FSAR.	3.2 5.2 5.4 6.1 6.2 6.3 6.4 6.6 8.3 9.1 9.2 9.3 9.4 9.5 10.3 10.4
1.30	Safety Guide 30 - Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	-	Not Applicable	Not Applicable	This RG endorses IEEE Std. 336-1971 for the installation, inspection, and testing of instrumentation and electric equipment. The NuScale design is based on NQA12008 and the NQA1a2009 addenda, as endorsed in RG 1.28, Rev. 4. NQA12008/1a-2009 (Subpart 2.4) references IEEE Std. 336-1985 (as opposed to IEEE Std. 336-1971). The substantive content and intent of RG1.30 is contained in Subpart 2.4 of NQA12008/1a-2009 and IEEE Std. 336-1985, which is applicable to the NuScale design per NQA12008/1a2009.	Not Applicable
1.31	Control of Ferrite Content in Stainless Steel Weld Metal	4	Conforms	Applicable	None.	4.5 5.2 5.3 5.4 6.1

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.32	Criteria for Power Systems for Nuclear Power Plants	3	Partially Conforms	Applicable	The NuScale passive safety systems do not require AC power to fulfill safety-related functions and the onsite electric AC power systems are not safety related. The EDSS conforms to RG 1.32.	8.3
1.33	Quality Assurance Program Requirements (Operation)	3	Not Applicable	Not Applicable	The NuScale certified design effort is being conducted under a QA program that implements the QA standards of NQA-1-2008/1a-2009, as endorsed by RG 1.28, Revision 4. It is anticipated and assumed that COL applicants referencing the NuScale design similarly will apply NQA-1-2008/1a-2009, consistent with the QAPD to be described in the DCA. The NRC is currently reviewing NEI 11-04, draft Revision 0, which provides guidance for developing a QAPD that implements NQA-1-2008/1a-2009 and RG 1.28, Revision 4. As detailed in Appendix 1 of NEI 11-04, the substantive content of RG 1.33 is subsumed into NQA-1-2008/1a2009 (with modifications described in RG1.28) and NEI1104. Thus, implementation of NQA-1-2008/1a-2009 (with modifications described in RG1.28) and NEI 11-04 would ensure that the substantive content of RG 1.33 is satisfied.	Not Applicable
1.34	Control of Electroslag Weld Properties	1	Conforms	Applicable	None.	5.2 5.3 5.4
1.35	Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments	3	Not Applicable	Not Applicable	The containment vessel is a steel containment (i.e., does not use concrete, prestressed or other, in its design).	Not Applicable
1.35.1	Determining Prestressing Forces for Inspection of Prestressed Concrete Containments	-	Not Applicable	Not Applicable	The containment vessel is a steel containment (i.e., does not use concrete, prestressed or other, in its design).	Not Applicable

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel	-	Conforms	Applicable	The NuScale design does not use nonmetallic thermal insulation on RCPB or CNV components. If nonmetallic thermal insulation is used on austenitic stainless steel components that are part of the RCPB, CNV or other safety-related or risk-significant systems, the nonmetallic thermal insulation is selected and controlled to ensure the impurity limits contained in RG 1.36 have been met to preclude stress corrosion cracking.	6.1 10.3
1.40	Qualification of Continuous Duty Safety-Related Motors for Nuclear Power Plants	1	Not Applicable	Not Applicable	The NuScale design does not use continuous duty Class 1E motors.	Not Applicable
1.41	Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments	-	Not Applicable	Not Applicable	This RG is not identified as an applicable RG in DSRS Section 8.1.	Not Applicable
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components	1	Conforms	Applicable	None.	5.2 5.3 5.4 6.1
1.44	Control of the Processing and Use of Stainless Steel	1	Partially Conforms	Applicable	This RG is applicable except for its specification of applying RG 1.37 for cleaning and flushing of finished surfaces. RG 1.37 has been withdrawn by the NRC.	4.5 5.2 5.3 5.4 6.1 10.3

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.45	Guidance on Monitoring and Responding to Reactor Coolant System Leakage	1	Partially Conforms	Applicable	The design satisfies RG 1.45 guidance in that two systems are utilized to detect leakage into the containment; containment pressure monitoring and leakage collection. Both leakage detection methods satisfy Regulatory Positions C.2.1 and C.2.2 in RG 1.45 in that: a) leakage to the primary reactor containment from unidentified sources can be detected, monitored, and quantified for rates = 0.05 gpm; and, b) response time (not including transport delay time) is < 1 hour for a leakage rate greater than 1 gpm. Regulatory Position C.2.4 is satisfied in that the containment pressure method is capable of performing its function following a seismic event that does not require plant shutdown (i.e., vacuum pump remains functional). C.2.5 is satisfied in that both methods permit calibration and testing during plant operation. Finally, radiation detectors in the CES condenser vent line provide an early indication of RCS leakage consistent with Regulatory Position C.2.3. All leakage is treated as unidentified because of the limited capability to identify or quantify RCS leakage.	5.2 6.2 9.3 11.5
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	1	Conforms	Applicable	None.	7.2 8.3
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel	1	Conforms	Applicable	None.	5.2 5.3 5.4 6.1 10.3

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
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	4	Not Applicable	Not Applicable	This guidance addresses engineered safety feature (ESF) filter and atmosphere cleanup systems designed for fission product removal in a post-design basis accident environment. The design does not rely on ESF filter and atmosphere cleanup systems to mitigate the consequences of a design basis accident. Non-safety-related normal ventilation systems provide atmosphere cleanup capability, as necessary, that meets the design, testing, and maintenance guidelines specified in RG 1.140. These systems provide appropriate containment, confinement, and filtering to limit releases of airborne radioactivity to the environment during normal and postulated accident conditions. However, these systems are not required following an accident, and accordingly receive no credit in the determination of the radiological consequences of an accident.	Not Applicable
1.53	Application of the Single-Failure Criterion to Safety Systems	2	Conforms	Applicable	None.	7.1 7.2 8.3 9.3
1.54	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants	2	Partially Conforms	Applicable	Portions of this RG govern operational aspects (e.g., maintenance of safety-related coatings) that are the responsibility of the COL applicant.	11.2 11.4 12.3
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components	2	Partially Conforms	Applicable	Applicable except 1) justifying alternative by meeting NB requirements vs. NE requirements and 2) for reference to 10CFR50.34(f)(3)(v), since per 10CFR50.34(f) and 10CFR52.47(a)(8), a design certification applicant does not have to show compliance with 10CFR50.34(f)(3)(v).	3.8 6.2
1.59	Design Basis Floods for Nuclear Power Plants	2	Not Applicable	Not Applicable	The design assumes the NPP is located above the maximum flood height (including wind induced wave run-up).	Not Applicable

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants	2	Not Applicable	Not Applicable	The Certified Seismic Design Response Spectra (CSDRS) was not developed using RG 1.60. However, it is demonstrated that the design envelops the RG 1.60 spectra anchored to 0.1g.	3.7
1.61	Damping Values for Seismic Design of Nuclear Power Plants	1	Conforms	Applicable	In accordance with the guidance of RG 1.61, an alternative damping value for the NPM substructure was determined. The NPM subsystem is comprised of vessels, bearing joints, cables, internals, friction surfaces, etc.	3.7 3.12 Appendix 3A 5.3
1.62	Manual Initiation of Protective Actions	1	Conforms	Applicable	This RG refers to Point 4 of BTP 7-19, Revision 5, dated March 2007. NuScale intends to apply BTP 7-19, Revision 6 (including Point 4).	7.1 7.2
1.63	Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants	3	Conforms	Applicable	IEEE 741-1997 is used for external circuit protection of electrical penetration assemblies. Although IEEE 741-1997 is not endorsed by RG 1.63 (endorsing IEEE 317-1983) the design philosophy does not deviate from RG 1.63.	3.11 8.1 8.3
1.65	Materials and Inspections for Reactor Vessel Closure Studs	1	Partially Conforms	Applicable	This RG provides guidance for use in selecting reactor vessel closure stud bolting materials and properties, and conducting preservice and inservice inspection of the closure studs. Performance of inservice inspection is the responsibility of the COL applicant.	3.13 5.3 6.1
1.68	Initial Test Programs for Water-Cooled Nuclear Power Plants	4	Partially Conforms	Applicable	This guidance is applicable except for aspects that (1) are BWR-specific or address specific PWR SSC design features not in the NuScale design; or (2) involve site-specific program implementation activities that are the responsibility of the COL applicant.	4.4 5.4 8.3 10.4 14.2 21.2
1.68.1	Initial Test Program of Condensate and Feedwater Systems for Light-Water Reactors	2	Partially Conforms	Applicable	This RG is applicable except for aspects that are BWR-specific or address specific unique PWR design features not in the NuScale design.	10.4

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.68.2	Initial Startup Test Program to Demonstrate Remote Shut-down Capability for Water-Cooled Nuclear Power Plants	2	Partially Conforms	Applicable	This guidance is applicable except for site-specific aspects including test performance, test report preparation, and records retention, which are the responsibility of the COL applicant.	
1.68.3	Preoperational Testing of Instrument and Control Air Systems	1	Partially Conforms	Applicable	This guidance is applicable except for site-specific aspects, including test performance and records retention, which are the responsibility of the COL applicant.	9.3
1.69	Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants	1	Partially Conforms	Applicable	This guidance is applicable to the design of concrete radiation shields. Site-specific aspects of this guidance, including development and implementation of a radiation shield test program, are the responsibility of the COL applicant.	12.3 9.3
1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)	3	Not Applicable	Not Applicable	RG 1.206 and NuScale Design Specific Review Standards (DSRS) are used.	Not Applicable
1.71	Welder Qualification for Areas of Limited Accessibility	1	Partially Conforms	Applicable	This guidance is applicable except for site-specific aspects, including specification of standards for weld fabrication and repair that are performed during construction, installation, and operation of a nuclear facility, which are the responsibility of the COL applicant.	4.5 5.2 5.3 5.4 6.1 10.3
1.72	Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin	2	Not Applicable	Not Applicable	The design does not involve the use of fiberglass piping in spray pond applications (or for that matter as part of the UHS design).	Not Applicable
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants	1	Conforms	Applicable	None.	3.11

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.75	Criteria for Independence of Electrical Safety Systems	3	Conforms	Applicable	None.	4.6 7.1 7.2 8.3 9.5
1.76	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants	1	Conforms	Applicable	Region 1 (bounding) characteristics are used as design parameters.	3.3 3.5 3.8
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors	-	Partially Conforms	Applicable	Portions of this RG pertain to assumptions for radiological consequence analysis. Per SRP Section 15.0.3, Section I, Areas of Review, Item 10 under subheading Review Interfaces, for the review of design certification applications, SRP Section 15.0.3 supersedes the radiological analyses, assumptions, acceptance criteria, and methodologies identified in SRP Section 15.4.8, which references guidance in RG 1.77.  The NRC has identified this RG as out of date, and in need of revision. The fuel and cladding failure criteria are superseded by the criteria provided in 4.2, Appendix B. The radiological criteria are superseded by the criteria in RG 1.183. However, the general approach and intent of RG 1.77 still apply and are used in Section 15.4.8 analyses.	4.3 15.0.3 15.4
1.78	Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	1	Partially Conforms	Applicable	Aspects of this RG related to control room habitability design within the scope of the NuScale design are applicable. Other aspects of this guidance require site-specific information (e.g., amount and location of toxic chemicals relative to the control room, and site-specific atmospheric dispersion factors) or specify operational, programmatic emergency planning activities. These aspects are the responsibility of the COL applicant.	6.4



Table 1.9-2: Conformance with Regulatory Guides (Continued)

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors	2	Partially Conforms	Applicable	The intent of this RG is applicable to the NuScale design, but the literal language refers to SSC design features not in the NuScale design. For example, the ECCS design does not use high pressure or low pressure safety injection pumps as described in this guidance. Rather, the ECCS design provides core decay heat removal by steam condensation and natural reactor coolant recirculation. Nevertheless, preoperational testing will be performed on the ECCS in a manner that satisfies the intent of this guidance. Much of this RG prescribes preoperational test implementation activities that are the responsibility of the COL applicant.	6.3
1.79.1	Initial Test Program of Emergency Core Cooling Systems for New Boiling-Water Reactors	-	Not Applicable	Not Applicable	RG 1.79.1 is applicable to BWRs only. However, the intent of this RG is applicable to the NuScale design, but the literal language refers to SSC design features not in the NuScale design.	Not Applicable
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants	1	Partially Conforms	Applicable	Portions of the onsite AC power system (including the non-Class1E backup diesel generators) are shared by the NPMs. Sharing of the EDSS-C satisfies the intent of RG 1.81.	8.3

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.82	Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident	4	Partially Conforms	Partially Applicable	<p>The NuScale design complies with the intent of RG 1.82 regulatory positions that address the design criteria, performance standards, and analysis methods related to water sources for long-term cooling. Although, the NuScale design is significantly different from the system designs the guidance was written to address.</p> <p>The NuScale design complies with the guidance with respect to debris generation, debris transport, coating debris, latent debris, downstream and chemical effects. The NuScale design is passive and does not include pumps, sumps, suction strainers, debris interceptor or trash racks and the design minimizes or negates the potential effect of non-condensables on coolant flow to the core. The NuScale design also does not require operator action to mitigate debris accumulation.</p> <p>The NuScale design does not comply with regulatory position C1.1 with the exception that NuScale does comply with the intent of the following regulatory positions:</p> <ul style="list-style-type: none"> <li>• Position C1.1.1.9 (assessment of the possibility of downstream clogging); and, position C1.1.1.10 (buildup of debris and chemical reaction products downstream).</li> <li>• Position C.1.1.2 (minimization of debris source term, cleanliness programs, monitoring/sampling for latent debris, insulation selection, restriction on coatings and cladding of carbon steel).</li> </ul>	6.3

Table 1.9-2: Conformance with Regulatory Guides (Continued)

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
					<p>Positions C1.1.3 and C1.1.4 are not applicable because the NuScale design does not rely on operator action to mitigate the consequences of debris accumulation and does not include active devices or systems to prevent debris accumulation.</p> <p>The NuScale design does not comply with regulatory position C.1.2 (alternative water sources for inoperable strainers).</p> <p>NuScale complies with the intent of regulatory position C.1.3 (evaluation of long term recirculation capability as applicable to the design) with the exception of the following:</p> <ul style="list-style-type: none"> <li>• Position C.1.3.1 (NPSH)</li> <li>• Portions of position C.1.3.2 that are not consistent with the NuScale design</li> </ul> <p>The NuScale design does not comply with regulatory positions C1.3.7 (upstream effects) or C.1.3.9 (strainer structural integrity).</p> <p>The NuScale design does not comply with regulatory position C1.3.12 (prototypical head loss testing).</p> <p>The NuScale design does not comply with regulatory position C.2 with the exception that the intent of chemical reaction effects (position 2.2) is met.</p> <p>The NuScale design does not comply with regulatory position C.3.</p>	
1.84	Design, Fabrication, and Materials Code Case Acceptability, ASME Section III	36	Conforms	Applicable	None.	3.12 3.13 4.5 5.2 5.4 6.1 10.3

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.86	Termination of Operating Licenses for Nuclear Reactors	-	Not Applicable	Applicable	This RG governs the process for terminating nuclear reactor operating licenses.	Not Applicable
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)	1	Not Applicable	Not Applicable	This RG applies to elevated-temperature reactors such as high-temperature gas-cooled reactors, liquid-metal fast-breeder reactors, and gas-cooled fast-breeder reactors.	Not Applicable
1.89	Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants	1	Partially Conforms	Applicable	This RG is applicable except for: (1) aspects that are BWR-specific or related to SSCs that are not relevant to the NuScale design (e.g., ice condenser containment, containment spray system, etc.); and (2) reference to RG 1.4 for source term, since the source term provisions of RG 1.4 have been superseded by RG 1.183 for new reactors.	3.11
1.90	Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons	2	Not Applicable	Not Applicable	This RG is applicable only to LWR designs that incorporate a pre-stressed concrete containment structure with grouted tendons. The NuScale containment vessel is steel (i.e., does not use concrete or grouted tendons in its design).	Not Applicable
1.91	Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants	2	Not Applicable	Applicable	This guidance governs the performance of site-specific evaluations and is the responsibility of the COL applicant.	2.2
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis	3	Conforms	Applicable	None.	3.7 3.8 3.9 3.10 3.12
1.93	Availability of Electric Power Sources	1	Not Applicable	Not Applicable	This RG is not identified as an applicable RG in DSRS Section 8.1.	Ch 8

Table 1.9-2: Conformance with Regulatory Guides (Continued)

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants	1	Not Applicable	Not Applicable	This RG is applicable only to BWR designs.	Not Applicable
1.97	Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants	4	Partially Conforms	Applicable	The power supply requirements in Section 6.6 of IEEE Std 497-2002 for Type B and C variables are satisfied with highly reliable power rather than Class 1E power in the NuScale design. The absence of active mitigation equipment eliminates the need for function restoration guides to address potential failures of active accident mitigation equipment. Consequently, the importance of Type B or C variables is diminished for the NuScale design. Regulatory Position C(1), is applicable only to current operating reactor licensees that choose to voluntarily convert to RG 1.97, Rev. 4.	3.11 6.2 7.2 11.5 11.6 12.3 12.5
1.98	Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor (for Comment)	-	Not Applicable	Not Applicable	This RG is applicable only to BWR designs.	Not Applicable
1.99	Radiation Embrittlement of Reactor Vessel Materials	2	Conforms	Applicable	None.	5.3 6.1
1.100	Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants	3	Partially Conforms	Applicable	This RG is applicable except for aspects related to: (1) when site-specific spectra exceed the certified design spectra (e.g., Position C1.2.1.g); and (2) qualification of new and replacement equipment in older unresolved safety issue A46 plants (e.g., Position C.1.2.2.j). Not applicable to electrical equipment. Site-specific guidance is the responsibility of the COL applicant.	3.10 3.11 7.2 8.3

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.101	Emergency Response Planning and Preparedness for Nuclear Power Reactors	5	Not Applicable	Applicable	This RG is limited to providing emergency response guidance for co-located licensees. As such, this RG is the responsibility of the COL applicant proposing to site a power plant such that the definition of co-located is met. Since RG 1.101, Revision 4, is the most current revision that endorses NUREG-0654/FEMA-REP-1, Revision 1, Revision 4 of RG 1.101 is applicable to the extent that it endorses (through NUREG-06554/FEMA-REP-1) the design-specific aspects of NUREG-0696.	Not Applicable
1.102	Flood Protection for Nuclear Power Plants	1	Not Applicable	Not Applicable	The design assumes the NPP is located above the maximum flood height (including wind induced wave run-up).	2.4 3.4
1.105	Setpoints for Safety-Related Instrumentation	3	Partially Conforms	Applicable	Chapter 15 analyses utilize the safety-related setpoints described in Chapter 7.	7.2 15.1 15.2 15.4 15.5 15.6 15.7
1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves	2	Not Applicable	Not Applicable	This RG governs the application of thermal overload protection devices to ensure that safety-related motor-operated valves perform their safety function. The design does not use safety-related motor operated valves.	Not Applicable
1.107	Qualification for Cement Grouting for Prestressing Tendons in Containment Structures	2	Not Applicable	Not Applicable	This RG is applicable only to LWR designs that use a prestressed concrete containment structure. The containment vessel is a steel containment (i.e., does not use concrete or prestressed tendons in its design).	Not Applicable

Table 1.9-2: Conformance with Regulatory Guides (Continued)

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFRPart50, Appendix I	1	Partially Conforms	Applicable	This RG is applicable except for specification of site-specific information (e.g., meteorological data). Site-specific guidance is the responsibility of the COL applicant.	2.3 11.2 11.3 11.5
1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors (for Comment)	1	Partially Conforms	Applicable	This RG is applicable except for aspects related to performance of a site-specific cost-benefit analysis. Site-specific guidance is the responsibility of the COL applicant.	11.2 11.3 11.4
1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors	1	Partially Conforms	Applicable	This RG is applicable except for specification of site-specific dispersion data. Site-specific guidance is the responsibility of the COL applicant.	2.3 11.3
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Nuclear Power Reactors	1	Partially Conforms	Applicable	This RG is applicable except for specification of site-specific information (e.g., meteorological data). Site-specific guidance is the responsibility of the COL applicant.	2.3 11.1 11.2 11.3 12.2
1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I	1	Partially Conforms	Applicable	This RG is applicable except for specification of site-specific dispersion data. Site-specific guidance is the responsibility of the COL applicant.	11.3

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.114	Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit	3	Partially Conforms	Applicable	Site-specific guidance is the responsibility of the COL applicant. Consistent with the discussion in RG 1.114, Section B.1, the ability of the COL applicant to meet this guidance is facilitated by the control room design and layout (including the designated surveillance area described in Position C.1.3). Due to advanced design and operational features unique to the NuScale power plant, portions of this guidance that implement operator staffing requirements of 10CFR50.54(m)(2)(i) and (iii) (e.g., Position C.1.5) are not applicable to COL applicants. It is more appropriate that the operating organization be based on these advanced plant design features rather than on staffing levels prescribed in 10CFR50.54(m)(2)(i) and (iii).	18.5 18.7
1.115	Protection Against Low-Trajectory Turbine Missiles	2	Not Applicable	Applicable	Site-specific guidance is the responsibility of the COL applicant.	3.5 9.1 10.1 10.2 10.3
1.117	Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants	2	Conforms	Applicable	None.	3.5 9.1
1.118	Periodic Testing of Electric Power and Protection Systems	3	Partially Conforms	Applicable	Site-specific guidance is the responsibility of the COL applicant.	7.2 8.3
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes (for Comment)	-	Conforms	Applicable	None.	5.4
1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components	1	Conforms	Applicable	None.	3.7



**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Supports	2	Conforms	Applicable	None.	3.9
1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants	2	Not Applicable	Not Applicable	The design does not require hydraulic structures.	Not Applicable
1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification	2	Conforms	Applicable	None.	4.2
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants	1	Not Applicable	Applicable	This guidance governs the development of an inservice inspection and surveillance program for dams, slopes, canals, and other water-control structures associated with emergency cooling water systems or flood protection of nuclear power plants. Water control structures and associated inservice inspection and surveillance programs are site-specific details. Site-specific guidance is the responsibility of the COL applicant.	21.2
1.128	Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	2	Partially Conforms	Applicable	The Regulatory Positions regarding VLA battery installations are appropriately considered in the installation design of the VRLA batteries.	8.3
1.129	Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	3	Partially Conforms	Applicable	The Regulatory Positions regarding VLA battery installations are appropriately considered for the VRLA batteries.	8.3
1.130	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Supports	3	Conforms	Applicable	None.	3.9
1.132	Site Investigations for Foundations of Nuclear Power Plants	2	Not Applicable	Applicable	This RG governs site investigations performed as part of site selection. Site-specific guidance is the responsibility of the COL applicant.	2.5

Table 1.9-2: Conformance with Regulatory Guides (Continued)

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.133	Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors	1	Departure	Applicable	The low fluid velocities resulting from natural circulation flow combined with a design that has only small lines entering the RPV minimizes the potential for loose parts entering or being generated in the RPV.	4.4
1.134	Medical Evaluation of Licensed Personnel at Nuclear Power Plants	3	Not Applicable	Applicable	This RG governs site-specific operational program activities. Site-specific guidance is the responsibility of the COL applicant.	Not Applicable
1.136	Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments	3	Not Applicable	Not Applicable	This guidance is applicable only to LWR designs that use concrete containments. The design uses a steel containment vessel as opposed to a concrete containment.	3.8
1.137	Fuel-Oil Systems for Standby Diesel Generators	2	Not Applicable	Not Applicable	The design does not rely on or include safety-related EDGs.	8.3
1.138	Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants	2	Not Applicable	Applicable	This guidance is related to site-specific laboratory investigation activities. Site-specific guidance is the responsibility of the COL applicant.	2.5
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	2	Partially Conforms	Applicable	Design-related aspects of this guidance are applicable. Aspects related to construction, testing, and repairs are the responsibility of the COL applicant. Recent versions of a number of codes and standards endorsed in this RG will be used. Code reconciliation is performed as appropriate and necessary to support the use of these later codes, unless otherwise endorsed by NRC guidance.	9.4 11.3 12.3
1.141	Containment Isolation Provisions for Fluid Systems	1	Conforms	Applicable	None.	6.2

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)	2	Partially Conforms	Applicable	The intent of this guidance is applicable but the specific language endorses ACI 349-1997 with specified exceptions. The 2006 version of the ACI 349 standard has been used. Aspects of Regulatory Positions C.1 and C.14 related to concrete structures within containment are not applicable to the design. The containment vessel is a steel component (as opposed to a typical structure), and does not involve the use of interior concrete structures.	3.8
1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants	2	Partially Conforms	Applicable	The aspects of this RG related to steam generator blowdown systems are not applicable to the design. Radioactive waste management system design criteria specified in this RG are applicable. Construction, installation, and testing criteria the responsibility of the COL applicant.	3.2 3.3 3.4 3.8 11.2 11.3 11.4 11.6 12.2
1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants	1	Not Applicable	Not Applicable	This RG does not include modeling of building wake effects. For the short distances that may be used for EAB and LPZ, Regulatory Guide 1.194 is used to determine representative atmospheric dispersion factors.	2.3
1.147	Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1	17	Partially Conforms	Applicable	Performance of inservice inspections per the ASME BPVC is the responsibility of the COL applicant.	5.2 6.6
1.149	Nuclear Power Plant Simulation Facilities for Use in Operator Training, License Examinations, and Applicant Experience Requirements	4	Not Applicable	Applicable	Simulation facilities and conduct of licensed operator training and qualification are the responsibility of the COL applicant.	Not Applicable

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.151	Instrument Sensing Lines	1	Partially Conforms	Applicable	This RG governs design and installation of safety-related instrument sensing lines in nuclear power plants. The aspects of this RG regarding installation criteria are the responsibility of the COL applicant.	7.2
1.152	Criteria for Use of Computers in Safety Systems of Nuclear Power Plants	3	Partially Conforms	Applicable	The NuScale I&C development lifecycle is different than that of the conceptual waterfall lifecycle listed in the RG. The applicable tasks from the RG lifecycle model will be mapped to the I&C development lifecycle. Compliance with Clause 5.5 of IEEE 7-4.3.2 is conditioned by the choice of field programmable gate array technology, which makes some tests not applicable (e.g., calculation tests, watchdog timer tests).	3.11 7.1 7.2
1.153	Criteria for Safety Systems	1	Conforms	Applicable	Applicable to EDSS.	3.11 8.3
1.155	Station Blackout	1	Partially Conforms	Applicable	The design conforms to the aspects of the RG as it pertains to passive plant designs.	5.4 6.2 8.4 9.3 9.5 10.3
1.156	Qualification of Connection Assemblies for Nuclear Power Plants	1	Conforms	Applicable	None.	3.11
1.157	Best-Estimate Calculations of Emergency Core Cooling System Performance	-	Not Applicable	Not Applicable	Best estimate calculations are not used.	Not Applicable
1.158	Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants	-	Not Applicable	Not Applicable	The DC system batteries are non-Class 1E.	3.11 8.3
1.159	Assuring the Availability of Funds for Decommissioning Nuclear Reactors	2	Not Applicable	Applicable	Decommissioning funding activities are the responsibility of the COL applicant.	Not Applicable

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	3	Not Applicable	Applicable	Monitoring the effectiveness of maintenance activities is the responsibility of the COL applicant.	17.6
1.161	Evaluation of Reactor Pressure Vessels with Charpy UpperShelf Energy Less Than 50 Ft-Lb	-	Not Applicable	Applicable	The Charpy upper-shelf energy of the NuScale reactor vessel materials will exceed the 50 ft-lb energy value (throughout the life of the vessel with significant margin) below which this guidance would apply. However, in the unlikely event the reactor vessel material surveillance program implemented during reactor operations indicates that this is not the case, the requirements of 10CFR50, Appendix G and the provisions of RG 1.161 would be the responsibility of the COL applicant (see discussion of RG1.162).	Not Applicable
1.162	Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels	-	Not Applicable	Applicable	If thermal annealing becomes necessary, the requirements of 10CFR50.66 and the provisions of RG1.162 would be the responsibility of the COL applicant. Annealing of the reactor vessel at site for radiation embrittlement is not anticipated to be necessary, as the core region materials for the vessel are predicted to maintain: the pressurized thermal shock, RTPTS, criteria of 2,700F and 3,000F as specified in 10CFR50.61(b)(2); End-of-Life Charpy upper-shelf energy of 50 ft-lb as specified in 10CFR50, Appendix G, Paragraph IV.1; and Operational flexibility to comfortably maintain the pressure-temperature limits as specified as Table1 of 10CFR50, Appendix G, during normal operation. These limits are expected to be observed with significant margins throughout the design life of the reactor vessel. It also should be noted that with the significant improvement in the low alloy steel forging technology in the recent years, the initial RTNDT values of the reactor vessel core region forgings are expected to be near or below 0F.	Not Applicable

Table 1.9-2: Conformance with Regulatory Guides (Continued)

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.163	Performance-Based Containment Leak-Test Program	-	Partially Conforms	Applicable	NuScale intends to apply this RG, as appropriate, in its development of an acceptable performance-based leak-test program in compliance with the performance-based Option B in Appendix J to 10CFR50. However, there may be certain aspects of this guidance for which, due to potential unique hazard and risk consideration, NuScale may propose alternate methods. For example, the ANSI standard and NEI guidance endorsed by this RG specify pneumatic testing of the containment vessel that would of necessity be performed in the reactor pool. Departures from this guidance will depend on NuScale's possible decision to propose a more risk-informed performance-based approach to containment leak-testing.'	6.2
1.166	Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post Earthquake Actions	-	Not Applicable	Applicable	This RG governs programmatic activities (earthquake planning and post-earthquake actions) that are the responsibility of the COL applicant.	3.7
1.167	Restart of a Nuclear Power Plant Shut Down by a Seismic Event	-	Not Applicable	Applicable	This RG governs post-earthquake inspections and tests that are the responsibility of the COL applicant.	Not Applicable

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.168	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	2	Partially Conforms	Applicable	This RG refers to Revision 1 of RG 1.152 as containing NRC endorsement of IEEE Std. 7 4.3.2-1993. The current Revision 3 of RG 1.152 endorses (with exceptions) IEEE Std. 7 4.3.2-2003. NuScale intends to apply RG 1.152, Revision 3 (unless superseded by a newer revision), and the 2003 version of IEEE Std. 7-4.3.2 that it endorses. For RG 1.168, the requirements of IEEE 1012 are tailored to the NuScale I&C development lifecycle, which is different than that of the conceptual waterfall lifecycle listed in IEEE 1012. The applicable tasks from IEEE 1012 to the I&C development lifecycle will be mapped. Some of the administrative mandatory requirements in the standard conflict with established Engineering or QA documentation requirements. The requirements of IEEE 1028 are tailored to the NuScale I&C development lifecycle. NuScale will identify the specific tasks IEEE 1028 to the NuScale I&C development lifecycle. NuScale will also take exception to some of the administrative mandatory requirements in the standard that conflict with established Engineering or quality documentation requirements.	7.2
1.169	Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	Partially Conforms	Applicable	For this RG, the requirements of IEEE 828 are tailored to the NuScale I&C development lifecycle, which is different than that of the conceptual waterfall lifecycle listed in RG 1.152. The applicable tasks from IEEE 828 will be mapped to the NuScale I&C development lifecycle. NuScale will also take exception to some of the administrative mandatory requirements in the standard that conflict with established Engineering or quality documentation requirements.	7.2

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.170	Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	Partially Conforms	Applicable	For this RG, the requirements of IEEE 829 are tailored to the NuScale I&C development lifecycle, which is different than that of the conceptual waterfall lifecycle listed in RG 1.152. The applicable tasks from IEEE 829 will be mapped to the NuScale I&C development lifecycle. NuScale will also take exception to some of the administrative mandatory requirements in the standard that conflict with established Engineering or quality documentation requirements.	7.2
1.171	Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	Partially Conforms	Applicable	For this RG, NuScale will take exception to some of the administrative mandatory requirements in IEEE 1008 that conflict with established Engineering or quality documentation requirements.	7.2
1.172	Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	Partially Conforms	Applicable	For this RG, NuScale will take exception to some of the administrative mandatory requirements in IEEE 830 standard that conflict with established Engineering or quality documentation requirements.	7.2
1.173	Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	Partially Conforms	Applicable	For this RG, the requirements of IEEE 1074 are tailored to the NuScale I&C development lifecycle, which is different than that of the conceptual waterfall lifecycle listed in RG 1.152. NuScale will map the applicable tasks from IEEE 1074 to the NuScale I&C development lifecycle. NuScale will also take exception to some of the administrative mandatory requirements in the standard that conflict with established Engineering or quality documentation requirements.	7.2
1.174	An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis	2	Not Applicable	Applicable	Applicable to changes in licensing basis; thus, not directly used for design certification.	Not Applicable



**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.175	An Approach for Plant-Specific, Risk Informed Decision making: Inservice Testing	-	Not Applicable	Applicable	This RG is applicable to licensees seeking change to licensing basis and is the responsibility of the COL applicant. NuScale is not using a risk-informed approach for ISI.	Not Applicable
1.177	An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications (Draft DG-1287)	1	Partially Conforms	Applicable	This RG is directed explicitly towards existing licensees seeking NRC approval of changes to their plant-specific technical specifications. Notwithstanding, NuScale intends to consider this guidance, as appropriate, in risk-informed decision making.	16.1
1.178	An Approach for Plant-Specific Risk-Informed Decision making for Inservice Inspection of Piping (Draft DG-1288)	1	Not Applicable	Applicable	This RG addresses the use of PRA in support of a risk-informed inservice inspection program for piping. Such a program is a plant-specific operational program that is the responsibility of the COL applicant.	Not Applicable
1.179	Standard Format and Content of License Termination Plans for Nuclear Power Reactors	1	Not Applicable	Applicable	This guidance governs site-specific decommissioning and license termination planning and implementation activities that are the responsibility of the COL applicant.	Not Applicable
1.180	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems	1	Partially Conforms	Applicable	Aspects of this guidance related to the design of SSCs to address effects of electromagnetic and radio-frequency interference (EMI/RFI) are applicable. Aspects of this guidance related to the design of site-specific SSCs and installation and testing practices for addressing the effects of EMI/RFI and power surges on safety-related I&C systems are the responsibility of the COL applicant.	3.11 7.2 9.5
1.181	Content of the Updated Final Safety Analysis Report in Accordance with 10CFR50.71(e)	-	Not Applicable	Applicable	This guidance governs site-specific reporting activities that are the responsibility of the COL applicant.	Not Applicable

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.183	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors (Draft DG-1199)	-	Partially Conforms	Applicable	For the typical large LWR, the limiting dose consequence analysis corresponds to the design basis LOCA; however, for the NuScale design, core damage is not expected for a design basis LOCA event. Thus, the RG1.183 guidance will only be partially applicable to the NuScale LOCA dose consequence analysis. The basis and justification for departures from the RG1.183 guidance for the limiting LOCA dose consequence analysis for NuScale are provided in a Technical Report. Notwithstanding the above, NuScale will use the alternative source term non-LOCA or transient-specific guidance of RG1.183 for Chapter 15 events that do not result in core damage.	6.4 9.3 12.2 15.0.2 15.0.3 15.6 15.7
1.184	Decommissioning of Nuclear Power Reactors	1	Not Applicable	Applicable	This RG governs site-specific decommissioning planning and implementation activities that are the responsibility of the COL applicant.	Not Applicable
1.185	Standard Format and Content for Post-Shutdown Decommissioning Activities Report	1	Not Applicable	Applicable	This RG governs site-specific decommissioning planning activities that are the responsibility of the COL applicant.	Not Applicable
1.186	Guidance and Examples for Identifying 10CFR50.2 Design Bases	-	Not Applicable	Not Applicable	This RG endorses NEI 97-04 Appendix B and provides non-mandatory guidance for operating reactor licensees in defining what constitutes design basis information and is the responsibility of the COL applicant.	Not Applicable
1.187	Guidance for Implementation of 10CFR50.59, Changes, Tests, and Experiments	-	Not Applicable	Applicable	This RG implements change process requirements that are the responsibility of the COL applicant.	Not Applicable
1.188	Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses	1	Not Applicable	Not Applicable	This RG is applicable only to operating reactor licensees seeking to renew their operating licenses.	Not Applicable

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.189	Fire Protection for Nuclear Power Plants	2	Partially Conforms	Applicable	This RG is applicable except for (1) directed toward a specific reactor design (e.g., BWR or non-LWR) or SSC conditions not relevant to the NuScale PWR design; and (2) related to site-specific fire protection systems and equipment or programmatic and procedural activities that are the responsibility of the COL applicant.	9.5 11.3 19.1
1.190	Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence	-	Partially Conforms	Applicable	None.	5.3
1.191	Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown	-	Not Applicable	Applicable	This RG governs site-specific fire protection program activities that are applicable only to holders of reactor licenses that have permanently ceased power operations.	Not Applicable
1.192	Operation and Maintenance Code Case Acceptability, ASME OM Code	1	Not Applicable	Applicable	Implementation of in-service testing per this code is the responsibility of the COL applicant.	3.9 5.2
1.193	ASME Code Cases Not Approved for Use	4	Conforms	Applicable	ASME code cases listed in RG 1.193 are not used unless authorized by the NRC pursuant to the requirements of 10 CFR 50.55a(z).	5.2
1.194	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants	-	Conforms	Applicable	None.	2.3 9.4 15.0.3
1.195	Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors	-	Not Applicable	Not Applicable	This RG pertains to TID14844 source terms used by licensees of existing reactors that are not authorized to use the alternative source term under 10CFR50.67. Therefore, RG 1.183 is specified to be used in lieu of RG 1.195 for new reactors and existing reactors authorized to use the alternative source term under 10CFR50.67.	Not Applicable

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.196	Control Room Habitability at Light-Water Nuclear Power Reactors	1	Partially Conforms	Applicable	Aspects of this RG related to control room habitability design within the scope of the standard plant design are applicable to the DCA. References to ESF ventilation systems are not applicable to the NuScale design. The NuScale control room habitability system neither relies on nor uses emergency filtration to protect operators during accident conditions. Rather, clean air is provided using compressed air tanks. Other aspects of this guidance specify operational, programmatic activities that are the responsibility of the COL applicant. These aspects include but are not limited to maintenance, configuration control, training, and post-construction comparison of system design, configuration, and operation with the plant licensing basis. The NuScale design will facilitate the ability of the COL applicant referencing the certified design to meet these aspects.	6.4
1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors	-	Partially Conforms	Applicable	This RG provides an acceptable approach for measuring inleakage into the control room envelope at nuclear power reactors to ensure that the control room is habitable during normal and accident conditions. These inleakage testing activities are outside the scope of the certified design, and are the responsibility of the COL applicant referencing the certified design. Notwithstanding, the NuScale design will facilitate the ability of the COL applicant referencing the NuScale design to satisfy this guidance.	6.4
1.198	Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites	-	Not Applicable	Applicable	This RG governs evaluation activities that require site-specific information not available to a design certification applicant. The evaluation governed by this guidance is the responsibility of the COL applicant referencing the certified design.	Not Applicable

Table 1.9-2: Conformance with Regulatory Guides (Continued)

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.199	Anchoring Components and Structural Supports in Concrete	-	Partially Conforms	Applicable	The intent of this guidance is applicable but the specific language endorses Appendix B of ACI 349-2001 with specified exceptions in the area of load combinations. NuScale uses the 2006 version of the ACI 349 standard.	3.8
1.200	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities	2	Conforms	Applicable	As referenced in SRP 19.0 with regard to PRA quality and technical adequacy.	19.1

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

<b>RG</b>	<b>Division Title</b>	<b>Rev.</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
1.201	Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance	1	Partially Conforms	Applicable	10CFR50.69 provides an alternative regulatory framework wherein a licensee may use a risk-informed process for categorizing SSCs according to their safety significance, and based on this process can remove SSCs of low safety significance from the scope of certain identified special treatment requirements. Thus, these requirements are applicable to licensees that choose this alternative framework. NuScale intends to use a risk-informed, performance-based approach to safety classification that blends the strengths of deterministic engineering judgment and probabilistic methods. Specifically, the NuScale approach to SSC safety classification will combine the traditional approach using the definitions of 10CFR50.2 and guidance of RG 1.26 and SRP Section 3.2.2, with the alternative regulatory framework similar to that prescribed in 10CFR50.69 and RG 1.201 (and NEI 00-04 endorsed by RG 1.201). This is consistent with SECY-03-0047 and SECY-10-0034, which recommend the use of a probabilistic, risk-informed approach for SSC safety classification. In this process, NuScale intends to apply the guidance of RG 1.201 and NEI 00-04 to the extent appropriate given the baseline risk metrics specific to the NuScale advanced reactor design.	3.2
1.202	Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors	-	Not Applicable	Applicable	This RG implements regulatory requirements for decommissioning cost estimates that are applicable only to licensees.	Not Applicable
1.203	Transient and Accident Analysis Methods	-	Conforms	Applicable	None.	15.0.2

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.204	Guidelines for Lightning Protection of Nuclear Power Plants	-	Partially Conforms	Applicable	The grounding and lightning protection systems are designed, installed, tested, and maintained in conformance with RG 1.204, with the exception that where IEEE Std. 666-1991 (Reaffirmed 1996) and IEEE Std. 1050-1996 are specified, IEEE Std. 666-2007 and IEEE Std. 1050-2004 instead are applied. Reconciliation of the two versions of each standard demonstrates the acceptability of the use of the later versions.	7.2 8.3
1.205	Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants	1	Not Applicable	Applicable	This RG is applicable only to reactor licensees or applicants that are developing or revising a risk-informed, performance-based fire protection program pursuant to 10CFR50.48(c). Development and implementation of a risk-informed, performance-based fire protection program would be the responsibility of COL applicants that reference the NuScale design, and that elect to implement the provisions of 10CFR50.48(c).	Not Applicable
1.206	Combined License Applications for Nuclear Power Plants (LWR Edition)	-	Partially Conforms	Applicable	This RG is used as the general template for the FSAR layout, with exceptions.	Ch. 1 through Ch. 19
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	-	Conforms	Applicable	None.	3.9 3.12 6.1
1.208	A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion	-	Not Applicable	Applicable	This guidance is related to the development of site-specific ground motion response spectra.	2.5

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.209	Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants	-	Conforms	Applicable	None.	3.11 App 3C 7.2
1.210	Qualification of Safety-Related Battery Chargers and Inverters for Nuclear Power Plants	-	Not Applicable	Not Applicable	No safety-related battery chargers or inverters; EDSS battery chargers are not located in a harsh environment.	N/A
1.211	Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants	-	Conforms	Applicable	None.	3.11
1.212	Sizing of Large Lead-Acid Storage Batteries	November 2008	Partially Conforms	Applicable	The NuScale DC power systems conform to the VRLA sizing guidance in IEEE Std. 485-1997, with consideration as appropriate for regulatory positions in RG 1.212 relevant to VRLA battery sizing.	8.3
1.213	Qualification of Safety-Related Motor Control Centers for Nuclear Power Plants	-	Not Applicable	Not Applicable	The NuScale electrical system design does not use safety-related motor control centers.	N/A
1.214	Response Strategies for Potential Aircraft Threats	1	Conforms	Applicable	Not publicly available.	19.5
1.215	Guidance for ITAAC Closure Under 10CFRPart52	2	Not Applicable	Applicable	This guidance describes acceptable methods of complying with the requirements of 10CFR52.99, which is applicable to COL applicants.	14.3
1.216	Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design-Basis Pressure	0	Conforms	Applicable	None.	3.8 6.2 19.2
1.217	Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts	-	Conforms	Applicable	None.	19.5 21.1



Table 1.9-2: Conformance with Regulatory Guides (Continued)

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.218	Condition-Monitoring Techniques for Electric Cables Used in Nuclear Power Plants	-	Not Applicable	Applicable	This guidance governs electric cable monitoring program activities that are not within the scope of design certification. Rather, these activities are the responsibility of and applicable to operating reactor licensees, including COL holders.	Not Applicable
1.219	Guidance on Making Changes to Emergency Plans for Nuclear Power Reactors	-	Not Applicable	Applicable	These requirements are applicable to operating reactor licensees, including COL holders.	Not Applicable
1.221	Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants	-	Conforms	Applicable	NuScale uses Region 1 (bounding) characteristics as design parameters.	3.3 3.5 3.8
1.226	Flexible Mitigation Strategies for Beyond-Design-Basis Events (Draft DG-1301)	-	Partially Conforms	Applicable	The RG, presently in draft, endorses, with clarifications, NEI 12-06 Rev 1A, Diverse and Flexible Coping Strategies (FLEX) Implementation Guide. NuScale is writing Chapter 20 meeting the applicable portions of the draft guidance. There is guidance in NEI 12-06 that is not applicable the NuScale design. These items are addressed in Chapter 20.	Ch 20
1.227	Wide-Range Spent Fuel Pool Level Instrumentation (Draft DG-1317)	-	Conforms	Applicable	The RG, presently in draft, endorses, with exceptions and clarifications, NEI 12-02 Rev 1, Industry Guidance for Compliance with NRC Order EA-12-051, To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation. NuScale is writing Chapter 20 meeting the applicable portions of the draft guidance. Spent fuel pool level instrumentation is addressed in Chapter 9.	Ch 20 9.2

Table 1.9-2: Conformance with Regulatory Guides (Continued)

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.228	Integrated Response Capabilities for Beyond-Design-Basis Events (Draft DG-1319)	-	Not Applicable	Applicable	The RG is presently in draft, it endorses, with clarifications, the following: <ul style="list-style-type: none"> <li>• NEI 12-01 Rev 0, Guidelines for Assessing Beyond-Design-Basis Accident Response Staffing and Communication Capabilities</li> <li>• NEI 13-06, Rev 0 Enhancements to Emergency Response Capabilities for Beyond-Design-Basis Events and Severe Accidents</li> <li>• NEI 14-01 Rev 0, Emergency Response Procedures and Guidelines for Beyond-Design-Basis Events and Severe Accidents</li> </ul> NuScale is writing Chapter 20 meeting the applicable portions of the draft guidance.	Ch 20
4.1	Radiological Environmental Monitoring for Nuclear Power Plants	2	Not Applicable	Applicable	This guidance governs site-specific, programmatic environmental monitoring activities that are the responsibility of the COL applicant.	Not Applicable
4.2	Preparation of Environmental Reports for Nuclear Power Stations	2	Not Applicable	Applicable	This guidance governs site-specific environmental evaluation activities that are the responsibility of a license or construction permit applicant.	Not Applicable
4.2S1	Supplement 1 to RG 4.2, Preparation of Supplemental Environmental Reports for Applications to Renew Nuclear Power Plant Operating Licenses	1	Not Applicable	Not Applicable	Revision 1 of RG 4.2S1 pending DG-4015 dated 7/2009 This guidance is Applicable only to licensees seeking renewal of their operating license.	Not Applicable
4.4	Reporting Procedure for Mathematical Models Selected to Predict Heated Effluent Dispersion in Natural Water Bodies	-	Not Applicable	Applicable	This guidance governs site-specific environmental activities related to modeling temperature impact of plant discharge on aquatic systems. These activities are not within the scope of the certified design, and are the responsibility of the COL applicant.	Not Applicable
4.7	General Site Suitability Criteria for Nuclear Power Stations	2	Not Applicable	Applicable	This guidance governs site-specific evaluation activities that are the responsibility of the COL applicant.	Not Applicable

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

<b>RG</b>	<b>Division Title</b>	<b>Rev.</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
4.9	Preparation of Environmental Reports for Commercial Uranium Enrichment Facilities	1	Not Applicable	Not Applicable	This guidance is Applicable only to uranium enrichment facilities.	Not Applicable
4.11	Terrestrial Environmental Studies for Nuclear Power Stations	2	Not Applicable	Applicable	This guidance governs site-specific environmental evaluation activities that are the responsibility of a license or construction permit applicant.	Not Applicable
4.13	Performance, Testing, and Procedural Specifications for Thermoluminescence Dosimetry: Environmental Applications	1	Not Applicable	Applicable	This guidance governs site-specific procedural activities that are the responsibility of a COL applicant or holder.	Not Applicable
4.14	Radiological Effluent and Environmental Monitoring at Uranium Mills	1	Not Applicable	Not Applicable	This guidance is Applicable only to uranium mills.	Not Applicable
4.15	Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) - Effluent Streams and the Environment	2	Not Applicable	Applicable	Applicable to COL applicants.	11.5
4.16	Monitoring and Reporting Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Cycle Facilities	2	Not Applicable	Not Applicable	This guidance is Applicable only to fuel cycle facilities.	Not Applicable
4.17	Standard Format and Content of Site Characterization Plans for High-Level-Waste Geologic Repositories	1	Not Applicable	Not Applicable	This guidance is Applicable only to geological repositories.	Not Applicable
4.18	Standard Format and Content of Environmental Reports for Near-Surface Disposal of Radioactive Waste	-	Not Applicable	Not Applicable	This guidance is Applicable only to near-surface disposal of radioactive waste, which is not within the scope of the NuScale certified design.	Not Applicable

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
4.19	Guidance for Selecting Sites for Near-Surface Disposal of Low-Level Radioactive Waste	-	Not Applicable	Not Applicable	This guidance is Applicable only to near-surface disposal of radioactive waste, which is not within the scope of the NuScale certified design.	Not Applicable
4.20	Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors	1	Not Applicable	Not Applicable	This guidance is Applicable only to non-reactor facilities.	Not Applicable
4.21	Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning	-	Partially Conforms	Applicable	This guidance is applicable, except for the portions that relate to site-specific, operational aspects that are the responsibility of the COL applicant referencing the NuScale design.	9.1 9.2 9.4 10.4 11.2 11.3 11.6 12.3.6 12.5
4.22	Decommissioning Planning During Operations	-	Not Applicable	Applicable	This RG is applicable to operating reactor licensees.	Not Applicable
5.3	Statistical Terminology and Notation for Special Nuclear Materials Control and Accountability	-	Not Applicable	Not Applicable	This RG is directed towards licensees of fuel processing and fuel fabrication facilities.	Not Applicable
5.4	Standard Analytical Methods for the Measurement of Uranium Tetrafluoride (UF <sub>4</sub> ) and Uranium Hexafluoride (UF <sub>6</sub> )	-	Not Applicable	Not Applicable	This RG is directed towards licensees of enrichment facilities.	Not Applicable
5.5	Standard Methods for Chemical, Mass Spectrometric, and Spectrochemical Analysis of Nuclear-Grade Uranium Dioxide Powders and Pellets	-	Not Applicable	Not Applicable	This RG is directed towards licensees of fuel fabrication facilities.	Not Applicable

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
5.7	Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas	1	Partially Conforms	Applicable	Site-specific, programmatic aspects of this guidance are the responsibility of the COL applicant referencing the NuScale design.	13.6 (via Security Technical Report)
5.8	Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Drying and Fluidized Bed Operations	1	Not Applicable	Not Applicable	Applicable to Part 70 facilities.	Not Applicable
5.9	Guidelines for Germanium Spectroscopy Systems for Measurement of Special Nuclear Material	2	Not Applicable	Not Applicable	This guidance governs processes, procedures, equipment, and methods that are not applicable to the NuScale reactor design.	Not Applicable
5.11	Nondestructive Assay of Special Nuclear Material Contained in Scrap and Waste	1	Not Applicable	Not Applicable	These RG details procedures acceptable to the NRC staff to provide a framework for the use of nondestructive assay in the measurement of scrap and waste components generated in conjunction with the processing of SNM. NuScale does not process SNM.	Not Applicable
5.12	General Use of Locks in the Protection and Controls of Facilities and Special Nuclear Materials	-	Partially Conforms	Applicable	Site-specific, programmatic aspects of this guidance are the responsibility of the COL applicant referencing the NuScale design.	13.6 (via Security Technical Report)
5.13	Conduct of Nuclear Material Physical Inventories	-	Not Applicable	Not Applicable	Applicable to Part 70 facilities.	Not Applicable
5.18	Limit of Error Concepts and Principles of Calculation in Nuclear Materials Control	-	Not Applicable	Not Applicable	This RG is not applicable to the DCA because NuScale is not a special nuclear material licensee.	Not Applicable
5.20	Training, Equipping, and Qualifying of Guards and Watchmen	-	Not Applicable	Applicable	Applicable to COL applicant.	Not Applicable
5.21	Nondestructive Uranium-235 Enrichment Assay by Gamma Ray Spectrometry	1	Not Applicable	Not Applicable	Applicable to Part 70 facilities.	Not Applicable
5.22	Assessment of the Assumption of Normality (Employing Individual Observed Values)	-	Not Applicable	Not Applicable	This RG is not applicable to the DCA because NuScale is not a special nuclear material licensee.	Not Applicable

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

<b>RG</b>	<b>Division Title</b>	<b>Rev.</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
5.23	In Situ Assay of Plutonium Residual Holdup	1	Not Applicable	Not Applicable	Applicable to Part 70 facilities.	Not Applicable
5.25	Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Equipment for Wet Process Operations	-	Not Applicable	Not Applicable	Applicable to Part 70 facilities.	Not Applicable
5.26	Selection of Material Balance Areas and Item Control Areas	1	Not Applicable	Not Applicable	Applicable to Part 70 facilities.	Not Applicable
5.27	Special Nuclear Material Doorway Monitors	-	Not Applicable	Applicable	Applicable to COL applicant.	Not Applicable
5.28	Evaluation of Shipper-Receiver Differences in the Transfer of Special Nuclear Materials	-	Not Applicable	Not Applicable	This RG is directed toward fuel processing and fuel fabrication licensees.	Not Applicable
5.29	Nuclear Material Control systems for Nuclear Power Plants	2	Not Applicable	Applicable	This RG is not applicable to the DCA but may be utilized by a COL applicant to meet the material control and accounting requirements in Subpart B of 10CFRPart74.	Not Applicable
5.31	Specially Designed Vehicle with Armed Guards for Road Shipment of Special Nuclear Material	1	Not Applicable	Applicable	Applicable to COL applicant.	Not Applicable
5.33	Statistical Evaluation of Material Unaccounted For	-	Not Applicable	Not Applicable	Applicable to Part 70 facilities.	Not Applicable
5.34	Nondestructive Assay for Plutonium in Scrap Material by Spontaneous Fission Detection	1	Not Applicable	Not Applicable	Applicable to Part 70 processing.	Not Applicable
5.36	Recommended Practice for Dealing with Outlying Observations	1	Not Applicable	Not Applicable	This RG is not applicable to the DCA because NuScale is not a special nuclear material licensee.	Not Applicable

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
5.39	General Methods for the Analysis of Uranyl Nitrate Solutions for Assay, Isotopic Distribution, and Impurity Determinations	-	Not Applicable	Not Applicable	This RG is not applicable to the DCA because NuScale is not an applicant for a special nuclear material in an unsealed form license.	Not Applicable
5.42	Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Equipment for Dry Process Operations	-	Not Applicable	Not Applicable	Applicable to Part 70 facilities.	Not Applicable
5.43	Plant Security Force Duties	-	Not Applicable	Applicable	Applicable to COL applicant.	Not Applicable
5.44	Perimeter Intrusion Alarm Systems	3	Partially Conforms	Applicable	Site-specific, programmatic aspects of this guidance are the responsibility of the COL applicant referencing the NuScale design.	13.6 (via Security Technical Report)
5.48	Design Considerations-Systems for Measuring the Mass of Liquids	-	Not Applicable	Applicable	Applicable to COL applicant.	Not Applicable
5.49	Internal Transfers of Special Nuclear Material (for Comment)	-	Not Applicable	Not Applicable	Issued for comment.	Not Applicable
5.51	Management Review of Nuclear Material Control and Accounting Systems (for Comment)	-	Not Applicable	Not Applicable	Issued for comment.	Not Applicable
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)	3	Not Applicable	Not Applicable	Not applicable to Nuclear Power Plants.	Not Applicable
5.53	Qualification, Calibration, and Error Estimation Methods for Nondestructive Assay	1	Not Applicable	Not Applicable	This RG is directed toward fuel processing licensees.	Not Applicable
5.54	Standard Format and Content of Safeguards Contingency Plans for Nuclear Power Plants	1	Not Applicable	Applicable	This guidance governs site-specific physical protection features and security program activities that are the responsibility of the COL applicant.	Not Applicable

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
5.55	Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities (for Comment)	-	Not Applicable	Applicable	Applicable to COL applicant.	Not Applicable
5.56	Standard Format and Content of Safeguards Contingency Plans for Transportation (for Comment)	-	Not Applicable	Applicable	Applicable to COL applicant.	Not Applicable
5.57	Shipping and Receiving Control of Strategic Special Nuclear Material	1	Not Applicable	Applicable	Applicable to COL applicant.	Not Applicable
5.58	Considerations for Establishing Traceability of Special Nuclear Material Accounting Measurements	1	Not Applicable	Applicable	Applicable to COL applicant.	Not Applicable
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	1	Not Applicable	Applicable	Applicable to COL applicant.	Not Applicable
5.60	Standard Format and Content of a Licensee Physical Protection Plan for Strategic Special Nuclear Material in Transit	-	Not Applicable	Applicable	Applicable to COL applicant.	Not Applicable
5.61	Intent and Scope of the Physical Protection Upgrade Rule Requirements for Fixed Sites	-	Not Applicable	Not Applicable	This guidance governs activities that are Applicable only to fuel cycle facilities.	Not Applicable
5.62	Reporting of Safeguards Events	1	Not Applicable	Applicable	This guidance governs reporting of site-specific security issues concerning SNM.	Not Applicable
5.63	Physical Protection for Transient Shipments	-	Not Applicable	Applicable	Applicable to COL applicant.	Not Applicable
5.65	Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls	-	Partially Conforms	Applicable	Site-specific, programmatic aspects of this guidance are the responsibility of the COL applicant referencing the NuScale design.	13.6 (via Security Technical Report)



**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
5.66	Access Authorization Program for Nuclear Power Plants	2	Not Applicable	Applicable	This guidance governs site-specific physical security program activities that are the responsibility of the COL applicant.	Not Applicable
5.68	Protection Against Malevolent Use of Vehicles at Nuclear Power Plants	-	Partially Conforms	Applicable	Although Applicable to the COL applicant, the design must allow compliance (e.g., F090 Site Layout Plan, which references parts of 73.55).	13.6 (via Security Technical Report)
5.69	Guidance for the Application of Radiological Sabotage Design-Basis Threat in the Design, Development and Implementation of a Physical Security Program that Meets 10CFR73.55 Requirements (SGI)	-	Partially Conforms	Applicable	COL applicant responsibility.	13.6 (via Security Technical Report)
5.70	Guidance for the Application of the Theft and Diversion Design-Basis Threat in the Design Development, and Implementation of a Physical Security Program that Meets CFR 73.45 and 73.46 (SGI)	-	Not Applicable	Applicable	COL applicant responsibility.	Not Applicable
5.71	Cyber Security Programs for Nuclear Facilities	-	Partially Conforms	Applicable	The portions of RG 5.71 that govern site-specific operational and programmatic activities (e.g., development and implementation of operational cyber security plans) are Applicable only to the COL applicant.	13.6 (via Security Technical Report)
5.73	Fatigue Management for Nuclear Power Plant Personnel	-	Not Applicable	Applicable	This RG is not applicable to the DCA but may be utilized by a COL applicant to meet the fatigue management requirements of 10CFR26 Subpart I.	Not Applicable
5.74	Managing the Safety/Security Interface	-	Not Applicable	Applicable	COL applicant responsibility.	Not Applicable
5.75	Training and Qualification of Security Personnel at Nuclear Power Reactor Facilities	-	Not Applicable	Applicable	COL applicant responsibility.	Not Applicable

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
5.76	Physical Protection Programs at Nuclear Power Reactors	-	Partially Conforms	Applicable	This guidance governs site-specific physical protection program activities that are the responsibility of the COL applicant.	13.6 (via Security Technical Report)
5.77	Insider Mitigation Program (OUO-SRI)	-	Not Applicable	Applicable	COL applicant responsibility.	Not Applicable
5.78	Physical Protection of Mixed Oxide Fuels in Nuclear Power Plants (SGI)	-	Not Applicable	Not Applicable	NuScale does not use mixed oxide fuels.	Not Applicable
5.79	Protection of Safeguard Information	-	Conforms	Applicable	None.	Not Applicable
5.80	Pressure-Sensitive and Tamper-Indicating Device Seals for Material Control and Accounting of Special Nuclear Material	-	Not Applicable	Not Applicable	This RG is not applicable to the DCA.	Not Applicable
5.81	Target Set Identification and Development for Nuclear Power Reactors (OUO-SRI)	-	Not Applicable	Applicable	COL applicant responsibility.	Not Applicable
5.83	Cyber Security Event Notifications	-	Not Applicable	Applicable	COL applicant responsibility.	13.6
5.84	Fitness-For-Duty for New Nuclear Power Plant Construction Sites	-	Not Applicable	Applicable	COL applicant responsibility.	13.7
8.2	Administrative Practices in Radiation Surveys and Monitoring	1	Not Applicable	Applicable	This guidance governs site-specific, programmatic activities related to radiation surveys and monitoring that are the responsibility of the COL applicant. However, the NuScale application for standard design certification will consider this guidance to the extent necessary to provide reasonable assurance that the COL applicant referencing the certified design can meet these requirements.	12.1 12.5

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
8.4	Personnel Monitoring Device - Direct-Reading Pocket Dosimeters	1	Not Applicable	Applicable	Revision 2 of RG 8.4 pending DG-8036 dated 4/2010. This guidance governs site-specific, programmatic activities related to the selection, maintenance, calibration, training, and reading of pocket dosimeters that are the responsibility of the COL applicant.	12.1 12.5
8.7	Instructions for Recording and Reporting Occupational Radiation Dose Data	2	Not Applicable	Applicable	This guidance governs site-specific, programmatic activities related to recording and reporting dose data that are the responsibility of the COL applicant.	12.1 12.5
8.8	Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable	3	Partially Conforms	Applicable	Implementation of this guidance is largely site-specific and is the responsibility of the COL applicant. However, the NuScale application for standard design certification considered this guidance to be applicable to the extent necessary to provide reasonable assurance that the COL applicant referencing the certified design can meet these requirements. The aspects of this guidance that are design-specific (i.e., pertain to design features, facilities, functions, and equipment that are technically relevant to the NuScale standard plant design - e.g., Position C.2) are Applicable to the DCA.	9.3 11.2 11.4 11.5 12.1 12.3 12.5
8.9	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program	1	Not Applicable	Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the COL applicant.	12.1 12.5
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable	1-R	Not Applicable	Applicable	The programmatic/operational aspects of this guidance are not applicable to the DCA. These site-specific aspects are the responsibility of the COL applicant referencing the certified design.	11.4 11.5 12.1 12.5
8.11	Applications of Bioassay for Uranium	-	Not Applicable	Not Applicable	This guidance governs programmatic activities that are applicable only to licensees for which uranium bioassay is required.	Not Applicable

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
8.13	Instruction Concerning Prenatal Radiation Exposure	3	Not Applicable	Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the COL applicant.	12.1 12.5
8.15	Acceptable Programs for Respiratory Protection	1	Not Applicable	Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the COL applicant.	12.1 12.5
8.18	Information Relevant to Ensuring that Radiation Exposures at Medical Institutions Will Be as Low as is Reasonably Achievable	2	Not Applicable	Not Applicable	This guidance governs activities applicable only to medical institutions.	Not Applicable
8.19	Occupational Radiation Dose Assessment in Light Water Reactor Power Plants - Design Stage Man-Rem Estimates	1	Conforms	Applicable	None.	12.4
8.20	Applications of Bioassay for Radioiodine	2	Not Applicable	Applicable	Revision 2 of RG 8.20 pending draft DG-8050 dated 9/2011. This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the COL applicant.	Not Applicable
8.21	Health Physics Surveys for Byproduct Material at NRC Licensed Processing and Manufacturing Plants	1	Not Applicable	Not Applicable	Applicable only to processing and manufacturing plants.	Not Applicable
8.22	Bioassay at Uranium Mills	1	Not Applicable	Not Applicable	Applicable only to uranium mills.	Not Applicable
8.23	Radiation Safety Surveys at Medical Institutions	1	Not Applicable	Not Applicable	This guidance governs activities applicable only to medical institutions.	Not Applicable
8.24	Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication	2	Not Applicable	Not Applicable	This guidance governs activities applicable only to facilities that process or fabricate fuel with uranium enriched with the U-235 isotope.	Not Applicable

**Table 1.9-2: Conformance with Regulatory Guides (Continued)**

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
8.25	Air Sampling in the Workplace	1	Not Applicable	Applicable	This guidance governs site-specific, programmatic activities related to air sampling in the workplace that are the responsibility of the COL applicant. However, the NuScale application for standard design certification will consider this guidance to the extent necessary to provide reasonable assurance that the COL applicant referencing the certified design can meet these requirements.	12.1 12.3
8.26	Applications of Bioassay for Fission and Activation Products	-	Not Applicable	Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the COL applicant.	Not Applicable
8.27	Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants	-	Not Applicable	Applicable	This guidance governs site-specific operational training programs, plans, and procedures that are the responsibility of the COL applicant.	12.1 12.5
8.28	Audible-Alarm Dosimeters	-	Not Applicable	Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the COL applicant.	12.1 12.5
8.29	Instruction Concerning Risks from Occupational Radiation Exposure	1	Not Applicable	Applicable	This guidance governs site-specific, programmatic training and instructional activities that are the responsibility of the COL applicant.	12.1 12.5
8.30	Health Physics Surveys in Uranium Recovery Facilities	1	Not Applicable	Not Applicable	This guidance governs activities applicable only to uranium recovery facilities.	Not Applicable
8.31	Information Relevant to Ensuring that Occupational Radiation Exposures at Uranium Recovery Facilities Will Be as Low as Is Reasonably Achievable	1	Not Applicable	Not Applicable	This guidance governs activities applicable only to uranium recovery facilities.	Not Applicable
8.32	Criteria for Establishing a Tritium Bioassay Program	-	Not Applicable	Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of a licensee authorized to possess nuclear material.	Not Applicable

Table 1.9-2: Conformance with Regulatory Guides (Continued)

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
8.34	Monitoring Criteria and Methods to Calculate Occupational Radiation Doses	-	Not Applicable	Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the COL applicant.	12.1 12.5
8.35	Planned Special Exposure	1	Not Applicable	Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the COL applicant.	12.1 12.5
8.36	Radiation Dose to the Embryo/Fetus	-	Not Applicable	Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the COL applicant.	12.1 12.5
8.37	ALARA Levels for Effluents from Materials Facilities	-	Not Applicable	Not Applicable	This guidance governs activities applicable only to materials facilities.	Not Applicable
8.38	Control of Access to High and Very High Radiation Areas in Nuclear Power Plants	1	Partially Conforms	Applicable	Implementation of this guidance is largely site-specific and is the responsibility of the COL applicant. However, the NuScale application for standard design certification considers this guidance to be Applicable to the extent necessary to provide reasonable assurance that the COL applicant referencing the certified design can meet these requirements.	12.1 12.3 12.5
8.39	Release of Patients Administered Radioactive Materials	-	Not Applicable	Not Applicable	This guidance governs activities applicable only to facilities that administer radio-pharmaceuticals.	Not Applicable
8.40	Methods for Measuring Effective Dose Equivalent from External Exposure	-	Not Applicable	Applicable	This guidance governs dosimetry methods for determining effective dose equivalent for external radiation exposures. These methods are the responsibility of the COL applicant.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 1.0, Rev 2: Introduction and Interfaces	II.1	No Specific Acceptance Criteria	-	-	No Specific Acceptance Criteria.	X
SRP 1.0, Rev 2: Introduction and Interfaces	II.2	SRP Acceptance Criteria Associated with Each Referenced SRP section	Conforms	Applicable	None.	Ch 1
SRP 1.0, Rev 2: Introduction and Interfaces	II.3	Performance of New Safety Features and Design Qualification Testing Requirements	Conforms	Applicable	None.	Ch 1
SRP 2.0, (March 2007): Site Characteristics and Site Parameters	II.1	Specific SRP Acceptance Criteria Contained in Related SRP Chapter 2 or Other Referenced SRP sections	Conforms	Applicable	This acceptance criterion is a pointer to other SRP sections.	2.0
SRP 2.0, (March 2007): Site Characteristics and Site Parameters	II.2	COL Application Referencing an Early Site Permit	Not Applicable	Applicable	This acceptance criterion is applicable only to COL applicants that do not reference the DCA.	2.0
SRP 2.0, (March 2007): Site Characteristics and Site Parameters	II.3	COL Application Referencing a Certified Design	Not Applicable	Applicable	This acceptance criterion is for COL applicants to meet the design parameters established in the Design Certification Application.	2.0
SRP 2.0, (March 2007): Site Characteristics and Site Parameters	II.4	COL Application Referencing an Early Site Permit and a Certified Design	Not Applicable	Applicable	This acceptance criterion is for COL applicants to meet the design parameters established in the Design Certification Application.	2.0
SRP 2.0, (March 2007): Site Characteristics and Site Parameters	II.5	COL Application Referencing Neither an Early Site Permit Nor a Certified Design	Not Applicable	Applicable	This acceptance criterion is applicable only to COL applicants that do not reference the DCA.	Not Applicable
SRP 2.0, (March 2007): Site Characteristics and Site Parameters	App A	Table 1: Examples of Site Characteristics and Site Parameters	Partially Conforms	Applicable	NuScale provides design Parameters where applicable.	Table 2.0-1
SRP 2.0, (March 2007): Site Characteristics and Site Parameters	App A	Table 2: Examples of Site-Related Design Parameters and Design Characteristics	Partially Conforms	Applicable	NuScale provides design Parameters where applicable.	Table 2.0-1
SRP 2.1.1, Rev 3: Site Location and Description	All	Specification of Location and Site Area Map	Not Applicable	Applicable	Site-specific.	2.1

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 2.1.2, Rev 3: Exclusion Area Authority and Control	All	Establishment of Authority, Exclusion or Removal of Personnel and Property, and Proposed and Permitted Activities	Not Applicable	Applicable	Site-specific.	2.1
SRP 2.1.3, Rev 3: Population Distribution	All	Population Data, Exclusion Area, Low-Population Zone, Nearest Population Center Boundary, and Population Density	Not Applicable	Applicable	Site-specific.	2.1
SRP 2.2.1-2.2.2, Rev 3: Identification of Potential Hazards in Site Vicinity	All	Various	Not Applicable	Applicable	Site-specific.	2.2
SRP 2.2.3, Rev 3: Evaluation of Potential Accidents	All	Event Probability and Design-Basis Event Analysis	Not Applicable	Applicable	Site-specific.	2.2
SRP 2.3.1, Rev 3: Regional Climatology	All	Various	Not Applicable	Applicable	Site-specific	Not Applicable
SRP 2.3.1, Rev 3: Regional Climatology	III.4.b.1	Postulated Site Parameters	Conforms	Applicable	None.	Table 2.0-1 2.3.1
SRP 2.3.1, Rev 3: Regional Climatology	III.4.b.2	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	Table 2.0-1 2.3.1
SRP 2.3.1, Rev 3: Regional Climatology	III.4.b.3	Site Parameters Summary Table	Conforms	Applicable	None.	Table 2.0-1 2.3.1
SRP 2.3.1, Rev 3: Regional Climatology	III.4.b.4	Basis for Site Parameters	Conforms	Applicable	None.	Table 2.0-1 2.3.1
SRP 2.3.2, Rev 3: Local Meteorology	II.1 thru II.4	Various	Not Applicable	Applicable	Site-specific.	2.3
SRP 2.3.2, Rev 3: Local Meteorology	III.4.b.i	Postulated Site Parameters	Conforms	Applicable	None.	Table 2.0-1 2.3.2
SRP 2.3.2, Rev 3: Local Meteorology	III.4.b.ii	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	Table 2.0-1 2.3.2
SRP 2.3.2, Rev 3: Local Meteorology	III.4.b.iii	Site Parameters Summary Table	Conforms	Applicable	None.	Table 2.0-1 2.3.2
SRP 2.3.2, Rev 3: Local Meteorology	III.4.b.iv	Basis for Site Parameters	Conforms	Applicable	None.	Table 2.0-1 2.3.2



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 2.3.3, Rev 3: Onsite Meteorological Measurements Program	All	Various	Not Applicable	Applicable	Site-specific.	2.3
SRP 2.3.4, Rev 3: Short-Term Atmospheric Dispersion Estimates for Accident Releases	All	Various	Not Applicable	Applicable	Site-specific.	2.3
SRP 2.3.4, Rev 3: Short-Term Atmospheric Dispersion Estimates for Accident Releases	III.6.b.1	Postulated Site Parameters	Conforms	Applicable	None.	2.3.4
SRP 2.3.4, Rev 3: Short-Term Atmospheric Dispersion Estimates for Accident Releases	III.6.b.2	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.3.4
SRP 2.3.4, Rev 3: Short-Term Atmospheric Dispersion Estimates for Accident Releases	III.6.b.3	Site Parameters Summary Table	Conforms	Applicable	None.	2.3.4
SRP 2.3.4, Rev 3: Short-Term Atmospheric Dispersion Estimates for Accident Releases	III.6.b.4	Basis for Site Parameters	Conforms	Applicable	None.	2.3.4
SRP 2.3.4, Rev 3: Short-Term Atmospheric Dispersion Estimates for Accident Releases	III (no number)	Applicable Short-Term (Post-Accident) Site Parameters - EAB, LPZ, and Control Room Atmospheric Dispersion Factors	Conforms	Applicable	None.	2.3.4
SRP 2.3.5, Rev 3: Long-Term Atmospheric Dispersion Estimates for Routine Releases	All	Various	Not Applicable	Applicable	Site-specific.	2.3
SRP 2.3.5, Rev 3: Long-Term Atmospheric Dispersion Estimates for Routine Releases	III.5.b.1	Postulated Site Parameters	Conforms	Applicable	None.	2.3.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 2.3.5, Rev 3: Long-Term Atmospheric Dispersion Estimates for Routine Releases	III.5.b.2	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.3.5
SRP 2.3.5, Rev 3: Long-Term Atmospheric Dispersion Estimates for Routine Releases	III.5.b.3	Site Parameters Summary Table	Conforms	Applicable	None.	2.3.5
SRP 2.3.5, Rev 3: Long-Term Atmospheric Dispersion Estimates for Routine Releases	III.5.b.4	Basis for Site Parameters	Conforms	Applicable	None.	2.3.5
SRP 2.3.5, Rev 3: Long-Term Atmospheric Dispersion Estimates for Routine Releases	III (no number)	Applicable Long-Term (Routine Release) Site Parameters - Maximum Annual Average Site Boundary Atmospheric Dispersion Factor	Conforms	Applicable	None.	2.3.5
SRP 2.4.1, Rev 3: Hydrologic Description	All	Various	Not Applicable	Applicable	Site-specific.	2.4
SRP 2.4.1, Rev 3: Hydrologic Description	III.7.B.i	Postulated Site Parameters	Conforms	Applicable	None.	2.4.1
SRP 2.4.1, Rev 3: Hydrologic Description	III.7.B.ii	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.4.1
SRP 2.4.1, Rev 3: Hydrologic Description	III.7.B.iii	Site Parameters Summary Table	Conforms	Applicable	None.	2.4.1
SRP 2.4.1, Rev 3: Hydrologic Description	III.7.B.iv	Basis for Site Parameters	Conforms	Applicable	None.	2.4.1
SRP 2.4.2, Rev 4: Floods	All	Various	Not Applicable	Applicable	Site-specific.	2.4
SRP 2.4.2, Rev 4: Floods	III.11.B.i	Postulated Site Parameters	Conforms	Applicable	None.	2.4.2
SRP 2.4.2, Rev 4: Floods	III.11.B.ii	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.4.2
SRP 2.4.2, Rev 4: Floods	III.11.B.iii	Site Parameters Summary Table	Conforms	Applicable	None.	2.4.2
SRP 2.4.2, Rev 4: Floods	III.11.B.iv	Basis for Site Parameters	Conforms	Applicable	None.	2.4.2
SRP 2.4.3, Rev 4: Probable Maximum Flood (PMF) on Streams and Rivers	All	Various	Not Applicable	Applicable	Site-specific.	2.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 2.4.3, Rev 4: Probable Maximum Flood (PMF) on Streams and Rivers	III.4.B.i	Postulated Site Parameters	Conforms	Applicable	None.	2.4.3
SRP 2.4.3, Rev 4: Probable Maximum Flood (PMF) on Streams and Rivers	III.4.B.ii	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.4.3
SRP 2.4.3, Rev 4: Probable Maximum Flood (PMF) on Streams and Rivers	III.4.B.iii	Site Parameters Summary Table	Conforms	Applicable	None.	2.4.3
SRP 2.4.3, Rev 4: Probable Maximum Flood (PMF) on Streams and Rivers	III.4.B.iv	Basis for Site Parameters	Conforms	Applicable	None.	2.4.3
SRP 2.4.4, Rev 3: Potential Dam Failures	All	Various	Not Applicable	Applicable	Site-specific.	2.4
SRP 2.4.4, Rev 3: Potential Dam Failures	III.8.B.i	Postulated Site Parameters	Conforms	Applicable	None.	2.4.4
SRP 2.4.4, Rev 3: Potential Dam Failures	III.8.B.ii	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.4.4
SRP 2.4.4, Rev 3: Potential Dam Failures	III.8.B.iii	Site Parameters Summary Table	Conforms	Applicable	None.	2.4.4
SRP 2.4.4, Rev 3: Potential Dam Failures	III.8.B.iv	Basis for Site Parameters	Conforms	Applicable	None.	2.4.4
SRP 2.4.5, Rev 3: Probable Maximum Surge and Seiche Flooding	All	Various	Not Applicable	Applicable	Site-specific.	2.4
SRP 2.4.5, Rev 3: Probable Maximum Surge and Seiche Flooding	III.7.B.i	Postulated Site Parameters	Conforms	Applicable	None.	2.4.5
SRP 2.4.5, Rev 3: Probable Maximum Surge and Seiche Flooding	III.7.B.ii	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.4.5
SRP 2.4.5, Rev 3: Probable Maximum Surge and Seiche Flooding	III.7.B.iii	Site Parameters Summary Table	Conforms	Applicable	None.	2.4.5
SRP 2.4.5, Rev 3: Probable Maximum Surge and Seiche Flooding	III.7.B.iv	Basis for Site Parameters	Conforms	Applicable	None.	2.4.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 2.4.6, Rev 3: Probable Maximum Tsunami Hazards	All	Various	Not Applicable	Applicable	Site-specific.	2.4
SRP 2.4.6, Rev 3: Probable Maximum Tsunami Hazards	III.9.B.i	Postulated Site Parameters	Conforms	Applicable	None.	2.4.6
SRP 2.4.6, Rev 3: Probable Maximum Tsunami Hazards	III.9.B.ii	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.4.6
SRP 2.4.6, Rev 3: Probable Maximum Tsunami Hazards	III.9.B.iii	Site Parameters Summary Table	Conforms	Applicable	None.	2.4.6
SRP 2.4.6, Rev 3: Probable Maximum Tsunami Hazards	III.9.B.iv	Basis for Site Parameters	Conforms	Applicable	None.	2.4.6
SRP 2.4.7, Rev 3: Ice Effects	All	Various	Not Applicable	Applicable	Site-specific.	2.4
SRP 2.4.7, Rev 3: Ice Effects	III.6.B.i	Postulated Site Parameters	Conforms	Applicable	None.	2.4.7
SRP 2.4.7, Rev 3: Ice Effects	III.6.B.ii	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.4.7
SRP 2.4.7, Rev 3: Ice Effects	III.6.B.iii	Site Parameters Summary Table	Conforms	Applicable	None.	2.4.7
SRP 2.4.7, Rev 3: Ice Effects	III.6.B.iv	Basis for Site Parameters	Conforms	Applicable	None.	2.4.7
SRP 2.4.8, Rev 3: Cooling Water Canals and Reservoirs	All	Various	Not Applicable	Applicable	Site-specific.	2.4
SRP 2.4.8, Rev 3: Cooling Water Canals and Reservoirs	III.5.B.1	Postulated Site Parameters	Conforms	Applicable	None.	2.4.8
SRP 2.4.8, Rev 3: Cooling Water Canals and Reservoirs	III.5.B.2	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.4.8
SRP 2.4.8, Rev 3: Cooling Water Canals and Reservoirs	III.5.B.3	Site Parameters Summary Table	Conforms	Applicable	None.	2.4.8
SRP 2.4.8, Rev 3: Cooling Water Canals and Reservoirs	III.5.B.4	Basis for Site Parameters	Conforms	Applicable	None.	2.4.8

Tier 2

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Revision 0

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 2.4.9, Rev 3: Channel Diversions	All	Various	Not Applicable	Applicable	Site-specific.	2.4
SRP 2.4.9, Rev 3: Channel Diversions	III.8.B.i	Postulated Site Parameters	Conforms	Applicable	None.	2.4.9
SRP 2.4.9, Rev 3: Channel Diversions	III.8.B.ii	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.4.9
SRP 2.4.9, Rev 3: Channel Diversions	III.8.B.iii	Site Parameters Summary Table	Conforms	Applicable	None.	2.4.9
SRP 2.4.9, Rev 3: Channel Diversions	III.8.B.iv	Basis for Site Parameters	Conforms	Applicable	None.	2.4.9
SRP 2.4.10, Rev 3: Flooding Protection Requirements	All	Various	Not Applicable	Applicable	Site-specific.	2.4
SRP 2.4.10, Rev 3: Flooding Protection Requirements	III.5.B.i	Postulated Site Parameters	Conforms	Applicable	None.	2.4.10
SRP 2.4.10, Rev 3: Flooding Protection Requirements	III.5.B.ii	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.4.10
SRP 2.4.10, Rev 3: Flooding Protection Requirements	III.5.B.iii	Site Parameters Summary Table	Conforms	Applicable	None.	2.4.10
SRP 2.4.10, Rev 3: Flooding Protection Requirements	III.5.B.iv	Basis for Site Parameters	Conforms	Applicable	None.	2.4.10
SRP 2.4.11, Rev 3: Low Water Considerations	All	Various	Not Applicable	Applicable	Site-specific.	2.4
SRP 2.4.11, Rev 3: Low Water Considerations	III.6.B.i	Postulated Site Parameters	Conforms	Applicable	None.	2.4.11
SRP 2.4.11, Rev 3: Low Water Considerations	III.6.B.ii	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.4.11
SRP 2.4.11, Rev 3: Low Water Considerations	III.6.B.iii	Site Parameters Summary Table	Conforms	Applicable	None.	2.4.11
SRP 2.4.11, Rev 3: Low Water Considerations	III.6.B.iv	Basis for Site Parameters	Conforms	Applicable	None.	2.4.11
SRP 2.4.12, Rev 3: Groundwater	II.1 thru II.5	Various	Not Applicable	Applicable	Site-specific.	2.4
SRP 2.4.12, Rev 3: Groundwater	III.6.B.i	Postulated Site Parameters	Conforms	Applicable	None.	2.4.12
SRP 2.4.12, Rev 3: Groundwater	III.6.B.ii	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.4.12

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 2.4.12, Rev 3: Groundwater	III.6.B.iii	Site Parameters Summary Table	Conforms	Applicable	None.	2.4.12
SRP 2.4.12, Rev 3: Groundwater	III.6.B.iv	Basis for Site Parameters	Conforms	Applicable	None.	2.4.12
SRP 2.4.13, Rev 3: Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	All	Various	Not Applicable	Applicable	Site-specific.	2.4
SRP 2.4.13, Rev 3: Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	III.5.B.i	Postulated Site Parameters	Conforms	Applicable	None.	2.4.13
SRP 2.4.13, Rev 3: Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	III.5.B.ii	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.4.13
SRP 2.4.13, Rev 3: Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	III.5.B.iii	Site Parameters Summary Table	Conforms	Applicable	None.	2.4.13
SRP 2.4.13, Rev 3: Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	III.5.B.iv	Basis for Site Parameters	Conforms	Applicable	None.	2.4.13
SRP 2.4.14, Rev 3: Technical Specifications and Emergency Operation Requirements	All	Various	Not Applicable	Applicable	Site-specific.	2.4
SRP 2.4.14, Rev 3: Technical Specifications and Emergency Operation Requirements	III.5.B.i	Postulated Site Parameters	Conforms	Applicable	None.	2.4.14

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 2.4.14, Rev 3: Technical Specifications and Emergency Operation Requirements	III.5.B.ii	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.4.14
SRP 2.4.14, Rev 3: Technical Specifications and Emergency Operation Requirements	III.5.B.iii	Site Parameters Summary Table	Conforms	Applicable	None.	2.4.14
SRP 2.4.14, Rev 3: Technical Specifications and Emergency Operation Requirements	III.5.B.iv	Basis for Site Parameters	Conforms	Applicable	None.	2.4.14
SRP 2.5.1, Rev 4: Basic Geologic and Seismic Information	All	Regional and Site Geology	Not Applicable	Applicable	Site-specific.	2.5
SRP 2.5.2, Rev 4: Vibratory Ground Motion	All	Various	Not Applicable	Applicable	Site-specific.	2.5
SRP 2.5.2, Rev 4: Vibratory Ground Motion	III.2.a	Postulated Site Parameters	Conforms	Applicable	None.	2.5.2
SRP 2.5.2, Rev 4: Vibratory Ground Motion	III.2.b	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.5.2
SRP 2.5.2, Rev 4: Vibratory Ground Motion	III.2.c	Site Parameters Summary Table	Conforms	Applicable	None.	2.5.2
SRP 2.5.2, Rev 4: Vibratory Ground Motion	III.2.d	Basis for Site Parameters	Conforms	Applicable	None.	2.5.2
SRP 2.5.3, Rev 4: Surface Faulting	All	Various	Not Applicable	Applicable	Site-specific.	2.5
SRP 2.5.3, Rev 4: Surface Faulting	III.2.a	Postulated Site Parameters	Conforms	Applicable	None.	2.5.3
SRP 2.5.3, Rev 4: Surface Faulting	III.2.b	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.5.3
SRP 2.5.3, Rev 4: Surface Faulting	III.2.c	Site Parameters Summary Table	Conforms	Applicable	None.	2.5.3
SRP 2.5.3, Rev 4: Surface Faulting	III.2.d	Basis for Site Parameters	Conforms	Applicable	None.	2.5.3

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 2.5.4, Rev 4: Stability of Subsurface Materials and Foundations	All	Various	Not Applicable	Applicable	Site-specific.	2.5
SRP 2.5.4, Rev 4: Stability of Subsurface Materials and Foundations	III.2.A	Postulated Site Parameters	Conforms	Applicable	None.	2.5.4
SRP 2.5.4, Rev 4: Stability of Subsurface Materials and Foundations	III.2.B	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.5.4
SRP 2.5.4, Rev 4: Stability of Subsurface Materials and Foundations	III.2.C	Site Parameters Summary Table	Conforms	Applicable	None.	2.5.4
SRP 2.5.4, Rev 4: Stability of Subsurface Materials and Foundations	III.2.D	Basis for Site Parameters	Conforms	Applicable	None.	2.5.4
SRP 2.5.5, Rev 4: Stability of Slopes	All	Various	Not Applicable	Applicable	Site-specific.	2.5
SRP 2.5.5, Rev 4: Stability of Slopes	III.2.A	Postulated Site Parameters	Conforms	Applicable	None.	2.5.5
SRP 2.5.5, Rev 4: Stability of Slopes	III.2.B	Site Parameters Included as Tier 1 Information	Conforms	Applicable	None.	2.5.5
SRP 2.5.5, Rev 4: Stability of Slopes	III.2.C	Site Parameters Summary Table	Conforms	Applicable	None.	2.5.5
SRP 2.5.5, Rev 4: Stability of Slopes	III.2.D	Basis for Site Parameters	Conforms	Applicable	None.	2.5.5
SRP 3.2.1, Rev 2: Seismic Classification	II.1	Seismic Design Classification to Meet GDC 2; 10 CFR 100, Appendix A; and 10 CFR 50, Appendix S	Conforms	Applicable	None.	3.2.1
SRP 3.2.2, Rev 2: System Quality Group Classification	II.1	Quality Group Classification to Meet GDC 1 and 10 CFR 50.55a	Conforms	Applicable	None.	3.2.2



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 3.2.2, Rev 2: System Quality Group Classification	Table 3.2.21	Summary of Construction Codes and Standards for Components of WaterCooled Nuclear Power Plants by NRC Quality Classification System (Page 3.2.2-12)	Partially Conforms	Applicable	This acceptance criterion is applicable except for reference to RG 1.85, which was withdrawn in 2004 because its guidance was updated and incorporated into RG 1.84.	Table 3.2-1
SRP 3.2.2, Rev 2: System Quality Group Classification	App. A and Table A-1	Additional Guidance for Classification of Systems and Components and Application of Quality Standards	Partially Conforms	Applicable	The intent of Table A-1 is applicable but some of the specific language refers to SSCs not part of the NuScale design. For example, the NuScale design does not include combustible gas control systems, emergency diesel generators, ESF rooms, or pressurizer power operated relief valves.	Table 3.2-1
SRP 3.3.1, Rev. 3: Wind Loadings	II.1	Most Severe Wind	Partially Conforms	Applicable	Bounding parameters are established.	3.3.1
SRP 3.3.1, Rev. 3: Wind Loadings	II.2	Design Wind Speed, Recurrence Interval, and Other Site-Related Wind Parameters	Conforms	Applicable	None.	3.3.1
SRP 3.3.1, Rev. 3: Wind Loadings	II.3	Procedures for Transforming Wind Speed Into Equivalent Pressure	Conforms	Applicable	None.	3.3.1
SRP 3.3.2: Rev. 3: Tornado Loads	II.1	Most Severe Tornado Wind and Associated Missiles	Partially Conforms	Applicable	Bounding parameters are established.	3.3.2
SRP 3.3.2: Rev. 3: Tornado Loads	II.2	Acceptance Criteria for Tornado Parameters and Spectrum of Tornado-Generated Missiles	Conforms	Applicable	None.	3.3.2
SRP 3.3.2: Rev. 3: Tornado Loads	II.3	Procedures for Transforming Tornado Parameters Into Equivalent Loads on Structures	Conforms	Applicable	None.	3.3.2
SRP 3.3.2: Rev. 3: Tornado Loads	II.4	Demonstrating That Failure of Structure or Component Not Designed for Tornado Loads Will Not Affect the Capability of Other SSCs to Perform Safety Functions	Conforms	Applicable	None.	3.3.2
SRP 3.4.1, Rev. 3: Internal Flood Protection for Onsite Equipment Failures	II.1	Seismic Design and Classification Requirements	Conforms	Applicable	None.	3.4.1

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 3.4.1, Rev. 3: Internal Flood Protection for Onsite Equipment Failures	II.2	Compliance with GDC 4	Conforms	Applicable	None.	3.4.1
SRP 3.4.2, Rev. 3: Protection of Structures Against Flood from External Sources	II.1	Most Severe Highest Flood and Groundwater Levels	Partially Conforms	Applicable	The NuScale Certified design assumes the NPP is above the maximum flood level.	3.4.2
SRP 3.4.2, Rev. 3: Protection of Structures Against Flood from External Sources	II.2	Highest Flood Level Below Grade - Consideration of Hydrostatic Effects and Wave Action	Conforms	Applicable	The NuScale Certified design assumes the NPP is above the maximum flood level.	3.4.2
SRP 3.4.2, Rev. 3: Protection of Structures Against Flood from External Sources	II.3	Highest Flood Level Above Grade - Consideration of Dynamic Loads From Wave Action	Conforms	Applicable	The NuScale Certified design assumes the NPP is above the maximum flood level.	3.4.2
SRP 3.5.1.1, Rev. 3: Internally-Generated Missiles (Outside Containment)	II.1	Statistical Significance of an Identified Missile by Probability Analysis	Conforms	Applicable	None.	3.5.1
SRP 3.5.1.1, Rev. 3: Internally-Generated Missiles (Outside Containment)	II.2	Acceptable Methods of Providing Missile Protection	Conforms	Applicable	None.	3.5.1
SRP 3.5.1.2, Rev. 3: Internally Generated Missiles (Inside Containment)	II.1	Statistical Significance of an Identified Missile by Probability Analysis	Conforms	Applicable	None.	3.5.1
SRP 3.5.1.2, Rev. 3: Internally Generated Missiles (Inside Containment)	II.2	Acceptable Methods of Providing Missile Protection	Conforms	Applicable	None.	3.5.1
NSDSRS 3.5.1.3, Rev. 0: Turbine Missiles	II.1	Probability of Unacceptable Damage From Turbine Missiles	Not Applicable	Applicable	COL Applicant to verify that TG missile generation is less than 1.0E-05.	3.5.1
NSDSRS 3.5.1.3, Rev. 0: Turbine Missiles	II.2	Turbine Missile Generation	Not Applicable	Not Applicable	The NuScale certified design assumes no turbine missile is generated.	Not Applicable
NSDSRS 3.5.1.3, Rev. 0: Turbine Missiles	II.3	Acceptably Low Missile Generation Probability	Not Applicable	Not Applicable	The NuScale certified design assumes no turbine missile is generated.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 3.5.1.3, Rev. 0: Turbine Missiles	II.4	Missile Generation Probability Tables From Turbine Manufacturers (Including Table 3.5.1.3-1)	Not Applicable	Not Applicable	The NuScale certified design assumes no turbine missile is generated.	Not Applicable
NSDSRS 3.5.1.3, Rev. 0: Turbine Missiles	II.5	Inservice Inspection and Test Program for Applicants Obtaining Turbine From Manufacturers without NRC-Approved Procedures for Calculating Missile Generation Probabilities	Not Applicable	Applicable	COL Applicant to verify that TG missile generation is less than 1.0E-05.	Not Applicable
NSDSRS 3.5.1.3, Rev. 0: Turbine Missiles	II.6	Protective Barriers	Not Applicable	Applicable	COL Applicant to verify that TG missile generation is less than 1.0E-05.	3.5.1
SRP 3.5.1.4, Rev. 4: Missiles Generated by Extreme Winds	II.1	Design Basis Tornado-Generated Missile Spectrum	Conforms	Applicable	The NuScale Certified design also includes RG 1.221 for Design Basis Hurricane-Generated Missiles.	3.5.1.4
SRP 3.5.1.4, Rev. 4: Missiles Generated by Extreme Winds	II.2	Statistical Significance of an Identified Missile by Probability	Conforms	Applicable	None.	3.5.1.4
SRP 3.5.1.4, Rev. 4: Missiles Generated by Extreme Winds	II.3	Identifying Appropriate Design Basis Missiles Generated by Natural Phenomena	Conforms	Applicable	None.	3.5.1.4
SRP 3.5.1.5, Rev 4: Site Proximity Missiles (Except Aircraft)	II.1	Compliance with 10 CFR 100	Not Applicable	Applicable	The NuScale certified design assumes no proximity missiles.	Not Applicable
SRP 3.5.1.5, Rev 4: Site Proximity Missiles (Except Aircraft)	II.2	Compliance with GDC 4	Not Applicable	Applicable	The NuScale certified design assumes no proximity missiles.	Not Applicable
SRP 3.5.1.6, Rev 4: Aircraft Hazards	II.1 and II.2	Various	Not Applicable	Applicable	The NuScale certified design assumes no aircraft hazard missiles.	3.5.1.6
SRP 3.5.1.6, Rev 4: Aircraft Hazards	III.8.B.1	Postulated Site Parameters	Conforms	Applicable	The NuScale certified design assumes no aircraft hazard missiles.	Table 2.0-1
SRP 3.5.1.6, Rev 4: Aircraft Hazards	III.8.B.2	Site Parameters Included as Tier 1 Information	Conforms	Applicable	The NuScale certified design assumes no aircraft hazard missiles.	Table 2.0-1
SRP 3.5.1.6, Rev 4: Aircraft Hazards	III.8.B.3	Site Parameters Summary Table	Conforms	Applicable	The NuScale certified design assumes no aircraft hazard missiles.	Table 2.0-1
SRP 3.5.1.6, Rev 4: Aircraft Hazards	III.8.B.4	Basis for Site Parameters	Conforms	Applicable	The NuScale certified design assumes no aircraft hazard missiles.	Table 2.0-1

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1.9-67

Revision 0

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 3.5.2, Rev 3: Structures, Systems, and Components to be Protected From Externally- Generated Missiles	II (no number)	Capability of SSCs to Withstand the Effects of Externally Generated Missiles	Conforms	Applicable	None.	3.5.2
SRP 3.5.3, Rev. 3: Barrier Design Procedures	II.1.A	For Local Damage Prediction - Concrete	Conforms	Applicable	None.	3.5.3
SRP 3.5.3, Rev. 3: Barrier Design Procedures	II.1.B	For Local Damage Prediction - Steel	Conforms	Applicable	None.	3.5.3
SRP 3.5.3, Rev. 3: Barrier Design Procedures	II.1.C	For Local Damage Prediction - Composite sections	Not Applicable	Not Applicable	This acceptance criterion specifies provisions when using composite or multi-element barriers. NuScale does not intend to use composite or multi- element barriers.	3.5.3
SRP 3.5.3, Rev. 3: Barrier Design Procedures	II.2	For Overall Damage Prediction	Partially Conforms	Applicable	This acceptance criterion is applicable except for reference to subtier ANSI/ AISC N690-1994 with Supplement 2 (2004). NuScale intends to use the 2006 version of this standard.	3.5.3
SRP 3.6.1, Rev 3: Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	II.1	Separation of High and Moderate Energy Fluid Systems From Essential Systems/Components	Conforms	Applicable	None.	3.6.1
SRP 3.6.1, Rev 3: Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	II.2	High and Moderate Energy Fluid Systems Are Enclosed	Conforms	Applicable	None.	3.6.1
SRP 3.6.1, Rev 3: Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	II.3	Cases Where Neither Physical Separation Nor Protective Enclosures Are Practical	Conforms	Applicable	None.	3.6.1

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 3.6.1, Rev 3: Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	II.4	Design Features	Conforms	Applicable	None.	3.6.1
SRP 3.6.1, Rev 3: Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	II.5	Effects of Postulated Failures	Conforms	Applicable	None.	3.6.1
SRP 3.6.2, Rev 2: Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	II.1	Postulated Pipe Rupture Locations Inside Containment	Conforms	Applicable	None.	3.6.2
SRP 3.6.2, Rev 2: Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	II.2	Postulated Pipe Rupture Locations Outside Containment	Conforms	Applicable	None.	3.6.2
SRP 3.6.2, Rev 2: Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	II.3	Methods of Analysis	Conforms	Applicable	None.	3.6.2
SRP 3.6.2, Rev 2: Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	III.1	Pipe Break Criteria	Conforms	Applicable	None.	3.6.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 3.6.2, Rev 2: Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	III.2	Dynamic Effects	Conforms	Applicable	None.	3.6.2
SRP 3.6.2, Rev 2: Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	III.3	Assumptions for Modeling Jet Impingement Forces	Partially Conforms	Applicable	As discussed in Section 3.6.2, jets are excluded by the use of an Integrated shield/restraint device.	3.6.2
SRP 3.6.2, Rev 2: Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	III.4	Analyses of Pipe Break Dynamic Effects on Mechanical Components and Supports	Conforms	Applicable	As discussed in Section 3.6.2, jets are excluded by the use of an Integrated shield/restraint device.	3.6.2
SRP 3.6.3, Rev 1: Leak- Before-Break Evaluation Procedures	II.1	Compliance with GDC 4	Conforms	Applicable	None.	3.6.3
SRP 3.6.3, Rev 1: Leak- Before-Break Evaluation Procedures	II.2	Low Probability of Pipe Rupture	Conforms	Applicable	None.	3.6.3
NSDSRS 3.7.1, Rev. 0: Seismic Design Parameters	II.1	Design Ground Motion	Conforms	Applicable	None.	3.7.1
NSDSRS 3.7.1, Rev. 0: Seismic Design Parameters	II.2	Percentage of Critical Damping Values	Conforms	Applicable	None.	3.7.1
NSDSRS 3.7.1, Rev. 0: Seismic Design Parameters	II.3	Supporting Media for Seismic Category I Structures	Conforms	Applicable	None.	3.7.1
NSDSRS 3.7.1, Rev. 0: Seismic Design Parameters	II.4	Review Considerations for DC and COL Applications	Conforms	Applicable	None.	3.7.1
NSDSRS 3.7.2, Rev 0: Seismic System Analysis	II.1	Seismic Analysis Methods	Conforms	Applicable	None.	3.7.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 3.7.2, Rev 0: Seismic System Analysis	II.2	Natural Frequencies and Responses	Conforms	Applicable	None.	3.7.2
NSDSRS 3.7.2, Rev 0: Seismic System Analysis	II.3	Procedures Used for Analytical Modeling	Conforms	Applicable	None.	3.7.2
NSDSRS 3.7.2, Rev 0: Seismic System Analysis	II.4	Soil-Structure Interaction	Conforms	Applicable	None.	3.7.2
NSDSRS 3.7.2, Rev 0: Seismic System Analysis	II.5	Development of In-Structure Response Spectra	Conforms	Applicable	None.	3.7.2
NSDSRS 3.7.2, Rev 0: Seismic System Analysis	II.6	Three Components of Design Ground Motion	Conforms	Applicable	None.	3.7.2
NSDSRS 3.7.2, Rev 0: Seismic System Analysis	II.7	Combination of Modal Responses	Conforms	Applicable	None.	3.7.2
NSDSRS 3.7.2, Rev 0: Seismic System Analysis	II.8	Interaction of Non-Seismic Category I Structures with Seismic Category I SSCs	Conforms	Applicable	None.	3.7.2
NSDSRS 3.7.2, Rev 0: Seismic System Analysis	II.9	Effects of Parameter Variations on Floor Response Spectra	Conforms	Applicable	None.	3.7.2
NSDSRS 3.7.2, Rev 0: Seismic System Analysis	II.10	Use of Equivalent Vertical Static Factors	Conforms	Applicable	None.	3.7.2
NSDSRS 3.7.2, Rev 0: Seismic System Analysis	II.11	Methods Used to Account for Torsional Effects	Conforms	Applicable	None.	3.7.2
NSDSRS 3.7.2, Rev 0: Seismic System Analysis	II.12	Comparison of Responses	Not Applicable	Applicable	NuScale will not be performing both time history analysis and response spectrum analysis in its analysis of structures.	3.7.2
NSDSRS 3.7.2, Rev 0: Seismic System Analysis	II.13	Analysis Procedure for Damping	Conforms	Applicable	None.	3.7.1 3.7.2
NSDSRS 3.7.2, Rev 0: Seismic System Analysis	II.14	Determination of Overturning Moments and Sliding Forces, Structure to Soil Pressures and Frictional Forces for Seismic Category I Structures	Conforms	Applicable	None.	3.7.2
NSDSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.1	Seismic Analysis Methods	Conforms	Applicable	None.	3.7.3

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.2	Determination of Number of Earthquake Cycles	Conforms	Applicable	None.	3.7.3
NSDSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.3	Procedures Used for Analytical Modeling	Conforms	Applicable	None.	3.7.3
NSDSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.4	Basis for Selection of Frequencies	Conforms	Applicable	None.	3.7.3
NSDSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.5	Analysis Procedure for Damping	Conforms	Applicable	None.	3.7.3
NSDSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.6	Three Components of Design Ground Motion	Conforms	Applicable	None.	3.7.3
NSDSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.7	Combination of Modal Responses	Conforms	Applicable	None.	3.7.3
NSDSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.8	Interaction of Non-Seismic Category I Subsystems with Seismic Category I SSCs	Conforms	Applicable	None.	3.7.3
NSDSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.9	Multiply-Supported Equipment and Components with Distinct Inputs	Conforms	Applicable	None.	3.7.3
NSDSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.10	Use of Equivalent Vertical Static Factors	Conforms	Applicable	None.	3.7.3
NSDSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.11	Torsional Effects of Eccentric Masses	Conforms	Applicable	None.	3.7.3
NSDSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.12	Seismic Category I Buried Piping, Conduits, and Tunnels	Conforms	Applicable	None.	3.7.3
NSDSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.13	Methods for Seismic Analysis of Seismic Category I Concrete Dams	Not Applicable	Applicable	The NuScale standard plant design does not involve the use of dams, whether concrete, earthen, or other.	3.7.3



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.14	Methods for Seismic Analysis of Above-Ground Tanks	Conforms	Applicable	None.	3.7.3
SRP 3.7.4, Rev 2: Seismic Instrumentation	II.1	Comparison with RG 1.12	Partially Conforms	Applicable	There is a COL item to comply. Locations are identified in conformance with RG 1.12, however seismic instrumentation cannot be placed inside containment.	3.7.4
SRP 3.7.4, Rev 2: Seismic Instrumentation	II.2	Comparison with RG 1.166	Not Applicable	Applicable	See RG 1.166 in Table 1.9-2.	3.7.4
SRP 3.7.4, Rev 2: Seismic Instrumentation	II.3	Comparison with the requirements of 10 CFR 20.1101 (ALARA)	Not Applicable	Applicable	Identified as an expectation for COL applicants.	Not Applicable
SRP 3.8.1, Rev 4: Concrete Containment	All	Various	Not Applicable	Not Applicable	NuScale does not have a concrete containment.	Not Applicable
NSDSRS 3.8.2, Rev. 0: Steel Containment	II.1	Description of the Containment	Conforms	Applicable	None.	3.8.2
NSDSRS 3.8.2, Rev. 0: Steel Containment	II.2	Applicable Codes, Standards, and Specifications	Conforms	Applicable	None.	3.8.2
NSDSRS 3.8.2, Rev. 0: Steel Containment	II.3	Loads and Loading Combinations	Conforms	Applicable	None.	3.8.2
NSDSRS 3.8.2, Rev. 0: Steel Containment	II.4	Design and Analysis Procedures	Conforms	Applicable	None.	3.8.2
NSDSRS 3.8.2, Rev. 0: Steel Containment	II.5	Structural Acceptance Criteria	Conforms	Applicable	None.	3.8.2
NSDSRS 3.8.2, Rev. 0: Steel Containment	II.6	Materials, Quality Control, and Special Construction Techniques	Conforms	Applicable	None.	3.8.2
NSDSRS 3.8.2, Rev. 0: Steel Containment	II.7	Testing and Inservice Surveillance Requirements	Conforms	Applicable	None.	3.8.2
SRP 3.8.3, Rev 4: Concrete and Steel Internal Structures of Steel or Concrete Containments	All	Various	Not Applicable	Not Applicable	The NuScale Containment does not have internal structures.	Not Applicable
NSDSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.1	Description of the Structures	Conforms	Applicable	None.	3.8.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSDSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.2	Applicable Codes, Standards, and Specifications	Conforms	Applicable	None.	3.8.4
NSDSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.3	Loads and Load Combinations	Conforms	Applicable	None.	3.8.4
NSDSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.3.A	Concrete Structures	Conforms	Applicable	None.	3.8.4
NSDSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.3.B	Steel Structures	Conforms	Applicable	None.	3.8.4
NSDSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.4	Design and Analysis Procedures	Conforms	Applicable	None.	3.8.4
NSDSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.5	Structural Acceptance Criteria	Conforms	Applicable	None.	3.8.4
NSDSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.6	Materials, Quality Control, and Special Construction Techniques	Conforms	Applicable	None.	3.8.4
NSDSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.7	Testing and Inservice Surveillance Requirements	Conforms	Applicable	None.	3.8.4
NSDSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.8	Masonry Walls	Not Applicable	Not Applicable	Masonry walls are not used in the NuScale standard plant design.	Not Applicable
NSDSRS 3.8.5, Rev. 0: Foundations	II.1	Description of the Foundation	Conforms	Applicable	None.	3.8.5
NSDSRS 3.8.5, Rev. 0: Foundations	II.2	Applicable Codes, Standards, and Specifications	Conforms	Applicable	None.	3.8.5
NSDSRS 3.8.5, Rev. 0: Foundations	II.3	Loads and Load Combinations	Conforms	Applicable	None.	3.8.5
NSDSRS 3.8.5, Rev. 0: Foundations	II.4	Design and Analysis Procedures	Conforms	Applicable	None.	3.8.5
NSDSRS 3.8.5, Rev. 0: Foundations	II.5	Structural Acceptance Criteria	Conforms	Applicable	None.	3.8.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSDSRS 3.8.5, Rev. 0: Foundations	II.6	Materials, Quality Control, and Special Construction Techniques	Conforms	Applicable	None.	3.8.5
NSDSRS 3.8.5, Rev. 0: Foundations	II.7	Testing and Inservice Surveillance Requirements	Conforms	Applicable	None.	3.8.5
SRP 3.9.1, Rev 3: Special Topics for Mechanical Components	II.1	Specification of Transients	Conforms	Applicable	None.	3.9.1
SRP 3.9.1, Rev 3: Special Topics for Mechanical Components	II.2	Computer Programs to be Used in Dynamic and Static Analyses	Conforms	Applicable	None.	3.9.1
SRP 3.9.1, Rev 3: Special Topics for Mechanical Components	II.3	Use of Experimental Stress Analysis Methods in Lieu of Analytical Methods	Not Applicable	Not Applicable	Experimental Stress Analysis Method is not used.	3.9.1
SRP 3.9.1, Rev 3: Special Topics for Mechanical Components	II.4	When Service Level D Limits are Specified for Code Class 1 and Core Support Components	Conforms	Applicable	None.	3.9.1
SRP 3.9.2, Rev 3: Dynamic Testing and Analysis of Systems, Structures, and Components	II.1	Vibration, Thermal Expansion, and Dynamic Effects Testing	Partially Conforms	Applicable	This acceptance criterion is applicable except for aspects related to test performance and associated corrective actions (as required), which are the responsibility of the COL applicant referencing the certified design.	3.9.2
SRP 3.9.2, Rev 3: Dynamic Testing and Analysis of Systems, Structures, and Components	II.2	Compliance with GDC 2	Conforms	Applicable	None.	3.9.2
SRP 3.9.2, Rev 3: Dynamic Testing and Analysis of Systems, Structures, and Components	II.3	Analytical Solutions to Predict Vibrations of Reactor Internals for Prototype Plants	Conforms	Applicable	None.	3.9.2
SRP 3.9.2, Rev 3: Dynamic Testing and Analysis of Systems, Structures, and Components	II.4	Preoperational Vibration and Stress Test Program	Conforms	Applicable	None.	3.9.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 3.9.2, Rev 3: Dynamic Testing and Analysis of Systems, Structures, and Components	II.5	Structural Design Adequacy of Reactor Internals and Reactor Coolant Piping	Conforms	Applicable	None.	3.9.2
SRP 3.9.2, Rev 3: Dynamic Testing and Analysis of Systems, Structures, and Components	II.6	Correlation of Tests and Analyses of Reactor Internals	Conforms	Applicable	None.	3.9.2
SRP 3.9.2, Rev 3: Dynamic Testing and Analysis of Systems, Structures, and Components	II.7	Test Specifications for New Applications	Conforms	Applicable	None.	3.9.2
SRP 3.9.3, Rev 3: ASME Code Class 1, 2, and 3 Components and Component Supports, and Core Support Structures	II.1	Loading Combinations, System Operating Transients, and Stress Limits	Conforms	Applicable	None.	3.9.3
SRP 3.9.3, Rev 3: ASME Code Class 1, 2, and 3 Components and Component Supports, and Core Support Structures	II.2	Design and Installation of Pressure Relief Devices	Conforms	Applicable	None.	3.9.3
SRP 3.9.3, Rev 3: ASME Code Class 1, 2, and 3 Components and Component Supports, and Core Support Structures	II.3	Component Supports	Not Applicable	Not Applicable	NRC Bulletin 88-11 is applicable only to PWR designs that incorporate a pressurizer separate from the reactor pressure vessel, with a surge line connecting the two. In the NuScale design, the pressurizer is integral (i.e., is located within) the reactor pressure vessel, such that there is no pressurizer surge line within which thermal stratification (that is the issue of this bulletin) would occur.	3.9.3
SRP 3.9.4, Rev 3: Control Rod Drive Systems	II.1	Adequacy of Descriptive Information	Conforms	Applicable	This acceptance criterion is applicable (seismic design per RG 1.29) but contains a typographical error. The wording is confusing because it mixes an SRP section reference with a RG.	3.9.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 3.9.4, Rev 3: Control Rod Drive Systems	II.2	Codes and Standards for Construction	Conforms	Applicable	None.	3.9.4
SRP 3.9.4, Rev 3: Control Rod Drive Systems	II.3	Load Combination Sets for Design and Service Conditions Defined in ASME Code Section III, NB-3113	Conforms	Applicable	None.	3.9.4
SRP 3.9.4, Rev 3: Control Rod Drive Systems	II.4	Operability Assurance Program	Conforms	Applicable	None.	3.9.4
SRP 3.9.5, Rev 3: Reactor Pressure Vessel Internals	II.1	Loads, Loading Combinations, and Limits for Portions Constructed to ASME Code Section NG	Conforms	Applicable	None.	3.9.3 3.9.5
SRP 3.9.5, Rev 3: Reactor Pressure Vessel Internals	II.2	Design and Construction of Core Support Structures	Conforms	Applicable	None.	3.9.3 3.9.5
SRP 3.9.5, Rev 3: Reactor Pressure Vessel Internals	II.3	Design Criteria, Loading Conditions, and Analyses for Design of Reactor Internals Other Than Core Support Structures	Conforms	Applicable	None.	3.9.2
SRP 3.9.5, Rev 3: Reactor Pressure Vessel Internals	II.4	Deformation Limits for Reactor Internals	Conforms	Applicable	None.	3.9.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 3.9.5, Rev 3: Reactor Pressure Vessel Internals	II.5	Design of Reactor Internals to Accommodate Asymmetric Blowdown Loads From Postulated Pipe Ruptures	Partially Conforms	Applicable	The intent of subtier NUREG-0609 is applicable but the specific language refers to a different type of LWR and SSC conditions not relevant to the NuScale design. Specifically, this guidance provides methodology for evaluation of loading transients and structural components, including containment subcompartment analysis, when a double-ended guillotine break of reactor coolant loop piping occurs at the reactor vessel inlet. The NuScale containment vessel design does not involve subcompartments. In addition, the NuScale design does not have reactor coolant loops. Notwithstanding the above, this guidance is applicable to the evaluation of loading transients and structural components for postulated breaks of chemical and volume control system (CVCS) piping and piping at the reactor vent valves.	3.9.5
SRP 3.9.5, Rev 3: Reactor Pressure Vessel Internals	II.6	Effects of Flow-Induced Vibration and Acoustic Resonances (Including Appendix A)	Partially Conforms	Applicable	This acceptance criterion (including Appendix A) is applicable except for aspects that are BWR-specific.	3.9.5
SRP 3.9.6, Rev 3: Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	II.1	Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints	Partially Conforms	Applicable	This acceptance criterion is applicable except for aspects related to functional design, qualification, and testing of safety-related pumps. Safety-related pumps are not used in the NuScale design.	3.9.6

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 3.9.6, Rev 3: Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	II.2	Inservice Testing Program for Pumps	Partially Conforms	Applicable	This acceptance criterion is applicable except for aspects related to inservice testing of safety-related pumps. Safety-related pumps are not used in the NuScale design. The only pumps that fall within the scope of this criterion in the NuScale design are the CVCS pumps. These pumps are ASME Class III because they contain reactor coolant during normal operation, but they serve no safety function. Therefore, relief from some testing requirements that are intended to confirm pumping capability may be requested in accordance with Acceptance Criterion II.5 of this SRP section.	3.9.6
SRP 3.9.6, Rev 3: Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	II.3	Inservice Testing Program for Valves	Conforms	Applicable	None.	3.9.6
SRP 3.9.6, Rev 3: Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	II.4	Inservice Testing Program for Dynamic Restraints	Conforms	Applicable	None.	3.9.6
SRP 3.9.6, Rev 3: Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	II.5	Relief Requests and Proposed Alternatives	Conforms	Applicable	None.	3.9.6

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 3.9.6, Rev 3: Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	II.6	Operational Programs	Not Applicable	Applicable	This acceptance criterion is related to operational activities, including implementation of pre-service testing, inservice testing and inspection, and motor-operated valve testing programs, that are the responsibility of the COL applicant referencing the certified design.	Not Applicable
SRP 3.9.7, Rev 0: Risk-Informed Inservice Testing	All	Various	Not Applicable	Applicable	This SRP section and its acceptance criteria are applicable only to reactor licensees and applicants that are developing or revising a risk-informed, performance-based inservice testing program. Development and implementation of a risk-informed, performance-based inservice testing program would be the responsibility of COL applicants that reference the NuScale certified design (upon approval by the NRC), and that elect to implement such a program.	Not Applicable
SRP 3.9.8, Rev 0: Standard Review Plan for the Review of Risk-Informed Inservice Inspection of Piping	All	Various	Not Applicable	Applicable	This SRP section and its acceptance criteria are applicable only to reactor licensees and applicants that are developing or revising a risk-informed, performance-based inservice inspection program for piping. Development and implementation of a risk-informed, inservice inspection program for piping would be the responsibility of COL applicants that reference the NuScale certified design (upon approval by the NRC), and that elect to implement such a program.	Not Applicable
SRP 3.10, Rev 3: Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	II.1	Qualification of Electrical Equipment and Associated Supports	Conforms	Applicable	None.	3.10



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 3.10, Rev 3: Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	II.2	Testing of Instrumentation Described in RG 1.97	Partially Conforms	Applicable	See RG 1.97 in Table 1.9-2	3.10
SRP 3.10, Rev 3: Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	II.3	Experience-Based Qualification	Not Applicable	Not Applicable	Experience based seismic qualification is not used.	3.10
SRP 3.10, Rev 3: Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	II.4	Records	Conforms	Applicable	The design certification indicates that a Records program is required and includes a COL item for the COL applicant to maintain one.	3.10
SRP 3.10, Rev 3: Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	II.5	Qualification Program for Valves that are Part of the Reactor Coolant Pressure Boundary	Conforms	Applicable	None.	3.10
SRP 3.10, Rev 3: Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	II.6	Documentation of Qualification Program	Conforms	Applicable	None.	3.10
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.1	Application of RG 1.89 for Environmental Qualification Program per 10 CFR 50.49	Partially Conforms	Applicable	See RG 1.89 in Table 1.9-2.	3.11
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.2	Application of Clarification Related to IEEE Std. 323 Criteria	Conforms	Applicable	None.	3.11
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.3	Application of RG 1.63 for Environmental Design and Qualification of Electrical Penetration Assemblies	Not Applicable	Not Applicable	See RG 1.63 in Table 1.9-2.	3.11.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.4	Application of RG 1.73 for Environmental Design and Qualification of Class 1E Electric Valve Operators	Conforms	Applicable	None.	3.11.2
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.5	Application of RG 1.89 for Environmental Qualification of Electrical Equipment Important to Safety	Partially Conforms	Applicable	See RG 1.89 in Table 1.9-2.	3.11.2
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.6	Application of RG 1.97 for Environmental Design and Qualification of PostAccident Monitoring Equipment	Partially Conforms	Applicable	See RG 1.97 in Table 1.9-2.	3.11.2
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.7	Application of RG 1.152 for Environmental design and qualification of computer-specific requirements	Conforms	Applicable	None.	3.11
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.8	Application of RG 1.153 for Environmental design and qualification of power, instrumentation, and control portions of the safety systems	Conforms	Applicable	None.	3.11
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.9	Application of RG 1.209 for Environmental design and qualification of safety-related computer-based I&C systems in mild environments	Conforms	Applicable	None.	3.11
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.10	Application of RG 1.211 for Environmental Qualification of Class 1E Electric Cables and Field Splices	Conforms	Applicable	None.	3.11.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.11	Application of RG 1.156 for Environmental Qualification of Class 1E Connection Assemblies	Conforms	Applicable	None.	3.11.2
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.12	Application of RG 1.158 for Environmental Qualification of Class 1E Lead Storage Batteries	Not Applicable	Not Applicable	See RG 1.158 in Table 1.9-2.	3.11.2
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.13	Application of RG 1.180 for Electromagnetic and Radio- Frequency Interference in Safety Related I&C Equipment	Partially Conforms	Applicable	See RG 1.180 in Table 1.9-2.	3.11.2
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.14	Application of RG 1.183 for Accident Source Term Used in Environmental Design and Qualification of Equipment Important to Safety	Partially Conforms	Applicable	See RG 1.183 in Table 1.9-2.	3.11.2
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.15	Application of RG 1.100 for Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants	Partially Conforms	Applicable	See RG 1.100 in Table 1.9-2.	3.11
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.16	Application of RG 1.204 for Environmental design and qualification of the lightning protection system	Not Applicable	Not Applicable	Lightning protection is not applicable to EQ because it is associated with an external/natural event.	3.11
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.17	Effects of Environmental Conditions for All Important to Safety Equipment	Conforms	Applicable	None.	3.11

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.18	Suitability of Materials, Parts, and Equipment Essential to Safety- Related Functions	Conforms	Applicable	None.	3.11
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.19	Qualification of Nonmetallic Parts	Conforms	Applicable	None.	3.11
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.20	Design/Purchase Specifications of Equipment to Perform Under Applicable Environmental Conditions	Conforms	Applicable	None.	3.11
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.21	Applicable documentation for Environmental Design and Qualification of Safety-Related Mechanical, Electrical, and I&C Equipment	Conforms	Applicable	None.	3.11
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.22	Maintenance/surveillance programs to provide assurance Assurance of Environmental Design and Qualification Status of Equipment in Mild and Harsh Environments	Not Applicable	Applicable	The programs are described and maintained by the COL applicant.	3.11
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.23	Operational Program Implementation	Not Applicable	Applicable	This is a COL applicant item.	3.11
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.24	Exposure of Organic Components on Engineered Safety Features Systems	Conforms	Applicable	None.	3.11

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSDSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.25	Design and Procurement Specifications	Not Applicable	Applicable	This is a COL applicant item.	3.11
SRP 3.12, Rev 1: ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports	II.A	Piping Analysis Methods	Conforms	Applicable	None.	3.12.3
SRP 3.12, Rev 1: ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports	II.B	Piping Modeling Techniques	Conforms	Applicable	None.	3.12.4
SRP 3.12, Rev 1: ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports	II.C	Piping Stress Analysis Criteria	Conforms	Applicable	None.	3.12.5
SRP 3.12, Rev 1: ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports	II.D	Piping Support Design	Conforms	Applicable	None.	3.12.6
SRP 3.13, Rev. 0: Threaded Fasteners - ASME Code Class 1, 2, and 3	II.1	Design Aspects (Including Table 3.13-1)	Conforms	Applicable	None.	3.13.1
SRP 3.13, Rev. 0: Threaded Fasteners - ASME Code Class 1, 2, and 3	II.2	Preservice and Inservice Inspection Requirements (Including Table 3.13-2)	Conforms	Applicable	None.	3.13.2
BTP 3-1, Rev 2: Classification of Main Steam Components Other Than the Reactor Coolant Pressure Boundary for BWR Plants	All	-	Not Applicable	Not Applicable	This guidance is applicable only to BWR plants.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
BTP 3-2, Rev 2: Classification of BWR/6 Main Steam and Feedwater Components Other Than the Reactor Coolant Pressure Boundary	All	-	Not Applicable	Not Applicable	This guidance is applicable only to BWR plants.	Not Applicable
BTP 3-3, Rev 3: Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	B.1	Plant Arrangement	Conforms	Applicable	None.	3.6
BTP 3-3, Rev 3: Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	B.2	Design Features	Conforms	Applicable	None.	3.6
BTP 3-3, Rev 3: Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	B.3	Analyses and Effects of Postulated Piping Failures	Conforms	Applicable	None.	3.6
BTP 3-3, Rev 3: Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	B.4	Implementation	Conforms	Applicable	None.	3.6
BTP 3-4, Rev 2: Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment	B.A	High-Energy Fluid System Piping	Conforms	Applicable	None.	3.6 15.1 15.2 15.5 15.6
BTP 3-4, Rev 2: Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment	B.B	Moderate-Energy Fluid System Piping	Conforms	Applicable	None.	3.6 15.1 15.2 15.5 15.6
BTP 3-4, Rev 2: Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment	B.C	Type of Breaks and Leakage Cracks in Fluid System Piping	Conforms	Applicable	None.	3.6 15.1 15.2 15.5 15.6

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 4.2, Rev 3: Fuel System Design	II.1	Design Bases	Conforms	Applicable	None.	4.2.1
SRP 4.2, Rev 3: Fuel System Design	II.1.A	Fuel System Damage	Conforms	Applicable	None.	4.2.1
SRP 4.2, Rev 3: Fuel System Design	II.1.B	Fuel Rod Failure	Conforms	Applicable	None.	4.2.1 4.2.3
SRP 4.2, Rev 3: Fuel System Design	II.1.C	Fuel Coolability	Conforms	Applicable	None.	4.2.1
SRP 4.2, Rev 3: Fuel System Design	II.2	Description and Design Drawings	Conforms	Applicable	None.	4.2.2
SRP 4.2, Rev 3: Fuel System Design	II.3	Design Evaluation	Conforms	Applicable	None.	4.2.1 4.2.3 4.2.4
SRP 4.2, Rev 3: Fuel System Design	II.4	Testing, Inspection, and Surveillance Plans	Conforms	Applicable	None.	4.2.1 4.2.4
SRP 4.2, Rev 3: Fuel System Design	App A	Evaluation of Fuel Assembly Structural Response to Externally Applied Forces	Conforms	Applicable	None.	4.2.1
SRP 4.2, Rev 3: Fuel System Design	App B	Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents	Conforms	Applicable	None.	4.2.1 15.0.0
SRP 4.3, Rev 0: Nuclear Design	II.1	Design Limits for Power Densities and Power Distributions	Conforms	Applicable	None.	4.3.1
SRP 4.3, Rev 0: Nuclear Design	II.2	Reactivity Coefficients	Conforms	Applicable	None.	4.3.2
SRP 4.3, Rev 0: Nuclear Design	II.3	Control Rod Patterns and Reactivity Worth	Conforms	Applicable	None.	4.3.2
SRP 4.3, Rev 0: Nuclear Design	II.4	Analytical Methods and Data	Conforms	Applicable	None.	4.3.3
NSDSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.1	Fuel Design Limits, Core Design, and Thermal Margin	Conforms	Applicable	None.	4.4.1 4.4.2
NSDSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.2	Subchannel Hydraulic Analysis Codes	Conforms	Applicable	None.	4.4.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.3	Core Oscillations and Thermal- Hydraulic Instabilities	Conforms	Applicable	None.	4.4.7
NSDSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.4	RPV Fluid Flow Calculations	Conforms	Applicable	None.	4.4.4
NSDSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.5	Technical Specifications	Conforms	Applicable	None.	16.1
NSDSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.6	Preoperational and Initial Test Programs	Conforms	Applicable	None.	4.4.2
NSDSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.7	Loose Parts Monitoring System	Departure	Applicable	Low flow in primary systems precludes damage from loose parts and the need for loose parts monitoring system.	4.4.6
NSDSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.8	Critical Heat Flux Calculations and Process Monitoring	Conforms	Applicable	None.	4.4.2 4.4.4 4.4.6
NSDSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.9	Instrumentation and Procedures for Detection and Recovery from Inadequate Core Cooling	Conforms	Applicable	None.	4.4.6
NSDSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.10	Core Stability Performance During Anticipated Transient without Scram Event	Not Applicable	Not Applicable	Diverse RTS signals prevent an ATWS from occurring. This prevents flow instabilities from occurring, so this AC is not applicable based on the current ATWS approach.	4.4.7 15.8
SRP 4.5.1, Rev 3: Control Rod Drive Structural Materials	II.1	Materials Specifications	Conforms	Applicable	RG 1.85 was withdrawn in 2004. Guidance was updated and incorporated into RG 1.84.	4.5.1
SRP 4.5.1, Rev 3: Control Rod Drive Structural Materials	II.2	Austenitic Stainless Steel Components	Conforms	Applicable	The NuScale QAPD is based on ANSI/ ASME NQA-1-2008 with NQA-1a-2009 addenda, as endorsed by RG 1.28, Rev. 4.	4.5.1
SRP 4.5.1, Rev 3: Control Rod Drive Structural Materials	II.3	Other Materials	Conforms	Applicable	None.	4.5.1



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 4.5.1, Rev 3: Control Rod Drive Structural Materials	II.4	Cleaning and Cleanliness Control	Conforms	Applicable	The NuScale QAPD is based on ANSI/ASME NQA-1-2008 with NQA-1a-2009 addenda, as endorsed by RG 1.28, Rev. 4.	4.5.1
SRP 4.5.2, Rev 3: Reactor Internal and Core Support Structure Materials	II.1	Materials	Conforms	Applicable	None.	4.5.2
SRP 4.5.2, Rev 3: Reactor Internal and Core Support Structure Materials	II.2	Controls on Welding	Conforms	Applicable	None.	4.5.2
SRP 4.5.2, Rev 3: Reactor Internal and Core Support Structure Materials	II.3	Nondestructive Examination	Conforms	Applicable	None.	4.5.2
SRP 4.5.2, Rev 3: Reactor Internal and Core Support Structure Materials	II.4	Austenitic Stainless Steels	Conforms	Applicable	None.	4.5.2
SRP 4.5.2, Rev 3: Reactor Internal and Core Support Structure Materials	II.5	Other Materials	Conforms	Applicable	None.	4.5.2
SRP 4.6, Rev 2: Functional Design of Control Rod Drive System	II.1	Environmental and Dynamic Effects - GDC 4	Conforms	Applicable	None.	4.6.2
SRP 4.6, Rev 2: Functional Design of Control Rod Drive System	II.2	Failure Modes and Effects - GDC 23	Conforms	Applicable	None.	4.6.2
SRP 4.6, Rev 2: Functional Design of Control Rod Drive System	II.3	Single Malfunction - GDC 25	Conforms	Applicable	None.	4.6.2
SRP 4.6, Rev 2: Functional Design of Control Rod Drive System	II.4	Operational Control and Reliability - GDC 26	Conforms	Applicable	NuScale does not interpret GDC 26 as requiring two safety related means of reactivity control. One of the independent reactivity control systems used to meet the requirements of GDC 26 in the NuScale design is the chemical volume control system, which is not safety related.	4.6.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 4.6, Rev 2: Functional Design of Control Rod Drive System	II.5	Combined Capability - GDC 27	Departure	Not Applicable	The NuScale design bases conform to a design-specific Principal Design Criterion (PDC) in lieu of GDC 27, as reflected in Section 3.1.	3.1 4.2 4.3 4.6.2
SRP 4.6, Rev 2: Functional Design of Control Rod Drive System	II.6	Reactivity Limits - GDC 28	Conforms	Applicable	None.	4.6.0.2 4.6.2
SRP 4.6, Rev 2: Functional Design of Control Rod Drive System	II.7	Protection Against Anticipated Operational Occurrences - GDC 29	Conforms	Applicable	None.	4.6.2
SRP 4.6, Rev 2: Functional Design of Control Rod Drive System	II.8	BWR Alternate Rod Injection System	Not Applicable	Not Applicable	This guidance is applicable only to BWR plants.	Not Applicable
BTP 4-1, Rev 3: Westinghouse Constant Axial Offset Control (CAOC)	All	-	Not Applicable	Not Applicable	This BTP is applicable only to PWR designs that use the Constant Axial Offset Control operating scheme. NuScale does not intend to use the Constant Axial Offset Control operating scheme.	Not Applicable
SRP 5.2.1.1, Rev 3: Compliance with the Codes and Standards Rule, 10 CFR 50.55a	II	Use of RG 1.26 to meet GDC 1 and 10 CFR 50.55a	Conforms	Applicable	None.	5.2.1
SRP 5.2.1.2, Rev 3: Applicable Code Cases	II.1	Use of RG 1.84 to meet GDC 1 and 10 CFR 50.55a	Conforms	Applicable	None.	5.2.1
SRP 5.2.1.2, Rev 3: Applicable Code Cases	II.2	Use of RG 1.147 to meet GDC 1 and 10 CFR 50.55a	Partially Conforms	Applicable	See RG 1.147 in Table 1.9-2.	5.2.1
SRP 5.2.1.2, Rev 3: Applicable Code Cases	II.3	Use of RG 1.192 to meet GDC 1 and 10 CFR 50.55a	Partially Conforms	Applicable	See RG 1.192 in Table 1.9-2.	5.2.1
SRP 5.2.2, Rev 3: Overpressure Protection	II.1	Material Specifications	Conforms	Applicable	None.	5.2.2
SRP 5.2.2, Rev 3: Overpressure Protection	II.2	Design Requirements for BWRs Operating at Power	Not Applicable	Not Applicable	This guidance is applicable only to BWR plants.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 5.2.2, Rev 3: Overpressure Protection	II.3	Design Requirements for PWRs Operating at Power	Partially Conforms	Applicable	The overpressure analysis does not assume a secondary safety-grade signal from the RPS initiates the reactor trip. NuScale does not have a secondary safety-grade reactor trip system.	5.2.2
SRP 5.2.2, Rev 3: Overpressure Protection	II.4	Design Requirements for PWRs Operating at Low Temperature (Startup, Shutdown)	Conforms	Applicable	None.	5.2.2
SRP 5.2.2, Rev 3: Overpressure Protection	II.5	Testing and Inspections	Conforms	Applicable	None.	5.2.2
SRP 5.2.2, Rev 3: Overpressure Protection	II.6	Technical Specifications	Partially Conforms	Applicable	Certain subtier guidance documents referenced in this acceptance criterion are not applicable or only partially applicable.	5.2.2
SRP 5.2.2, Rev 3: Overpressure Protection	II.7	TMI Action Plan Requirements	Conforms	Applicable	None.	5.2.2
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.1	Material Specifications	Partially Conforms	Applicable	This acceptance criterion is applicable except for references to subtier guidance that is applicable only to BWRs.	5.2.3
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.2	Compatibility of Materials with the Reactor Coolant	Partially Conforms	Applicable	This acceptance criterion is applicable except for references to subtier guidance that is applicable only to BWRs.	5.2.3
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.3	Fabrication and Processing of Ferritic Materials	Conforms	Applicable	None.	5.2.3
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.3.A	Fracture Toughness - 10 CFR 50, Appendix G	Conforms	Applicable	None.	5.2.3
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.3.B	Control of Ferritic Steel Welding	Conforms	Applicable	None.	5.2.3
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.3.C	NDE of Ferritic Steel Tubular Products	Conforms	Applicable	None.	5.2.3

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.4	Fabrication and Processing of Austenitic Stainless Steel	Partially Conforms	Applicable	This acceptance criterion is applicable except for references to subtier guidance that is applicable only to BWRs or to large LWRs that use nonmetallic thermal insulation on reactor coolant pressure boundary components.	5.2.3
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.4.A	GDC 4 Compatibility of Components - Measures to Avoid Sensitization in Austenitic Stainless Steel	Partially Conforms	Applicable	This acceptance criterion is applicable except for references to subtier guidance that is applicable only to BWRs.	5.2.3
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.4.B	GDC 4 Compatibility of Components - Controls to Avoid Stress Corrosion Cracking in Austenitic Stainless Steel	Partially Conforms	Applicable	This acceptance criterion is applicable except for references to subtier guidance that is applicable only to BWRs, and to subtier RG 1.37, which endorses use of NQA-1-1994. The NuScale design is based on NQA-1-2008 and the NQA-1a-2009 addenda, rather than NQA-1-1994.	5.2.3
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.4.C	Compatibility of Austenitic Stainless Steel Materials with Thermal Insulation	Not Applicable	Not Applicable	This acceptance criterion is applicable only to LWRs that use nonmetallic thermal insulation on reactor coolant pressure boundary components. NuScale is not intending to use nonmetallic thermal insulation on reactor coolant pressure boundary components.	5.2.3
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.4.D	Control of Welding of Austenitic Stainless Steels	Partially Conforms	Applicable	This acceptance criterion is applicable except for references to subtier guidance that is applicable only to BWRs.	5.2.3
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.4.E	NDE of Austenitic Stainless Steel Tubular Products	Conforms	Applicable	None.	5.2.3
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.4.G	Operational Programs	Not Applicable	Applicable	This acceptance criterion governs plant-specific programmatic information that is the responsibility of the COL applicant.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.1	System Boundary Subject to Inspection	Conforms	Applicable	None.	5.2.4
NSDSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.2	Accessibility	Conforms	Applicable	None.	5.2.4
NSDSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.3	Examination Categories and Methods	Conforms	Applicable	None.	5.2.4
NSDSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.4	Inspection Intervals	Conforms	Applicable	None.	5.2.4
NSDSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.5	Evaluation of Examination Results	Conforms	Applicable	None.	5.2.4
NSDSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.6	System Pressure Tests	Conforms	Applicable	None.	5.2.4
NSDSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.7	Code Exemptions	Conforms	Applicable	None.	5.2.4
NSDSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.8	Relief Requests	Conforms	Applicable	As stated, at the DC or COL application, ASME Section XI examination requirements should be met and no relief requests submitted. 10 CFR 50.55a(z)(1) allows alternatives that provide an acceptable level of quality and safety.	5.2.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.9	Code Cases	Conforms	Applicable	None.	5.2.4
NSDSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.10	Augmented ISI to Protect Against Postulated Piping Failures	Conforms	Applicable	None.	5.2.4
NSDSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.11	Other Inspection Programs	Partially Conforms	Applicable	Although a boric acid control program will not be fully established, a brief description of the program is provided in the DCA.	5.2.4
NSDSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.12	Operational Programs	Not Applicable	Applicable	This acceptance criterion governs plant-specific programmatic information that is the responsibility of the COL applicant.	Not Applicable
NSDSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.13	ITAAC	Partially Conforms	Applicable	A portion of this acceptance criterion is applicable only to COL applicants.	5.2.4
NSDSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.14	Risk Informed ISI Program	Not Applicable	Applicable	This acceptance criterion governs plant-specific programmatic information that is the responsibility of the COL applicant.	Not Applicable
NSDSRS 5.2.5, Rev 0: Reactor Coolant Pressure Boundary Leakage Detection	II.1	Criteria to Meet GDC 2	Conforms	Applicable	None.	5.2.5
NSDSRS 5.2.5, Rev 0: Reactor Coolant Pressure Boundary Leakage Detection	II.2	Criteria to Meet GDC 14	Conforms	Applicable	None.	5.2.5
NSDSRS 5.2.5, Rev 0: Reactor Coolant Pressure Boundary Leakage Detection	II.3	Criteria to Meet GDC 30	Conforms	Applicable	None.	5.2.5
NSDSRS 5.3.1, Rev 0: Reactor Vessel Materials	II.1	Materials	Conforms	Applicable	None.	5.3.1

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 5.3.1, Rev 0: Reactor Vessel Materials	II.2	Special Processes Used for Manufacture and Fabrication of Components	Conforms	Applicable	None.	5.3.1
NSDSRS 5.3.1, Rev 0: Reactor Vessel Materials	II.3	Special Methods for Nondestructive Examination	Conforms	Applicable	None.	5.3.1
NSDSRS 5.3.1, Rev 0: Reactor Vessel Materials	II.4	Special Controls and Special Processes Used for Ferritic Steels and Austenitic Stainless Steels	Conforms	Applicable	None.	5.3.1
NSDSRS 5.3.1, Rev 0: Reactor Vessel Materials	II.5	Fracture Toughness	Conforms	Applicable	None.	5.3.1
NSDSRS 5.3.1, Rev 0: Reactor Vessel Materials	II.6	Material Surveillance	Conforms	Applicable	None.	5.3.1
NSDSRS 5.3.1, Rev 0: Reactor Vessel Materials	II.7	Reactor Vessel Fasteners	Conforms	Applicable	None.	5.3.1
NSDSRS 5.3.2, Rev 0: Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	II.1.A	Pressure-Temperature - Applicable Regulations, Codes, and Basis Documents	Conforms	Applicable	None.	5.3.2
NSDSRS 5.3.2, Rev 0: Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	II.1.B	Pressure-Temperature Requirements	Conforms	Applicable	None.	5.3.2
NSDSRS 5.3.2, Rev 0: Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	II.2.A	Upper-Shelf Energy - Applicable Regulations, Codes, and Basis Documents	Conforms	Applicable	None.	5.3.2
NSDSRS 5.3.2, Rev 0: Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	II.2.B	Upper-Shelf Energy Requirements	Conforms	Applicable	None.	5.3.1 5.3.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 5.3.2, Rev 0: Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	II.3.A	Pressurized Thermal Shock - Applicable Regulations, Codes, and Basis Documents	Conforms	Applicable	None.	5.3.2
NSDSRS 5.3.2, Rev 0: Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	II.3.B	Pressurized Thermal Shock Requirements	Conforms	Applicable	None.	5.3.2
NSDSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.1	Design	Conforms	Applicable	None.	5.3.3
NSDSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.2	Materials of Construction	Conforms	Applicable	None.	5.3.3
NSDSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.3	Fabrication Methods	Conforms	Applicable	None.	5.3.3
NSDSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.4	Inspection Requirements	Conforms	Applicable	None.	5.3.3
NSDSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.5	Shipment and Installation	Conforms	Applicable	None.	5.3.3
NSDSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.6	Operating Conditions	Conforms	Applicable	None.	5.3.3
NSDSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.7	Inservice Surveillance	Conforms	Applicable	None.	5.3.3
NSDSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.8	Operational Programs	Not Applicable	Applicable	This acceptance criterion governs plant-specific programmatic information that is the responsibility of the COL applicant.	Not Applicable
NSDSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.9	10 CFR 52.47(b)(1) compliance	Not Applicable	Applicable	This requirement applies to plant- specific verification and is the responsibility of the COL applicant.	Not Applicable



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 5.4.1.1, Rev 3: Pump Flywheel Integrity (PWR)	All	Various	Not Applicable	Not Applicable	This SRP section and its acceptance criteria (II.1 through II.7) are applicable only to PWR designs that use reactor coolant pumps. The NuScale reactor design differs from that of typical PWRs in that the NuScale design does not require or include reactor coolant pumps. Rather, the NuScale design uses passive natural circulation of the primary coolant, eliminating the need for reactor coolant pumps.	Not Applicable
NSDSRS 5.4.2.1, Rev 0: Steam Generator Materials	II.1	Selection, Processing, Testing, and Inspection of Materials	Conforms	Applicable	None.	5.4.1
NSDSRS 5.4.2.1, Rev 0: Steam Generator Materials	II.2	Steam Generator Design	Conforms	Applicable	None.	5.4.1
NSDSRS 5.4.2.1, Rev 0: Steam Generator Materials	II.3	Fabrication and Processing of Ferritic Materials	Conforms	Applicable	None.	5.4.1
NSDSRS 5.4.2.1, Rev 0: Steam Generator Materials	II.4	Fabrication and Processing of Austenitic Stainless Steel	Conforms	Applicable	None.	5.4.1
NSDSRS 5.4.2.1, Rev 0: Steam Generator Materials	II.5	Compatibility of Materials with the Primary (Reactor) and Secondary Coolant and Cleanliness Control	Conforms	Applicable	None.	5.4.1
NSDSRS 5.4.2.1, Rev 0: Steam Generator Materials	II.6	Provisions for Accessing the Secondary Side of the Steam Generator	Conforms	Applicable	None.	5.4.1
NSDSRS 5.4.2.2, Rev 0: Steam Generator Program	II.1	Steam Generator Tube Susceptibility to Degradation	Conforms	Applicable	None.	5.4
NSDSRS 5.4.2.2, Rev 0: Steam Generator Program	II.2	Steam Generator Monitoring Program Elements	Partially Conforms	Applicable	A portion of this acceptance criterion is applicable to COL applicants referencing a certified design. Consistent with this guidance, the NuScale technical specifications that are included with its application for design certification will specify the guidelines for a SG Monitoring Program.	5.4.1

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 5.4.2.2, Rev 0: Steam Generator Program	II.3	Steam Generator Program Elements in Technical Specifications	Partially Conforms	Applicable	Certain subtier guidance documents referenced in this acceptance criterion are only partially applicable.	5.4.1
NSDSRS 5.4.2.2, Rev 0: Steam Generator Program	II.4	Steam Generator Tube Repair Criteria	Conforms	Applicable	None.	5.4.1
NSDSRS 5.4.2.2, Rev 0: Steam Generator Program	II.5	Steam Generator Tube Repair Methods	Conforms	Applicable	None.	5.4.1
NSDSRS 5.4.2.2, Rev 0: Steam Generator Program	II.6	Steam Generator Tube Preservice Inspection	Conforms	Applicable	None.	5.4.1
NSDSRS 5.4.2.2, Rev 0: Steam Generator Program	II.7	Steam Generator Tube Inspection	Conforms	Applicable	None.	5.4.1
NSDSRS 5.4.2.2, Rev 0: Steam Generator Program	II.8	Periodic Tube Inspection and Testing in Certified Design Technical Specifications	Partially Conforms	Applicable	A portion of this acceptance criterion is applicable to COL applicants referencing a certified design. Consistent with this guidance, the NuScale technical specifications that are included with its application for design certification will specify the guidelines for periodic inspection and testing of the steam generator tube portion of the reactor coolant pressure boundary.	5.4.1
NSDSRS 5.4.2.2, Rev 0: Steam Generator Program	II.9	Operational Programs	Not Applicable	Applicable	This acceptance criterion governs plant-specific programmatic activities that are the responsibility of the COL applicant referencing a certified design.	Not Applicable
SRP 5.4.6, Rev 4: Reactor Core Isolation Cooling System (BWR)	All	Various	Not Applicable	Not Applicable	This SRP section and its acceptance criteria (II.1 through II.10) are applicable only to BWRs.	Not Applicable
NSDSRS 5.4.7, Rev 0: Decay Heat Removal (DHR) System Responsibilities	II.1 thru II.3	Various	Conforms	Applicable	None.	5.4.3

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 5.4.7, Rev 0: Decay Heat Removal (DHR) System Responsibilities	II.4	GDC 5	Not Applicable	Not Applicable	This criterion only applies to shared SSCs and RCPB. The decay heat removal system (DHRS) is not shared between reactor modules. Each reactor module contains its own DHRS. Also, the DHRS is connected to the secondary system and is not considered part of the RCPB.	Not Applicable
NSDSRS 5.4.7, Rev 0: Decay Heat Removal (DHR) System Responsibilities	II.5	GDC 14	Not Applicable	Not Applicable	The DHRS is connected to the secondary system and does not directly interface with the RCPB.	Not Applicable
NSDSRS 5.4.7, Rev 0: Decay Heat Removal (DHR) System Responsibilities	II.6	GDC 19	Conforms	Applicable	None.	5.4.3
NSDSRS 5.4.7, Rev 0: Decay Heat Removal (DHR) System Responsibilities	II.7	GDC 34	Conforms	Applicable	None.	5.4.3
NSDSRS 5.4.7, Rev 0: Decay Heat Removal (DHR) System Responsibilities	II.8	GDC 54	Alternate Compliance	Applicable	This closed-loop DHRS outside the containment is directly connected to the closed-loop SG system within the RPV providing dual passive barriers between the RCS and the reactor pool outside the NPM. Breaches of this piping system outside containment is not considered credible because the system is a welded design with a system design pressure equivalent to the RPV, designed to Class 2 requirements in accordance with ASME BPV Code, Section III, and meets the applicable criteria of NRC Branch Technical Position 3-4, Revision 2. As a result, leakage detection and isolation capabilities of this piping system from containment are not considered important to safety.	5.4.3

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 5.4.7, Rev 0: Decay Heat Removal (DHR) System Responsibilities	II.9	DHRS Interface with other systems	Conforms	Applicable	None.	5.4.3
SRP 5.4.8, Rev 3: Reactor Water Cleanup System (BWR)	All	Various	Not Applicable	Not Applicable	This SRP section and its acceptance criteria (II.1 through II.4) are applicable only to BWRs.	Not Applicable
SRP 5.4.11, Rev 4: Pressurizer Relief Tank	All	Various	Not Applicable	Not Applicable	This SRP section and its acceptance criteria (II.1 and II.2) are applicable only to PWRs that use a pressurizer relief tank. Specifically, the NuScale design differs from that of large PWRs in that no pressurizer relief tank is used in the NuScale design. Fluid relieved through the reactor coolant system overpressure protection system is routed directly to the containment vessel.	Not Applicable
SRP 5.4.12, Rev 1: Reactor Coolant System High Point Vents	All	Various	Not Applicable	Not Applicable	Because of the integral reactor coolant system configuration, non-condensable gases accumulating in the pressurizer space will not interfere with core cooling during or after design basis accidents. As described in Section 3.1, the NuScale design supports an exemption from 50.46a and considers that 10 CFR 50.34(f)(2)(vi) is not technically relevant to the NuScale design.	Not Applicable
SRP 5.4.13, (March 2007): Isolation Condenser System (BWR)	All	Various	Not Applicable	Not Applicable	This SRP section and its acceptance criteria (II.1 through II.12) are applicable only to BWRs.	Not Applicable
BTP 5-1, Rev 3: Monitoring of Secondary Side Water Chemistry in PWR Steam Generators	B.1	Secondary Water Chemistry Program Meeting Industry Guidelines	Conforms	Applicable	None.	5.4.1 10.3.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
BTP 5-1, Rev 3: Monitoring of Secondary Side Water Chemistry in PWR Steam Generators	B.2	Sampling Schedule for Critical Parameters	Partially Conforms	Applicable	A portion of this acceptance criterion governs information that is site-specific and thus is the responsibility of the COL applicant referencing the certified design.	5.4.1 10.3.5
BTP 5-1, Rev 3: Monitoring of Secondary Side Water Chemistry in PWR Steam Generators	B.3	Records	Partially Conforms	Applicable	A portion of this acceptance criterion governs information that is site-specific and thus is the responsibility of the COL applicant referencing the certified design.	5.4.1 10.3.5
BTP 5-1, Rev 3: Monitoring of Secondary Side Water Chemistry in PWR Steam Generators	B.4	Program Change Evaluation and Reporting	Not Applicable	Applicable	This acceptance criterion governs information that is site-specific and thus is the responsibility of the COL applicant referencing the certified design.	Not Applicable
BTP 5-2, Rev 3: Overpressurization Protection of Pressurized-Water Reactors While Operating at Low Temperatures	B.1	System Design, Installation, and Capabilities to Prevent Exceeding Technical Specifications and NRC Regulatory Requirements	Conforms	Applicable	None.	5.2.2
BTP 5-2, Rev 3: Overpressurization Protection of Pressurized-Water Reactors While Operating at Low Temperatures	B.2	Low-Temperature Overpressure Protection Operability	Partially Conforms	Applicable	Conforms to ASME Section XI Appendix G Criteria.	5.2.2
BTP 5-2, Rev 3: Overpressurization Protection of Pressurized-Water Reactors While Operating at Low Temperatures	B.3	System Designed to Withstand Single Active Component Failure	Conforms	Applicable	None.	5.2.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
BTP 5-2, Rev 3: Overpressurization Protection of Pressurized- Water Reactors While Operating at Low Temperatures	B.4	System Instrumentation and Controls Design	Conforms	Applicable	None.	5.2.2
BTP 5-2, Rev 3: Overpressurization Protection of Pressurized- Water Reactors While Operating at Low Temperatures	B.5	System Operability Testing	Conforms	Applicable	None.	5.2.2 Ch 16
BTP 5-2, Rev 3: Overpressurization Protection of Pressurized- Water Reactors While Operating at Low Temperatures	B.6	Applicable Guidance	Conforms	Applicable	None.	5.2.2
BTP 5-2, Rev 3: Overpressurization Protection of Pressurized- Water Reactors While Operating at Low Temperatures	B.7	System Design to Withstand Operating-Basis Earthquake	Conforms	Applicable	None.	5.2.2
BTP 5-2, Rev 3: Overpressurization Protection of Pressurized- Water Reactors While Operating at Low Temperatures	B.8	Backup Electrical Power Source	Partially Conforms	Applicable	The intent of this guidance - that the low temperature overpressure protection (LTOP) system should not depend on the availability of offsite power to perform its function - is applicable to the NuScale design.	5.2.2
BTP 5-2, Rev 3: Overpressurization Protection of Pressurized- Water Reactors While Operating at Low Temperatures	B.9	Analyses Considering Inadvertent System Actuation	Conforms	Applicable	None.	5.2.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
BTP 5-2, Rev 3: Overpressurization Protection of Pressurized- Water Reactors While Operating at Low Temperatures	B.10	Interlocks to Ensure Overpressure Protection	Partially Conforms	Applicable	The intent of this acceptance criterion is applicable but the specific language refers to large LWR designs that provide pressure relief from a low-pressure system not normally connected to the primary system. In the NuScale design, the LTOP system is not connected to a lowpressure system. However, the intent of this guidance - to ensure that the LTOP system is not inadvertently isolated from the primary system - is applicable to the DCA.	5.2.2
BTP 5-3, Rev 2: Fracture Toughness Requirements	1	Preservice Fracture Toughness Test Requirements	Partially Conforms	Applicable	This acceptance criterion is applicable except as indicated in the comments below for Acceptance Criteria 1.1 and 1.2.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	1.1	Determination of RT <sub>NDT</sub> for Vessel Materials	Partially Conforms	Applicable	Portions of this acceptance criterion are applicable only to older plants for which fracture toughness testing on vessel material did not include all tests necessary to determine RTNDT. The remainder of this guidance is applicable to the DCA.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	1.2	Estimation of Charpy V-Notch Upper Shelf Energies	Partially Conforms	Applicable	This guidance is applicable except for reference to subtier NUREG-0744, which applies only to operating reactors that do not meet the minimum fracture toughness acceptance criteria defined in this BTP 5-3.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	1.3	Reporting Requirements	Conforms	Applicable	None.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	2	Operating Limitations for Fracture Toughness	Conforms	Applicable	None.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	2.1	Pressure-Temperature Operating Limitations	Conforms	Applicable	None.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	2.2	Recommended Bases for Operating Limitations	Conforms	Applicable	None.	5.3

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
BTP 5-3, Rev 2: Fracture Toughness Requirements	2.3	Reporting Requirements	Conforms	Applicable	None.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	3	Inservice Surveillance of Fracture Toughness	Partially Conforms	Applicable	This acceptance criterion is applicable except as indicated in the comments below for Acceptance Criteria 3.4 and 3.5.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	3.1	Surveillance Program Requirements	Conforms	Applicable	None.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	3.2	SAR Criteria	Conforms	Applicable	None.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	3.3	Surveillance Test Procedures	Conforms	Applicable	None.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	3.4	Reporting Criteria	Not Applicable	Applicable	This acceptance criterion governs plant-specific reporting criteria that are the responsibility of the COL holder to implement.	Not Applicable
BTP 5-3, Rev 2: Fracture Toughness Requirements	3.5	Technical Specification Changes	Not Applicable	Applicable	This acceptance criterion governs plant-specific activities that are the responsibility of the COL holder to implement.	Not Applicable
BTP 5-3, Rev 2: Fracture Toughness Requirements	4	Pressurized Thermal Shock (PWR Only)	Conforms	Applicable	None.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	4.1	Pressurized Thermal Shock Requirements	Conforms	Applicable	None.	5.3
NSDSRS BTP 5-4, Rev 0: Design Requirements of the Residual Heat Removal System	B.1	Functional Requirements	Conforms	Applicable	None.	5.4.3
NSDSRS BTP 5-4, Rev 0: Design Requirements of the Residual Heat Removal System	B.2	Pressure Relief Requirements	Conforms	Applicable	None.	5.4.3
NSDSRS BTP 5-4, Rev 0: Design Requirements of the Residual Heat Removal System	B.3	Test Requirements	Conforms	Applicable	None.	5.4.3



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS BTP 5-4, Rev 0: Design Requirements of the Residual Heat Removal System	B.4	Operational Procedures	Partially Conforms	Applicable	The procedures governed by this acceptance criterion are largely site-specific and are the responsibility of the COL applicant. Consistent with SRP Section 1.0, Appendix A; RG 1.206, Regulatory Position C.III.4; and ESP/DC/ COL-ISG-015, the DCA contains COL information items, as appropriate, that describe the information that is deferred to the COL applicant referencing the certified design. Notwithstanding the above, NuScale considers this acceptance criterion to be applicable to the DCA to the extent necessary to ensure that the COL applicant can satisfy this guidance.	5.4.3
NSDSRS BTP 5-4, Rev 0: Design Requirements of the Residual Heat Removal System	B.5	Implementation	Conforms	Applicable	None.	5.4.3
SRP 6.1.1, Rev 2: Engineered Safety Features Materials	II.1	Materials and Fabrication	Conforms	Applicable	None.	6.1.1
SRP 6.1.1, Rev 2: Engineered Safety Features Materials	II.1.A	Austenitic Stainless Steels	Conforms	Applicable	None.	6.1.1
SRP 6.1.1, Rev 2: Engineered Safety Features Materials	II.1.B	Ferritic Steel Welding	Conforms	Applicable	None.	6.1.1
SRP 6.1.1, Rev 2: Engineered Safety Features Materials	II.2	Composition and Compatibility of ESF Systems Fluids	Conforms	Applicable	This guidance is applicable except the NuScale design does not provide a method for post-accident pH control as addressed in BTP 6-1.	6.1.1
SRP 6.1.1, Rev 2: Engineered Safety Features Materials	II.3	Component and Systems Cleaning	Partially Conforms	Applicable	RG 1.37 has been withdrawn by the NRC.	6.1.1

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 6.1.1, Rev 2: Engineered Safety Features Materials	II.4	Thermal Insulation	Conforms	Applicable	None.	6.1.1
SRP 6.1.2, Rev 3: Protective Coating Systems (Paints) - Organic Materials	All	Various	Not Applicable	Not Applicable	This SRP section is applicable only to the use of protective coatings on surfaces inside the containment. The NuScale reactor module design as proposed for design certification does not involve the use of protective coatings inside the containment vessel.	6.1.2
NSDSRS 6.2.1, Rev 0: Containment Functional Design	No specific requirements listed.	Applicable acceptance criteria are addressed in SRP 2.4.6, 2.4.10, 2.4.12, 3.9.3, 19.0 and DSRS 3.8.2.	See the applicable SRP or DSRS.	Applicable	See SRP 2.4.6, 2.4.10, 2.4.12, 3.9.3, 19.0, and NSDSRS 3.8.2.	2.4 3.8.2 3.9.3 6.2.1 19.2
NSDSRS 6.2.1.1.A, Rev 0: Containment	II.1	Design Margin for Containment Design Pressure	Conform	Applicable	The peak containment pressure for the limiting event is less than the design pressure (See Section 6.2.1).	6.2.1
NSDSRS 6.2.1.1.A, Rev 0: Containment	II.2	Reducing Containment Pressure Following Postulated Design Basis Accident	Conforms	Applicable	None.	6.2.1
NSDSRS 6.2.1.1.A, Rev 0: Containment	II.3	Containment Heat Removal Capability and Design Margin - LOCA Assumptions	Conforms	Applicable	None.	6.2.1 6.2.2
NSDSRS 6.2.1.1.A, Rev 0: Containment	II.4	Containment Heat Removal Capability and Design Margin - Containment Response Analysis Assumptions	Conforms	Applicable	None.	6.2.1 6.2.2
NSDSRS 6.2.1.1.A, Rev 0: Containment	II.5	Protection of Containment from External Pressure Conditions	Conforms	Applicable	None.	6.2.1
NSDSRS 6.2.1.1.A, Rev 0: Containment	II.6	Containment Monitoring Instrumentation	Conforms	Applicable	None.	6.2.1
NSDSRS 6.2.1.1.A, Rev 0: Containment	II.7	Design of Containment Internal Structures and System Components	Conforms	Applicable	None.	6.2.1
NSDSRS 6.2.1.1.A, Rev 0: Containment	II.8	Evaluation of Accident Involving Generated Hydrogen	Conforms	Applicable	None.	6.2.1

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSDSRS 6.2.1.1.A, Rev 0: Containment	II.9	Evaluation of an Accident on other Modules	Conforms	Applicable	None.	6.2.1
SRP 6.2.1.1.B, Draft Rev 3: Ice Condenser Containments	All	Various	Not Applicable	Not Applicable	This SRP section and its acceptance criteria are applicable only to applicants for plant designs that involve Ice Condenser Containments. The NuScale reactor design does not use an Ice Condenser Containment.	Not Applicable
SRP 6.2.1.1.C, Rev 7: Pressure Suppression Type BWR Containments	All	Various	Not Applicable	Not Applicable	This SRP section and its acceptance criteria are applicable only to applicants for BWR designs that involve Pressure Suppression Type Containments.	Not Applicable
SRP 6.2.1.2, Rev 3: Subcompartment Analysis	All	Various	Not Applicable	Not Applicable	This SRP section and its acceptance criteria (II.1 through II.4) are applicable only to LWR designs that involve a containment structure that houses subcompartments. The NuScale containment vessel design does not involve subcompartments housing high-energy piping as defined in this guidance (or internal compartments as referred to in GDC 50).	Not Applicable
NSDSRS 6.2.1.3, Rev 0: Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)	II.1	Compliance with GDC 50 and 10 CFR 50, Appendix K < paragraph I.A - Sources of Heat during the LOCA	Conforms	Applicable	The energy from metal-water reactions is not included. See Section 6.2.1.	6.2.1
NSDSRS 6.2.1.3, Rev 0: Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)	II.2	Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certification Applications	Conforms	Applicable	None.	6.2.1
NSDSRS 6.2.1.3, Rev 0: Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)	II.3	Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Combined License (COL) Applications	Not Applicable	Applicable	This acceptance criterion is applicable only to COL applicants.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 6.2.1.4, Rev 0: Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	II.1	Sources of Energy	Conforms	Applicable	None.	6.2.1
NSDSRS 6.2.1.4, Rev 0: Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	II.2	Mass and Energy Release Rate	Conforms	Applicable	None.	6.2.1
NSDSRS 6.2.1.4, Rev 0: Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	II.3	Single-Failure Analyses	Conforms	Applicable	None.	6.2.1
SRP 6.2.1.5, Rev 3: Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	All	Containment Pressure Model for ECCS Performance Analysis; Containment Response Analyses Conservatism	Not Applicable	Not Applicable	This SRP section and its acceptance criteria are applicable only to PWRs for which a postulated LOCA results in core uncovery. For such PWR designs, recovery from a postulated LOCA involves a core reflood phase. For the NuScale reactor design, a LOCA does not result in core uncovery.	6.2.1
NSDSRS 6.2.2, Rev 0: Containment Heat Removal Systems	II.1	GDC 5, Sharing of Structures, Systems, and Components	Conforms	Applicable	None.	6.2.2
NSDSRS 6.2.2, Rev 0: Containment Heat Removal Systems	II.2	GDC 38, Containment Heat Removal	Conforms	Applicable	None.	6.2.2
NSDSRS 6.2.2, Rev 0: Containment Heat Removal Systems	II.3	GDC 39, Inspection of Containment Heat Removal System	Conforms	Applicable	None.	6.2.2
NSDSRS 6.2.2, Rev 0: Containment Heat Removal Systems	II.4	GDC 40, Testing of Containment Heat Removal System	Not Applicable	Not Applicable	Section 3.1.4 states the NPM design does not conform to GDC 40 and the design supports an exemption.	6.2.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 6.2.2, Rev 0: Containment Heat Removal Systems	II.5	10 CFR 50.46(b)(5), long-term cooling, including adequate water level (head) margin RRVs), in the presence of LOCA-generated and latent debris	Conforms	Applicable	None.	6.2.2
NSDSRS 6.2.2, Rev 0: Containment Heat Removal Systems	II.6	Compliance with 10 CFR 50.46(b)(5) as it relates to requirements for long-term cooling	Conforms	Applicable	None.	6.2.2
SRP 6.2.3, Rev 3: Secondary Containment Functional Design	All	Various	Not Applicable	Not Applicable	This SRP section and its acceptance criteria (II.1 through II.4) are applicable only to LWR designs that incorporate both a primary and secondary containment. The NuScale containment vessel design does not include a secondary containment.	Not Applicable
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.1	Instrument Line Isolation	Conforms	Applicable	No instrumentation process lines penetrate containment.	6.2.4
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.2	Isolation of and Leak Detection in Lines in Engineered Safety Feature (or Related) Systems	Conforms	Applicable	None.	6.2.4
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.3	Isolation of and Leak Detection in Lines in Systems Needed for Safe Shutdown	Conforms	Applicable	None.	6.2.4
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.4	Containment Isolation Valve Requirements	Conforms	Applicable	None.	6.2.4
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.5	Containment Isolation Valve Requirements for Engineered Safety Feature (or Related) Systems	Conforms	Applicable	None.	6.2.4
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.6	Use of Sealed-Closed Barriers in Place of Automatic Isolation Valves	Conforms	Applicable	None.	6.2.4
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.7	Use of Relief Valves as Isolation Valves	Not Applicable	Applicable	Relief valves are not used as CIVs.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.8	Classification of Essential or NonEssential Systems	Conforms	Applicable	None.	6.2.4
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.9	Location of Isolation Valves Outside Containment	Conforms	Applicable	None.	6.2.4
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.10	Loss of Power to Automatic Isolation Valves	Conforms	Applicable	None.	6.2.4
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.11	Isolation Reliability	Conforms	Applicable	None.	6.2.4
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.12	Parameter Diversity for Initiation of Containment Isolation	Conforms	Applicable	None.	6.2.4
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.13	Radiation Monitors for Initiation of Containment Isolation on Open Paths to the Environs	Conforms	Applicable	Radiation monitoring is not included as a containment isolation signal for any penetration.	6.2.4
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.14	Isolation Valve Closure Times	Conforms	Applicable	None.	6.2.4
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.15	Use of Closed System Inside Containment	Conforms	Applicable	None.	6.2.4
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.16	Specific Design Criteria for Containment Isolation Components	Conforms	Applicable	None.	6.2.4
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.17	Provisions to Allow Control Room Operator Actions	Conforms	Applicable	None.	6.2.4
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.18	Operability and Leakage Rate Testing	Conforms	Applicable	None.	6.2.4
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.19	Reopening of Containment Isolation Valves	Conforms	Applicable	None.	6.2.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.20	Station Blackout	Conforms	Applicable	None.	6.2.4
NSDSRS 6.2.4, Rev 0: Containment Isolation System	II.21	Source Term in Radiological Calculations	Conforms	Applicable	None.	6.2.4
NSDSRS 6.2.5, Rev 0: Combustible Gas Control in Containment	II.1	Analysis of Hydrogen and Oxygen Concentration Control and Distribution in Containment	Partially Conforms	Applicable	Systems to control hydrogen concentrations within containment are not required because combustion has no impact on CNV integrity. NuScale is submitting a request to be exempted from the requirements.	6.2.5
NSDSRS 6.2.5, Rev 0: Combustible Gas Control in Containment	II.2	Equipment Survivability and Containment Structural Integrity	Conforms	Applicable	None.	6.2.5
NSDSRS 6.2.5, Rev 0: Combustible Gas Control in Containment	II.3	Ensuring a Mixed Atmosphere	Conforms	Applicable	None.	6.2.5
NSDSRS 6.2.5, Rev 0: Combustible Gas Control in Containment	II.4	Design Requirements of GDC 41	Partially Conforms	Applicable	The NuScale design does not include a containment atmospheric cleanup system. Containment integrity is assured without systems to control hydrogen and oxygen concentrations within containment.	6.2.5
NSDSRS 6.2.5, Rev 0: Combustible Gas Control in Containment	II.5	Inspection and Test Requirements of GDC 41, GDC 42, and GDC 43	Not Applicable	Not Applicable	The NuScale design does not include a containment atmospheric cleanup system. Containment integrity is assured without systems to control hydrogen and oxygen concentrations within containment.	6.2.5
NSDSRS 6.2.5, Rev 0: Combustible Gas Control in Containment	II.6	Containment Structural Integrity Analysis	Conforms	Applicable	None.	6.2.5
NSDSRS 6.2.6, Rev 0: Containment Leakage Testing	All	Various	Conforms	Applicable	None.	6.2.6

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 6.2.7, Rev 1: Fracture Prevention of Containment Pressure Boundary	All	Various	Conforms	Applicable	None.	6.2.7
NSDSRS 6.3, Rev 0: Emergency Core Cooling System	II.1	ECCS Acceptance Criteria of 10 CFR 50.46	Conforms	Applicable	None.	6.3.1 6.3.3
NSDSRS 6.3, Rev 0: Emergency Core Cooling System	II.2	Single-Failure Consideration	Conforms	Applicable	None.	6.3.1
NSDSRS 6.3, Rev 0: Emergency Core Cooling System	II.3	Inservice Inspection and Operability Testing	Departure	Applicable	None.	6.3.2
NSDSRS 6.3, Rev 0: Emergency Core Cooling System	II.4	Combined Reactivity Control System Capability and Actuation Provisions	Conforms	Applicable	The guidance in this acceptance criterion related to actuation signals is applicable to ECCS actuation. With regard to the requirements of GDC 27, related to the NuScale ECCS does not perform a poison addition safety function nor does it provide any makeup function. NuScale interprets GDC 27 as allowing ECCS poison addition to be credited, but not requiring it. Note that the NuScale design supports an exemption to GDC 27 with regards to reliably controlling reactivity changes and maintaining the capability of cooling the core as described in Section 3.1.	6.3.1
NSDSRS 6.3, Rev 0: Emergency Core Cooling System	II.5	Water Hammer	Conforms	Applicable	None.	6.3.1
NSDSRS 6.3, Rev 0: Emergency Core Cooling System	II.6	Design of Non-Safety-Related Portions of ECCS	Conforms	Applicable	None.	6.3.1
NSDSRS 6.3, Rev 0: Emergency Core Cooling System	II.7	ECCS Interfaces and Shared Systems	Conforms	Applicable	None.	6.3.1



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 6.3, Rev 0: Emergency Core Cooling System	II.8	Long Term Cooling	Conforms	Applicable	None.	6.3.1
NSDSRS 6.3, Rev 0: Emergency Core Cooling System	II.9	ECCS Outage Times and Reports on Unavailability	Conforms	Applicable	None.	6.3.2
NSDSRS 6.3, Rev 0: Emergency Core Cooling System	II.10	Programmatic Requirements	Conforms	Applicable	None.	6.3.1
SRP 6.4, Rev 3: Control Room Habitability System	II.1	Control Room Emergency Zone	Conforms	Applicable	None.	6.4
SRP 6.4, Rev 3: Control Room Habitability System	II.2	Ventilation System Criteria	Conforms	Applicable	None.	6.4
SRP 6.4, Rev 3: Control Room Habitability System	II.3	Pressurization Systems	Conforms	Applicable	None.	6.4
SRP 6.4, Rev 3: Control Room Habitability System	II.4	Emergency Standby Atmosphere Filtration System	Not Applicable	Not Applicable	This guidance is applicable only to reactor designs that rely on emergency filtration for control room habitability during a design basis accident. The NuScale control room habitability system neither relies on nor uses emergency filtration to protect operators during accident conditions. Rather, clean air is provided using compressed air tanks.	Not Applicable
SRP 6.4, Rev 3: Control Room Habitability System	II.5	Relative Location of Source and Control Room	Not Applicable	Not Applicable	This guidance is applicable only to reactor designs that rely on the control room emergency ventilation system for control room habitability during a design basis accident. The NuScale control room habitability system uses compressed air tanks as a clean air source during postulated accident events. This eliminates the potential for radioactive material or toxic gases to enter the control room via ventilation system inlets.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 6.4, Rev 3: Control Room Habitability System	II.6.A	Dose Guidelines for Current Operating Reactors That Do Not Implement an Alternative Source Term	Not Applicable	Applicable	This guidance is applicable only to currently operating reactors.	Not Applicable
SRP 6.4, Rev 3: Control Room Habitability System	II.6.B	Dose Guidelines for New Reactors and Licensees That Implement an Alternative Source Term	Conforms	Applicable	The subtier RG 1.183 is partially applicable.	6.4.1
SRP 6.4, Rev 3: Control Room Habitability System	II.7	Toxic Gas Hazards	Partially Conforms	Applicable	Programmatic requirements are the COL applicant responsibility.	6.4
SRP 6.5.1, Rev 4: ESF Atmosphere Cleanup Systems	II	First full paragraph on Page 6.5.16, Design, Testing, and Maintenance of ESF Atmosphere Cleanup System Air Filtration and Adsorption Units	Not Applicable	Not Applicable	The NuScale Power Plant design does not utilize engineered safety feature (ESF) filter systems or ESF ventilation systems to mitigate the consequences of a design basis accident (DBA). In the NuScale Power Plant design there is a nonsafety-related Reactor Building heating ventilating and air conditioning (HVAC) system which includes filtering; however, it is not credited in the dose analysis.	Not Applicable
SRP 6.5.2, Rev 4: Containment Spray as a Fission Product Cleanup System	All	Various	Not Applicable	Not Applicable	This SRP section and its acceptance criteria (II.1 through II.3) are applicable only to large LWRs with containment spray systems. The NuScale containment vessel design does not incorporate a spray system.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 6.5.3, Rev 3: Fission Product Control Systems and Structures	II.1	Primary Containment	Partially Conforms	Applicable	A portion of this acceptance criterion and its subtier guidance is applicable only to LWR designs that include containment fission product clean-up systems. The NuScale containment vessel does not contain fission product clean-up systems, nor does it include or require pressure suppression systems (e.g., suppression pools or active containment heat removal systems such as containment spray) that serve a fission product removal/dose mitigation function. Rather, fission product control is inherent in the passive design of the NuScale reactor module, wherein the compact containment vessel is submerged in the reactor pool. Therefore, the aspects of this guidance related to these systems are not applicable to the DCA. This guidance is applicable to the review of certain NuScale containment parameters and design features, such as design leakage rate and systems leakage prior to containment isolation.	6.5.3
SRP 6.5.3, Rev 3: Fission Product Control Systems and Structures	II.2	Secondary Containment	Not Applicable	Not Applicable	This acceptance criterion is applicable only to LWRs that incorporate both a primary and secondary containment. The NuScale containment vessel design does not include a secondary containment.	Not Applicable
SRP 6.5.3, Rev 3: Fission Product Control Systems and Structures	II.4	Other Fission Product Control Systems	Not Applicable	Not Applicable	The only credited ESF fission product control system in the NuScale Power Plant design is the containment vessel (CNV) in conjunction with the containment isolation valves and passive containment isolation barriers.	1.2.1 6.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 6.5.4, Draft Rev 4: Ice Condenser as a Fission Product Cleanup System	All	Various	Not Applicable	Not Applicable	This SRP section and its acceptance criteria are applicable only to applicants for plant designs that involve Ice Condenser Containments. The NuScale reactor design does not use an Ice Condenser Containment.	Not Applicable
SRP 6.5.5, Rev 1: Pressure Suppression Pool as a Fission Product Cleanup System	All	Various	Not Applicable	Not Applicable	This SRP section and its acceptance criteria are applicable only to large LWRs that credit a Pressure Suppression Pool for fission product scrubbing and retention (i.e., BWRs). The NuScale reactor design does not credit or use a Suppression Pool.	Not Applicable
NSDSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.1	Components Subject to Inspection	Conforms	Applicable	None.	6.6.1
NSDSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.2	Accessibility	Conforms	Applicable	None.	6.6.2
NSDSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.3	Examination Categories and Methods	Conforms	Applicable	None.	6.6.3
NSDSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.4	Inspection Intervals	Conforms	Applicable	None.	6.6.4
NSDSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.5	Evaluation of Examination Results	Conforms	Applicable	None.	6.6.5
NSDSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.6	System Pressure Tests	Conforms	Applicable	None.	6.6.7

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev; Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 6.6, Rev. 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.7	Structural Supports	Conforms	Applicable	None.	6.6.1 6.6.5 Table 6.6-1
NSDSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.8	Augmented ISI to Protect Against Postulated Piping Failures	Conforms	Applicable	None.	6.6.8
NSDSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.9	Code Exemptions	Conforms	Applicable	None.	6.6
NSDSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.10	Relief Requests	Conforms	Applicable	None.	6.6
NSDSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.11	Code Cases	Conforms	Applicable	None.	6.6
NSDSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.12	Operational Programs	Not Applicable	Applicable	The operational program and implementation milestones governed by this acceptance criterion are site- specific and are the responsibility of the COL applicant. Consistent with SRP Section 1.0, Appendix A; RG 1.206, Regulatory Position C.III.4; and ESP/DC/ COL-ISG-015, the DCA contains COL information items that describe the information that is deferred to the COL applicant referencing the certified design.	Not Applicable
NSDSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.13	Risk Informed ISI Program	Not Applicable	Not Applicable	NuScale is not implementing a Risk Informed ISI Program.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 6.7, Draft Rev 3: Main Steam Isolation Valve Leakage Control System (BWR)	All	Various	Not Applicable	Not Applicable	This SRP section and its acceptance criteria are applicable only to BWRs.	Not Applicable
BTP 6-1, (March 2007): pH for Emergency Coolant Water for Pressurized Water Reactors	B.1	Minimum pH for Emergency Coolant Water	Conforms	Applicable	This acceptance criterion is applicable but certain language in BTP 6-1, which would be applied by Acceptance Criterion B.1, refers to SSCs that are not in the NuScale design. Specifically, the NuScale design does not use a containment spray system or a sump. However, during ECCS operation, ECCS water does collect inside the NuScale containment vessel for recirculation back to the reactor core, and thus the intent of this acceptance criterion is applicable to the DCA.	6.1.1
BTP 6-1, (March 2007): pH for Emergency Coolant Water for Pressurized Water Reactors	B.2	Spray Water pH and Water Chemistry Requirements for Fission Product Removal	Partially Conforms	Applicable	The intent of a portion of this acceptance criterion is applicable but the specific language refers to SSCs that are not in the NuScale design. Specifically the NuScale design does not use a containment spray system or a sump. However, during ECCS operation, ECCS water does collect inside the NuScale containment vessel for recirculation back to the reactor core, and thus the pH guideline contained in this acceptance criterion is applicable to the DCA.	

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
BTP 6-1, (March 2007): pH for Emergency Coolant Water for Pressurized Water Reactors	B.3	Hydrogen Generation from Aluminum Corrosion	Conforms	Applicable	This acceptance criterion is applicable but certain language in BTP 6-1, which would be applied by Acceptance Criterion B.3, refers to SSCs that are not in the NuScale design. Specifically, the NuScale design does not use a containment spray system or a sump. However, during ECCS operation, ECCS water does collect inside the NuScale containment vessel for recirculation back to the reactor core, and thus the intent of this acceptance criterion is applicable to the DCA.	6.3.2
BTP 6-2, Rev 3: Minimum Containment Pressure Model for PWR ECCS Performance	All (B.1 thru B.3)	Various	Not Applicable	Not Applicable	This guidance is applicable only to PWRs for which a postulated LOCA results in core uncover. For such PWR designs, recovery from a postulated LOCA involves a core reflood phase. For the NuScale reactor design, a LOCA does not result in core uncover.	Not Applicable
BTP 6-3, Rev 3: Determination of Bypass Leakage Paths in Dual Containment Plants	All	Various	Not Applicable	Not Applicable	These acceptance criteria (B.1 through B.9) are applicable only to large LWRs that incorporate both a primary and secondary containment. The NuScale containment vessel design does not include a secondary containment.	Not Applicable
BTP 6-4, Rev 3: Containment Purging During Normal Plant Operations	All (B.1 thru B.5)	Various	Not Applicable	Not Applicable	This guidance pertains to containment purge systems used to vent containment directly to the environs. While the NuScale containment vessel design includes an evacuation system, it serves a different purpose than a purge system, and does not provide an open path to the environs. (The NuScale containment vessel evacuation system valve closure times are addressed under SRP Section 6.2.4.)	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
BTP 6-5, Rev 3: Currently the Responsibility of Reactor Systems Piping From the RWST (or BWST) and Containment Sump(s) to the Safety Injection Pumps	All	Various	Not Applicable	Not Applicable	This guidance is applicable only to LWR ECCS designs that rely on safety injection pumps and refueling (or borated) water storage tanks. The NuScale ECCS design does not use pumps or refueling water storage tanks (or equivalent).	Not Applicable
NSDSRS 7.0, Rev 0: Instrumentation and Controls - Introduction and Overview of Review Process	All	Various	Conforms	Applicable	This NSDSRS section provides a general description of the process for reviewing I&C systems that is applicable to the DCA. However, this guidance does not contain specific acceptance criteria. Rather, specific acceptance criteria for SRP Chapter 7 are provided in the individual SRP Chapter 7 sections, and are summarized in SRP Section 7.1, SRP Table 7-1, and SRP Appendix 7.1-A.	7.0
NSDSRS Appendix 7.0-A, Rev 0: I&C - Hazard Analysis	-	I&C - Hazard Analysis	Conforms	Applicable	None.	7.1.7
NSDSRS Appendix 7.0-B, Rev 0: I&C - System Architecture	-	I&C - System Architecture	Conforms	Applicable	None.	7.0.4
NSDSRS Appendix 7.0-C, Rev 0: I&C - Simplicity	-	I&C - Simplicity	Conforms	Applicable	None.	7.1.6
NSDSRS Appendix 7.0-D, Rev 0: I&C - References	-	References	-	-	None.	-
NSDSRS 7.1.1, Rev 0: Fundamental Design Principals	All	Specific SRP Acceptance Criteria Applicable to I&C Systems Important to Safety are Listed in Table 7.0-1.	Conforms	Applicable	None.	7.1.1
NSDSRS 7.1.2, Rev 0: Independence	II.1	Ensure compliance to current version of RG 1.75	Conforms	Applicable	RG 1.75 endorses IEEE Std 384-1992, Standard Criteria for Independence of Class 1E Equipment and Circuits, with identified exceptions and clarifications.	7.1.2
NSDSRS 7.1.2, Rev 0: Independence	II.2	Ensure compliance to current version of RG 1.152	Conforms	Applicable	None.	7.1.2



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 7.1.3, Rev 0: Redundancy	-	Conformance with RG 1.53	Conforms	Applicable	None.	7.1.3
NSDSRS 7.1.4, Rev 0: Predictability and Repeatability	-	Predictability and Repeatability	Conforms	Applicable	There are no specific NSDSRS acceptance criteria in this section. However, the guidance provided below should be used to review the acceptability of the I&C system output predictability and repeatability.	7.1.4
NSDSRS 7.1.5, Rev 0: Diversity and Defense in Depth	II.1	Methods for performing D3 analyses of reactor protection systems	Conforms	Applicable	NUREG/CR-6303, Method for Performing D3 Analyses of Reactor Protection Systems, issued December 1994, summarizes several D3 analyses performed after 1990 and presents an acceptable method for performing such analyses.	7.1.5
NSDSRS 7.1.5, Rev 0: Diversity and Defense in Depth	II.2	SECY-93-087	Conforms	Applicable	The SRM for SECY-93 087 describes the NRC position on defense-in-depth in Item 18.II.Q.	7.1.5
NSDSRS 7.1.5, Rev 0: Diversity and Defense in Depth	II.3	GL 85-06	Conforms	Applicable	Generic Letter (GL) 85-06, Quality Assurance Guidance for ATWS Equipment That Is Not Safety-Related, dated April 16, 1985, provides quality assurance guidance for nonsafety-related ATWS equipment.	7.1.5
NSDSRS 7.1.5, Rev 0: Diversity and Defense in Depth	II.4	Conformance to RG 1.53	Conforms	Applicable	None.	7.1.5
NSDSRS 7.1.5, Rev 0: Diversity and Defense in Depth	II.5	Conformance to RG 1.62	Conforms	Applicable	See RG 1.62 in Table 1.9-2.	7.1.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 7.1.5, Rev 0: Diversity and Defense in Depth	II.6	Conformance to IEEE Std. 7-4.3.2	Conforms	Applicable	IEEE Std. 7-4.3.2 provides guidance on performing an engineering evaluation of software CCF for digital-based systems, including use of manual action and nonsafety- related systems, or components, or both, to provide means to accomplish the function that would otherwise be defeated by the CCF.	7.1.5
NSDSRS 7.2.1 Rev. 0: Quality	II.1	Conformance to RG 1.28	Conforms	Applicable	See RG 1.28 in Table 1.9-2.	7.2.1
NSDSRS 7.2.1 Rev. 0: Quality	II.2	Conformance to RG 1.152	Conforms	Applicable	See RG 1.152 in Table 1.9-2.	7.2.1
NSDSRS 7.2.1 Rev. 0: Quality	II.3	Conformance to RG 1.168	Partially Conforms	Applicable	See RG 1.168 in Table 1.9-2.	7.2.1
NSDSRS 7.2.1 Rev. 0: Quality	II.4	Conformance to RG 1.169	Partially Conforms	Applicable	See RG 1.169 in Table 1.9-2.	7.2.1
NSDSRS 7.2.1 Rev. 0: Quality	II.5	Conformance to RG 1.170	Partially Conforms	Applicable	See RG 1.170 in Table 1.9-2.	7.2.1
NSDSRS 7.2.1 Rev. 0: Quality	II.6	Conformance to RG 1.171	Partially Conforms	Applicable	See RG 1.171 in Table 1.9-2.	7.2.1
NSDSRS 7.2.1 Rev. 0: Quality	II.7	Conformance to RG 1.172	Partially Conforms	Applicable	See RG 1.172 in Table 1.9-2.	7.2.1
NSDSRS 7.2.1 Rev. 0: Quality	II.8	Conformance to RG 1.173	Partially Conforms	Applicable	See RG 1.173 in Table 1.9-2.	7.2.1
NSDSRS 7.2.2, Rev 0: Equipment Qualification	II.1	Conformance to IEEE Std 7- 4.3.2	Conforms	Applicable	Digital I&C safety systems conform to the guidance in Section 5.4 of IEEE Std 7- 4.3.2, IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations, as endorsed (with identified exceptions and clarifications) by the version of RG 1.152, Criteria for Use of Computers in Safety Systems of Nuclear Power Plants, in place 6 months before the docket date of the application.	7.2.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 7.2.2, Rev 0: Equipment Qualification	II.2	Conformance to RG 1.209	Conforms	Applicable	None.	7.2.2
NSDSRS 7.2.2, Rev 0: Equipment Qualification	II.3	Conformance to RG 1.151	Partially Conforms	Applicable	See RG 1.151 in Table 1.9-2.	7.2.2
NSDSRS 7.2.2, Rev 0: Equipment Qualification	II.4	Conformance to RG 1.180	Partially Conforms	Applicable	See RG 1.180 in Table 1.9-2.	7.2.2
NSDSRS 7.2.2, Rev 0: Equipment Qualification	II.5	Conformance to RG 1.204	Partially Conforms	Applicable	The safety systems conform to the guidance in the version of RG 1.204, Guidelines for Lightning Protection of Nuclear Power Plants, in place 6 months before the docket date of the application. See RG 1.204 in Table 1.9-2.	7.2.2
NSDSRS 7.2.3, Rev 0: Reliability, Integrity, and Completion of Protective Action	II.1	Conformance to IEEE Std 7-4.3.2	Conforms	Applicable	Digital I&C safety systems conform to the reliability, integrity, and completion of protective action guidance in Sections 5.5, and 5.15 of IEEE Std 7-4.3.2, as endorsed by the version of RG 1.152 in place 6 months before the docket date of the application.	7.2.3
NSDSRS 7.2.4, Rev 0: Operating and Maintenance Bypasses	II.1	Conformance to RG 1.47	Conforms	Applicable	None.	7.2.4
NSDSRS 7.2.5, Rev 0: Interlocks	II.1	Conformance to IEEE Std 7-4.3.2	Conforms	Applicable	For computer-based interlocks, the components and system conform to the guidance for digital computers in IEEE Std 7-4.3.2, as endorsed (with identified exceptions and clarifications) by the version of RG 1.152 in place 6 months before the docket date of the application.	7.2.5
NSDSRS 7.2.6, Rev 0: Derivation of System Inputs	All	Various	Conforms	Applicable	There are no specific NSDSRS acceptance criteria in this section. However, the guidance provided below will be used to review the acceptability of information associated with derivation of system inputs.	7.2.6
NSDSRS 7.2.7, Rev 0: Setpoints	II.1	Conformance to RG 1.105	Partially Conforms	Applicable	See RG 1.105 in Table 1.9-2.	7.2.7

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 7.2.7, Rev 0: Setpoints	II.2	NRC Regulatory Issue Summary (RIS) 2006-17	Conforms	Applicable	NRC Regulatory Issue Summary (RIS) 2006-17, NRC Staff Position on the Requirements of 10 CFR 50.36, Technical Specifications, Regarding Limiting Safety System Settings during Periodic Testing and Calibration of Instrument Channels, discusses issues that could occur during testing of LSSSs and which therefore, may have an adverse effect on equipment operability.	7.2.7
NSDSRS 7.2.7, Rev 0: Setpoints	II.3	Generic Letter (GL) 91-04	Conforms	Applicable	Generic Letter (GL) 91-04, Guidance on Preparation of a Licensee Amendment Request for Changes in Surveillance Intervals to accommodate a 24-Month Fuel Cycle, provides guidance on issues that should be addressed by the setpoint analysis when calibration intervals are extended from 12 or 18 to 24 months.	7.2.7
NSDSRS 7.2.8, Rev 0: Auxiliary Features	All	Various	Conforms	Applicable	There are no specific NSDSRS acceptance criteria in this section. However, the guidance provided below will be used to review the acceptability of information associated with auxiliary features.	7.2.8
NSDSRS 7.2.9, Rev 0: Control of Access, Identification, and Repair	II.1	Conformance to IEEE Std 7-4.3.2	Conforms	Applicable	Digital I&C safety systems and components conform to the identification guidance in Section 5.11 of IEEE Std 7-4.3.2, as endorsed by the version of RG 1.152 in place 6 months before the docket date of the application.	7.2.9
NSDSRS 7.2.9, Rev 0: Control of Access, Identification, and Repair	II.2	Conformance to RG 1.75	Conforms	Applicable	None.	7.2.9

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 7.2.10, Rev 0: Interaction Between Sense and Command Features and Other Systems	All	Varies	Conforms	Applicable	There are no specific NSDSRS acceptance criteria in this section. However, the guidance provided is used to review the acceptability of the information associated with interaction between sense and command features and other systems.	7.2.10
NSDSRS 7.2.11, Rev 0: Multi-Unit Stations	II.1	Conformance to RG 1.53	Conforms	Applicable	None.	7.2.11
NSDSRS 7.2.12, Rev 0: Automatic and Manual Controls	II.1	Conformance to RG 1.62	Conforms	Applicable	See RG 1.62 in Table 1.9-2.	7.2.12
NSDSRS 7.2.13, Rev 0: Displays and Monitoring	II.1	Conformance to RG 1.97	Partially Conforms	Applicable	See RG 1.97 in Table 1.9-2.	7.2.13
NSDSRS 7.2.13, Rev 0: Displays and Monitoring	II.2	Conformance to RG 1.47	Conforms	Applicable	None.	7.2.13
NSDSRS 7.2.13, Rev 0: Displays and Monitoring	II.3	SECY-93-087	Conforms	Applicable	The SRM on SECY-93-087, Item II.T, Control Room Annunciator Alarm Reliability, provides general guidance on the alarm system interface with operator workstations.	7.2.13
NSDSRS 7.2.14, Rev 0: Human Factors Considerations	All	Various	Conforms	Applicable	There are no specific NSDSRS acceptance criteria in this section. However, the guidance provided is used to review the acceptability of information associated with human factors considerations.	7.2.14
NSDSRS 7.2.15, Rev 0: Capability for Test and Calibration	II.1	Conformance to IEEE Std 7-4.3.2	Conforms	Applicable	Digital I&C safety systems and components conform to the guidance related to capability for test and calibration in Sections 5.7, 5.5.2, and 5.5.3 of IEEE Std 7-4.3.2, as endorsed by the version of RG 1.152 in place 6 months before the docket date of the application.	7.2.15
NSDSRS 7.2.15, Rev 0: Capability for Test and Calibration	II.2	Conformance to RG 1.118	Partially Conforms	Applicable	See RG 1.118 in Table 1.9-2.	7.2.15

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 8.1, Rev 0: Electric Power - Introduction	II (No Number)	Specific SRP Acceptance Criteria Contained in SRP Sections 8.2, 8.3.1, 8.3.2, and 8.4 (summarized in Table 8-1)	Partially Conforms	Applicable	NSDSRS Table 8-1 provides a matrix of the NRC requirements, guidance, and Commission policy documents, and industry codes and standards that are applied as acceptance criteria and guidance to the review of the electrical systems described in Sections 8.2, 8.3.1, 8.3.2, and 8.4. Some of these documents are not relevant or are only partially relevant to the NuScale design.	8.1.4 8.2.2 8.3.1 8.3.2 8.4
NSDSRS 8.2, Rev 0: Offsite Power System	II.1	Compliance with GDC 5	Not Applicable	Applicable	Conformance with GDC 5 is the responsibility of the COL applicant as described in Section 8.2.2.	8.2.2
NSDSRS 8.2, Rev 0: Offsite Power System	II.2	Compliance with GDC 17	Departure	Departure	The NuScale design supports an exemption from GDC 17 that includes the associated requirements for the offsite power system.	8.2.3
NSDSRS 8.2, Rev 0: Offsite Power System	II.3	Compliance with GDC 18	Departure	Departure	The NuScale design supports an exemption from GDC 18 that includes the associated requirements for the offsite power system.	8.2.3
NSDSRS 8.2, Rev 0: Offsite Power System	II.4	Compliance with GDC 33	Departure	Departure	The NuScale design supports an exemption from GDC 33.	8.2.3
NSDSRS 8.2, Rev 0: Offsite Power System	II.4	Compliance with GDCs 34, 35, 38, 41, and 44	Departure	Departure	NuScale complies with a set of principal design criteria in lieu of these GDC.	8.2.3
NSDSRS 8.2, Rev 0: Offsite Power System	II.5	Compliance with 10 CFR 50.63 - Passive Design	Conforms	Applicable	The details regarding conformance with 10 CFR 50.63 are described in Section 8.4, Station Blackout.	8.2.3 8.4
NSDSRS 8.2, Rev 0: Offsite Power System	II.6	Compliance with 10 CFR 50.65(a)(4)	Not Applicable	Applicable	As indicated in Section 17.6, the development of the maintenance rule (10 CFR 50.65) program is the responsibility of the COL applicant referencing the certified design.	8.2.3
NSDSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.1	Compliance with GDC 2	Conforms	Applicable	Onsite AC power systems conform to GDC 2 to the extent described in Section 8.3.1.2.1.	8.1.4 8.3.1

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.2	Compliance with GDC 4	Conforms	Applicable	Onsite AC power systems conform to GDC 4 to the extent described in Section 8.3.1.2.2.	8.1.4 8.3.1
NSDSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.3	Compliance with GDC 5	Partially Conforms	Applicable	Onsite AC power systems conform to GDC 5 to the extent described in Section 8.3.1.	8.1.4 8.3.1
NSDSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.4	Compliance with GDC 17	Departure	Departure	The NuScale design supports an exemption from GDC 17 that includes the associated requirements for the onsite AC power system.	8.3.1
NSDSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.5	Compliance with GDC 18	Departure	Departure	The NuScale design supports an exemption from GDC 18 that includes the associated requirements for the onsite AC power system.	8.1.4 8.3.1
NSDSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II (No Number)	Compliance with GDC 33	Departure	Departure	The NuScale design supports an exemption from GDC 33.	8.3.1
NSDSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II (No Number)	Compliance with GDCs 34, 35, 38, 41, and 44	Departure	Departure	NuScale complies with a set of principal design criteria in lieu of these GDC.	8.3.1
NSDSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.6	Compliance with GDC 50	Conforms	Applicable	The AC and DC electrical power system circuits do not penetrate the containment vessels; therefore, conformance with GDC 50 is not necessary.	8.3.1
NSDSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.7	Compliance with 10 CFR 50.65(a)(4)	Not Applicable	Applicable	The 10 CFR 50.65(a)(4) assessment is applied to main onsite AC power system SSCs that: (1) are determined to meet the 10 CFR 50.65(b) criteria; and (2) a risk-informed evaluation process has shown to be significant to public health and safety. As indicated in Section 17.6, the development of the maintenance rule (10 CFR 50.65) program - including the identification of SSCs that require assessment per 10 CFR 50.65(a)(4) - is the responsibility of the COL applicant referencing the certified design.	8.3.1

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.8	Compliance with 10 CFR 50.55a(h)	Not Applicable	Not Applicable	No onsite electrical AC power system equipment is required to conform to 10CFR50.55a(h) and IEEE Std.603-1991.	8.1.1 8.3.1
NSDSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.9	Compliance with 10 CFR 52.47(b)(1)	Conforms	Applicable	None.	8.1
NSDSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 2	Conforms	Applicable	The onsite DC power systems conform to GDC 2 to the extent described in Section 8.3.2.2.2.	8.3.2
NSDSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 4	Conforms	Applicable	The onsite DC power systems conform to GDC 4 to the extent described in Section 8.3.2.2.2.	8.3.2
NSDSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 5	Conforms	Applicable	The onsite DC power systems conform to GDC 5 to the extent described in Section 8.3.2.2.2.	8.3.2
NSDSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 17	Departure	Departure	The NuScale design supports an exemption from GDC 17 that includes the associated requirements for the onsite DC power systems.	8.3.2
NSDSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 18	Departure	Departure	The NuScale design supports an exemption from GDC 18 that includes the associated requirements for the onsite DC power systems.	8.3.2
NSDSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 33	Departure	Departure	The NuScale design supports an exemption from GDC 33.	8.3.2
NSDSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 34, 35, 38, 41, and 44	Departure	Departure	Nuscale complies with a set of principal design criteria in lieu of these GDC.	8.3.2
NSDSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 50	Conforms	Applicable	The DC electrical power system circuits do not penetrate the containment vessels. Therefore conformance with GDC 50 is not necessary.	8.3.2
NSDSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.1	Conformance with RG 1.32	Partially Conforms	Applicable	See RG 1.32 in Table 1.9-2.	8.3.2
NSDSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.2	Conformance with RG 1.75	Conforms	Applicable	See RG 1.75 in Table 1.9-2.	8.3.2
NSDSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.3	Conformance with RG 1.81	Partially Conforms	Applicable	See RG 1.81 in Table 1.9-2.	8.3.2



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.4	Conformance with RG 1.118	Partially Conforms	Applicable	See RG 1.118 in Table 1.9-2.	8.3.2
NSDSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.5	Conformance with RG 1.153	Conforms	Applicable	See RG 1.153 in Table 1.9-2.	8.3.2
NSDSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.6	Conformance with RG 1.153	Conforms	Applicable	See RG 1.153 in Table 1.9-2.	8.3.2
NSDSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.7	Conformance with RG 1.63	Not Applicable	Not Applicable	See RG 1.63 in Table 1.9-2.	8.1.4 8.3.2
NSDSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.8	Conformance with RG 1.160	Not Applicable	Applicable	RG 1.160 provides guidance for implementing the maintenance rule requirements of 10 CFR 50.65. As indicated in Section 17.6, the development of the maintenance rule (10 CFR 50.65) program - including the identification of SSC that require assessment per 10 CFR 50.65(a)(4) - is the responsibility of the COL applicant referencing the certified design.	8.3.2
NSDSRS 8.4, Rev 0: Station Blackout	II.1	Compliance with 10 CFR 50.63 and the guidelines of RG 1.155	Partially Conforms	Applicable	The details regarding conformance with 10 CFR 50.63 and the guidelines of RG 1.155 as they pertain to passive designs are described in Section 8.4, Station Blackout.	8.4
NSDSRS 8.4, Rev 0: Station Blackout	II.2	Use of Alternate AC Power Sources and RTNSS for Plants of Passive Design	Partially Conforms	Applicable	As described in Section 8.4, all safety-related functions can be performed without reliance on AC power for 72 hours after an SBO event, and as described in Section 19.3 a RTNSS process has been implemented. Consequently, the Alternate AC Power Source is not applicable to the NuScale design.	8.4 19.3
NSDSRS 8.4, Rev 0: Station Blackout	II.3	Independence of SBO-related power sources	Partially Conforms	Applicable	Although DC power supplies are not required to meet the SBO mitigation requirements of 10 CFR 50.63, the independence of SBO related power supplies (EDSS) is described in Section 8.3.	8.3.2 8.4.3

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP Appendix 8-A, Rev1: General Agenda, Station Site Visits	All	Various	Not Applicable	Not Applicable	This SRP appendix governs staff visits to plant sites as part of licensing reviews during the operating or COL stage.	Not Applicable
SRP BTP 8-1, Rev 3: Requirements on Motor- Operated Valves in the ECCS Accumulator Lines	All (B.1 thru B.4)	Various	Not Applicable	Not Applicable	The NuScale reactor design does not use safety injection tanks (or equivalent) in response to a design basis accident. Design and operation of the NuScale ECCS also do not involve motor-operated valves.	Not Applicable
SRP BTP 8-2, Rev 3: Use of Diesel-Generator Sets for Peaking	B.	Use of Onsite Emergency Power Diesel-Generator Sets for Purposes Other Than Supplying Standby Power is Prohibited	Conforms	Applicable	The backup diesel generators are used only for supplying standby power to designated loads when needed, and are not interconnected with other AC power sources except for short periods for the purpose of load testing.	8.1.1 8.3.1
SRP BTP 8-3, Rev 3: Stability of Offsite Power Systems	B.1	Grid Reliability	Not Applicable	Applicable	The analysis of grid stability is the responsibility of the COL applicant that references the NuScale design certification as described in Section 8.2.3.	8.2.3
SRP BTP 8-3, Rev 3: Stability of Offsite Power Systems	B.2	Grid Capacity	Not Applicable	Applicable	The analysis of grid stability is the responsibility of the COL applicant that references the NuScale design certification as described in Section 8.2.3.	8.2.3
SRP BTP 8-4, Rev 3: Application of the Single Failure Criterion to Manually Controlled Electrically Operated Valves	All (B.1 through B.5)	Various	Not Applicable	Not Applicable	BTP8-4 establishes the acceptability of disconnecting power to electrical components of a fluid system as one means of designing against a single failure that might cause an undesirable component action. Removal of electric power from safety-related valves is not used in the design as a means of satisfying the single failure criterion.	8.1.1 8.3.1

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP BTP 8-5, Rev 3: Supplemental Guidance for Bypass and Inoperable Status Indication for Engineered Safety Features Systems	All (B.1 thru B.6)	Design Criteria Reflecting Importance of Providing Accurate Information to the Operator and Reducing the Possibility of Adversely Affecting Monitored Safety Systems	Conforms	Applicable	None.	8.1.1 8.3.1 8.3.2
SRP BTP 8-6, Rev 3: Adequacy of Station Electric Distribution System Voltages (for Passive Design)	B.1	Isolation of important to safety loads from degraded grid conditions	Not Applicable	Not Applicable	The undervoltage provisions contained in BTP 8-6 are not relevant to the NuScale plant design since a loss of voltage or degraded voltage condition on the offsite power system would have no reasonable likelihood of adversely affecting the performance of plant safety functions.	8.1.4 8.3.1
SRP BTP 8-6, Rev 3: Adequacy of Station Electric Distribution System Voltages (for Passive Design)	B.2	Monitoring of degraded grid conditions	Not Applicable	Not Applicable	The undervoltage provisions contained in BTP 8-6 are not relevant to the NuScale plant design since a loss of voltage or degraded voltage condition on the offsite power system would have no reasonable likelihood of adversely affecting the performance of plant safety functions.	8.1.4 8.3.1
SRP BTP 8-6, Rev 3: Adequacy of Station Electric Distribution System Voltages (for Passive Design)	B.3	Actions upon detecting a degraded grid condition	Not Applicable	Not Applicable	The undervoltage provisions contained in BTP 8-6 are not relevant to the NuScale plant design since a loss of voltage or degraded voltage condition on the offsite power system would have no reasonable likelihood of adversely affecting the performance of plant safety functions.	8.1.4 8.3.1
SRP BTP 8-7, Rev 3: Criteria for Alarms and Indications Associated with Diesel- Generator Unit Bypassed and Inoperable Status	All	Design Criteria Reflecting Importance of Providing Accurate Information to the Operator and Reducing the Possibility of Adversely Affecting Monitored Safety Systems	Not Applicable	Not Applicable	The NuScale plant does not require or include safety related emergency diesel generators.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP BTP 8-8, (Feb 2012): Onsite (Emergency Diesel Generators) and Offsite Power Sources Allowed Outage Time Extensions	All	Various	Not Applicable	Not Applicable	With the nonreliance on AC power for safety-related functions, the operating restrictions (i.e., Technical Specifications Allowed Outage Times) for inoperable AC power sources specified in this guidance are not appropriate to apply.	Not Applicable
SRP BTP 8-9, Rev. 0: Open Phase Conditions in Electric Power System	B.1	Electrical system design to address open phase condition	Partially Conforms	Applicable	Compliance with BTP 8-9 is described in Section 8.2.3.	8.2.3
SRP BTP 8-9, Rev. 0: Open Phase Conditions in Electric Power System	B.2	Criteria for evaluating open phase conditions for active plant designs	Not Applicable	Not Applicable	Not applicable to passive plant designs.	8.2.1, 8.2.3
SRP BTP 8-9, Rev. 0: Open Phase Conditions in Electric Power System	B.3	Criteria for evaluating open phase conditions for passive plant designs	Partially Conforms	Applicable	Compliance with BTP 8-9 is described in Section 8.2.3.	8.2.3
SRP 9.1.1, Rev 3: Criticality Safety of Fresh and Spent Fuel Storage and Handling	II.1	Specific Criteria to Meet GDC 62	Conforms	Applicable	None.	9.1.1.3
NSDSRS 9.1.2, Rev 0: New and Spent Fuel Storage	II.1	Specific Criteria to Meet GDC 2	Conforms	Applicable	None.	9.1.2.1 9.1.2.3
NSDSRS 9.1.2, Rev 0: New and Spent Fuel Storage	II.2	Specific Criteria to Meet GDC 4	Conforms	Applicable	None.	9.1.2.1 9.1.2.3
NSDSRS 9.1.2, Rev 0: New and Spent Fuel Storage	II.3	Specific Criteria to Meet GDC 5	Conforms	Applicable	None.	9.1.2.1 9.1.2.3
NSDSRS 9.1.2, Rev 0: New and Spent Fuel Storage	II.4	Specific Criteria to Meet GDC 61	Conforms	Applicable	An ESF ventilation system is not required (see RG 1.52).	9.1.2.1 9.1.2.3
NSDSRS 9.1.2, Rev 0: New and Spent Fuel Storage	II.5	Specific Criteria to Meet GDC 63	Conforms	Applicable	None.	9.1.2.1 9.1.2.3 9.1.2.5
NSDSRS 9.1.2, Rev 0: New and Spent Fuel Storage	II.6	Specific Criteria to Meet 10 CFR 20.1101(b)	Conforms	Applicable	None.	9.1.2.1 9.1.2.2 9.1.2.3
NSDSRS 9.1.2, Rev 0: New and Spent Fuel Storage	II.7	Criticality Monitors and Subcriticality Margin	Conforms	Applicable	None.	9.1.1

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 9.1.3, Rev 0: Spent Fuel Pool Cooling and Cleanup System	II.1	Specific Criteria to Meet GDC 2	Partially Conforms	Applicable	The design conforms except that: (1) The normal makeup water supply system and its source are not seismic Category I and the system is not designed to Quality Group C per RG 1.26. The ultimate heat sink (UHS) system is a seismic Category I supply system and source for spent fuel cooling and shielding for accident conditions. A redundant UHS makeup supply line is designed to Quality Group C and seismic Category I requirements. (2) An ESF ventilation system is not required (see RG 1.52).	9.1.3.1 9.1.3.2 9.1.3.3
NSDSRS 9.1.3, Rev 0: Spent Fuel Pool Cooling and Cleanup System	II.2	Specific Criteria to Meet GDC 4	Partially Conforms	Applicable	This design conforms except that: (1) The normal makeup water supply system and its source are not designed to accommodate the effects of postulated accidents. The UHS system is the supply system and source for spent fuel cooling and shielding that are designed to accommodate the effects of postulated accidents. A redundant UHS makeup supply line is designed to meet GDC 4. (2) An ESF ventilation system is not required (see RG 1.52).	9.1.3.1 9.1.3.3
NSDSRS 9.1.3, Rev 0: Spent Fuel Pool Cooling and Cleanup System	II.3	Specific Criteria to Meet GDC 5	Conforms	Applicable	None.	9.1.3.1
NSDSRS 9.1.3, Rev 0: Spent Fuel Pool Cooling and Cleanup System	II.4	Specific Criteria to Meet GDC 61	Conforms	Applicable	None.	9.1.3.1 9.1.3.2
NSDSRS 9.1.3, Rev 0: Spent Fuel Pool Cooling and Cleanup System	II.5	Specific Criteria to Meet GDC 63	Conforms	Applicable	None.	9.1.3.1 9.1.3.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 9.1.3, Rev 0: Spent Fuel Pool Cooling and Cleanup System	II.6	Specific Criteria to Meet 10 CFR 20.1101(b)	Conforms	Applicable	None.	9.1.3.1 9.1.3.3
NSDSRS 9.1.3, Rev 0: Spent Fuel Pool Cooling and Cleanup System	II.7	ITAAC for Design Certification Applications	Conforms	Applicable	None.	14.3
NSDSRS 9.1.3, Rev 0: Spent Fuel Pool Cooling and Cleanup System	II.8	ITAAC for Combined License Applications	Not Applicable	Applicable	This acceptance criterion is applicable only to COL applicants. Acceptance Criterion II.7 above governs ITAAC development for the DCA.	Not Applicable
SRP 9.1.4, Rev 3: Light Load Handling System (Related to Refueling)	II.1	Specific Criteria to Meet GDC 2	Conforms	Applicable	None.	9.1.4
SRP 9.1.4, Rev 3: Light Load Handling System (Related to Refueling)	II.2	Specific Criteria to Meet GDC 5	Conforms	Applicable	None.	9.1.4
SRP 9.1.4, Rev 3: Light Load Handling System (Related to Refueling)	II.3	Specific Criteria to Meet GDC 61	Conforms	Applicable	None.	9.1.4
SRP 9.1.4, Rev 3: Light Load Handling System (Related to Refueling)	II.4	Specific Criteria to Meet GDC 62	Conforms	Applicable	None.	9.1.4
SRP 9.1.5, Rev 1: Overhead Heavy Load Handling Systems	II.1	Specific Criteria to Meet GDC 1	Conforms	Applicable	None.	9.1.5
SRP 9.1.5, Rev 1: Overhead Heavy Load Handling Systems	II.2	Specific Criteria to Meet GDC 2	Conforms	Applicable	None.	9.1.5
SRP 9.1.5, Rev 1: Overhead Heavy Load Handling Systems	II.3	Specific Criteria to Meet GDC 4	Conforms	Applicable	None.	9.1.5
SRP 9.1.5, Rev 1: Overhead Heavy Load Handling Systems	II.4	Specific Criteria to Meet GDC 5	Conforms	Applicable	None.	9.1.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 9.2.1, Rev 5: Station Service Water System	II.1	Protection Against Natural Phenomena (GDC 2)	Partially Conforms	Applicable	The NuScale site cooling water system does not provide essential cooling to safety-related SSCs and are not considered safety-related or important-to-safety. The applicability of GDC 2 to the NuScale cooling water system reviewed under this acceptance criterion is limited to aspects ensuring that a failure of the nonsafety-related SSC does not result in an adverse effect on a safety-related SSC. For the NuScale design, this is provided by the design and construction of the nonsafety related cooling water systems to meet the provisions of RG 1.29, Regulatory Position C.2.	9.2.7 (Used for Site Cooling Water System (SCWS))
SRP 9.2.1, Rev 5: Station Service Water System	II.2	Environmental and Dynamic Effects (GDC 4)	Partially Conforms	Applicable	The NuScale site cooling water system does not provide essential cooling to safety-related SSCs and are not considered safety-related or risk-significant. The applicability of GDC 4 to the NuScale cooling water system reviewed under this acceptance criterion is limited to aspects ensuring that a failure of the nonsafety-related SSC does not result in an adverse effect on a safety-related SSC.	9.2.7 (Used for Site Cooling Water System (SCWS))
SRP 9.2.1, Rev 5: Station Service Water System	II.3	Sharing of Structures, Systems, and Components (GDC 5)	Conforms	Applicable	The NuScale site cooling water system does not provide essential cooling to safety-related SSCs and are not safety-related or risk-significant. The design and layout of these systems satisfy GDC 5. Specifically, sharing of the site cooling water system between units has no reasonable likelihood of adversely affecting essential SSCs and associated safety functions.	9.2.7 (Used for Site Cooling Water System (SCWS))

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 9.2.1, Rev 5: Station Service Water System	II.4	Cooling Water System (GDC 44)	Not Applicable	Not Applicable	The site cooling water system does not serve a safety-related cooling or accident mitigation function.	Not Applicable
SRP 9.2.1, Rev 5: Station Service Water System	II.5	Cooling Water System Inspection (GDC 45)	Not Applicable	Not Applicable	The site cooling water system does not serve a safety-related cooling or accident mitigation function. Thus, the provisions of GDC 45 (cooling water system inspection) are not applicable to these cooling water systems.	Not Applicable
SRP 9.2.1, Rev 5: Station Service Water System	II.6	Cooling Water System Testing (GDC 46)	Not Applicable	Not Applicable	The site cooling water system does not serve a safety-related cooling or accident mitigation function. Thus, the provisions of GDC 46 (cooling water system testing) are not applicable to these cooling water systems.	Not Applicable
SRP 9.2.2, Rev 4: Reactor Auxiliary Cooling Water System	II.1	Protection Against Natural Phenomena	Partially Conforms	Applicable	For the NuScale design, the system function contemplated by this SRP section is similar to that provided by the reactor component cooling water (RCCW) system. However, unlike a typical reactor auxiliary cooling water system, the NuScale RCCW system does not serve a safety-related cooling (or heat transfer) function, and thus it is not considered to be a safety-related system. Specifically, the NuScale design does not rely on the RCCW system to remove heat from safety-related components.	9.2.2
SRP 9.2.2, Rev 4: Reactor Auxiliary Cooling Water System	II.2	Environmental and Dynamic Effects	Partially Conforms	Applicable	See comment above for Acceptance Criterion II.1.	9.2.2
SRP 9.2.2, Rev 4: Reactor Auxiliary Cooling Water System	II.3	Sharing of Structures, Systems, and Components	Not Applicable	Not Applicable	See comment above for Acceptance Criterion II.1.	Not Applicable
SRP 9.2.2, Rev 4: Reactor Auxiliary Cooling Water System	II.4	Cooling Water System	Not Applicable	Not Applicable	See comment above for Acceptance Criterion II.1.	Not Applicable



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 9.2.2, Rev 4: Reactor Auxiliary Cooling Water System	II.5	Cooling Water System Inspection	Not Applicable	Not Applicable	See comment above for Acceptance Criterion II.1.	Not Applicable
SRP 9.2.2, Rev 4: Reactor Auxiliary Cooling Water System	II.6	Cooling Water System Testing	Not Applicable	Not Applicable	See comment above for Acceptance Criterion II.1.	Not Applicable
SRP 9.2.4, Rev 3: Potable and Sanitary Water Systems	II.1	Control of Releases of Radioactive Materials to the PWSW	Partially Conforms	Applicable	The NuScale Potable and Sanitary Water Systems do not interface with system containing potential radioactivity. The NuScale Potable and Sanitary Water Systems are designed such that failure will not result in flooding or other adverse impacts on essential SSCs.	9.2.4
SRP 9.2.5, Rev 3: Ultimate Heat Sink	II.1	Protection Against Natural Phenomena	Partially Conforms	Applicable	Since RG 1.27 is not applicable to the NuScale design, compliance with GDC 2 is demonstrated by adherence to RG 1.13, Regulatory Positions C.1 and C.2. The NuScale UHS, which provides both spent fuel cooling and containment heat removal, is protected from natural phenomena and site related events by the Seismic Category I RXB structure and with a Seismic Category I emergency makeup line.	9.2.5
SRP 9.2.5, Rev 3: Ultimate Heat Sink	II.2	Sharing of Structures, Systems, and Components	Conforms	Applicable	None.	9.2.5
SRP 9.2.5, Rev 3: Ultimate Heat Sink	II.3	Cooling Water System	Partially Conforms	Applicable	This acceptance criterion is applicable except for aspects related to the use of fiberglass piping (see RG 1.72).	9.2.5
SRP 9.2.5, Rev 3: Ultimate Heat Sink	II.4	Cooling Water System Inspection	Conforms	Applicable	None.	9.2.5
SRP 9.2.5, Rev 3: Ultimate Heat Sink	II.5	Cooling Water System Testing	Conforms	Applicable	None.	9.2.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 9.2.6, Rev 3: Condensate Storage Facilities	II.1	Protection Against Natural Phenomena	Partially Conforms	Applicable	NuScale condensate storage system is neither safety-related nor risk-significant. The condensate storage systems and components are located outside the seismic category I reactor building. The effects of discharging water from a condensate storage facility failure have no reasonable potential to adversely impact the operation of safety-related systems or safe operation of the plant. Consistent with Regulatory Position C.2 of RG 1.29, no portion of the NuScale condensate storage system requires design and construction to withstand the safe-shutdown earthquake.	9.2.6 10.4.7
SRP 9.2.6, Rev 3: Condensate Storage Facilities	II.2	Environmental and Dynamic Effects Design Basis	Partially Conforms	Applicable	See comment above for Acceptance Criterion II.1.	9.2.6 10.4.7
SRP 9.2.6, Rev 3: Condensate Storage Facilities	II.3	Sharing of Structures, Systems, and Components	Partially Conforms	Applicable	Sharing of the condensate storage facilities does not impair the ability of safety-related or risk-significant SSCs to perform their safety functions.	9.2.6 10.4.7
SRP 9.2.6, Rev 3: Condensate Storage Facilities	II.4	Control of Radioactive Releases to the Environment	Partially Conforms	Applicable	See comment above for Acceptance Criterion II.1.	9.2.6 10.4.7
SRP 9.2.6, Rev 3: Condensate Storage Facilities	II.5	10 CFR 20.1406 Compliance	Partially Conforms	Applicable	See comment above for Acceptance Criterion II.1.	9.2.6 10.4.7
SRP 9.2.7, Rev 0: Chilled Water System	II.1	Quality Standards and Records	Not Applicable	Not Applicable	The NuScale CHWS does not perform any safety or containment isolation functions.	9.2.8

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 9.2.7, Rev 0: Chilled Water System	II.2	Protection Against Natural Phenomena	Partially Conforms	Applicable	This criterion is based on RG 1.29 position C.1 for safety-related portions, and position C.2 for nonsafety-related positions. Position C.1 is not applicable since the CHWS is not safety related or risk-significant. The NuScale CHWS complies with position C.2 in that the SSCs whose structural failure could affect the operability of safety-related SSCs are designed as Seismic Category II.	9.2.8
SRP 9.2.7, Rev 0: Chilled Water System	II.3	Environmental and Dynamic Effects	Partially Conforms	Applicable	See comment above for Acceptance Criterion II.2.	9.2.8
SRP 9.2.7, Rev 0: Chilled Water System	II.4	Sharing of Structures, Systems, and Components	Not Applicable	Not Applicable	The NuScale CHWS is a nonsafety-system and does not perform any safety functions.	9.2.8
SRP 9.2.7, Rev 0: Chilled Water System	II.5	Cooling Water System	Not Applicable	Not Applicable	The NuScale CHWS is a nonsafety-system and does not perform any safety functions.	9.2.8
SRP 9.2.7, Rev 0: Chilled Water System	II.6	Cooling Water System Inspection	Not Applicable	Not Applicable	The NuScale CHWS is a nonsafety-system and does not perform any safety functions.	9.2.8
SRP 9.2.7, Rev 0: Chilled Water System	II.7	Cooling Water System Testing	Not Applicable	Not Applicable	The NuScale CHWS is a nonsafety-system and does not perform any safety functions.	9.2.8
SRP 9.2.7, Rev 0: Chilled Water System	II.8	Minimization of Contamination	Conforms	Applicable	The CHWS is designed a closed system and is precluded from connection to other radiological contaminated systems in order to minimize contamination.	9.2.8
SRP 9.3.1, Rev 2: Compressed Air System	II.1	Specific Criteria to Meet GDC 1	Not Applicable	Not Applicable	NuScale compressed air systems are non-safety, non-risk-significant systems.	9.3.1
SRP 9.3.1, Rev 2: Compressed Air System	II.2	Specific Criteria to Meet GDC 2	Not Applicable	Not Applicable	NuScale compressed air systems are non-safety, non-risk-significant systems.	9.3.1

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 9.3.1, Rev 2: Compressed Air System	II.3	Specific Criteria to Meet GDC 5	Not Applicable	Not Applicable	NuScale compressed air systems are non-safety, non-risk-significant systems.	9.3.1
SRP 9.3.1, Rev 2: Compressed Air System	II.4	Specific Criteria to Meet 10 CFR 50.63	Partially Conforms	Applicable	The intent of this acceptance criterion and its subtier guidance - to maintain the ability to withstand and recover from a SBO lasting a specified minimum duration - is applicable. However, much of the specific language refers to reactor plant designs such as large LWRs, and is not relevant to the NuScale plant design. The NuScale plant design meets the intent of this guidance largely in its passive design and associated reduced reliance on AC power in coping with design basis events. Specifically, compressed air is not required to achieve core cooling in the event of a station blackout in the NuScale design.	9.3.1 8.4
SRP 9.3.2, Rev 3: Process and Post-Accident Sampling Systems	II.1	Sampling Capability	Partially Conforms	Applicable	This acceptance criterion is applicable except for aspects that are BWR-specific, or not part of the NuScale design (e.g., refueling water storage tank, pressurizer relief tank, and containment sump).	9.3.2
SRP 9.3.2, Rev 3: Process and Post-Accident Sampling Systems	II.2	10 CFR 20.1406. Minimization of contamination	Conforms	Applicable	None.	9.3.2
SRP 9.3.2, Rev 3: Process and Post-Accident Sampling Systems	II.3	Technical Specifications	Not Applicable	Not Applicable	This was addressed in NRC-approved TSTF 366-A and is no longer applicable.	Not Applicable
SRP 9.3.2, Rev 3: Process and Post-Accident Sampling Systems	II.4	Process Sampling System Functional Design	Conforms	Applicable	None.	9.3.2
SRP 9.3.2, Rev 3: Process and Post-Accident Sampling Systems	II.5	Seismic Design and Quality Group Classification	Conforms	Applicable	None.	9.3.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 9.3.2, Rev 3: Process and Post-Accident Sampling Systems	II.6	ITAAC	Conforms	Applicable	None.	9.3.2
SRP 9.3.3, Rev 3: Equipment and Floor Drainage System	II.1	Protection Against Natural Phenomena	Conforms	Applicable	None.	9.3.3
SRP 9.3.3, Rev 3: Equipment and Floor Drainage System	II.2	Environmental and Dynamic Effects	Conforms	Applicable	None.	9.3.3
SRP 9.3.3, Rev 3: Equipment and Floor Drainage System	II.3	Control of Releases of Radioactive Material to the Environment	Conforms	Applicable	Isolation of the portions of the drain system that penetrate the containment barrier is discussed. There are no portions of the NuScale drain system that penetrate the containment barrier.	9.3.3
NSDSRS 9.3.4, Rev 0: Chemical and Volume Control System (PWR) (Including Boron Recovery System)	II.1	CVCS Functional Performance during Adverse Environmental Phenomena; Pumping Capacity; and defense-in-depth RCS makeup	Partially Conforms	Applicable	The only CVCS safety related function is to preclude an inadvertent boron dilution of the reactor coolant system.	9.3.4
NSDSRS 9.3.4, Rev 0: Chemical and Volume Control System (PWR) (Including Boron Recovery System)	II.2	Single Failure Criteria and GDC 5	Conforms	Applicable	The single failure criteria apply only to the two safety related demineralized water isolation valves provided to preclude an inadvertent boron dilution of the reactor coolant system.	9.3.4
NSDSRS 9.3.4, Rev 0: Chemical and Volume Control System (PWR) (Including Boron Recovery System)	II.3	Minimization of contamination	Conforms	Applicable	None.	9.3.4
NSDSRS 9.3.4, Rev 0: Chemical and Volume Control System (PWR) (Including Boron Recovery System)	II.4	Components of the RCPB, quality classification and seismic design classification	Conforms	Applicable	The CVCS is located outside the RCPB.	9.3.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSDSRS 9.3.4, Rev 0: Chemical and Volume Control System (PWR) (Including Boron Recovery System)	II.5	Chemical and Volume Control System Design and Arrangement	Conforms	Applicable	None.	9.3.4
NSDSRS 9.3.4, Rev 0: Chemical and Volume Control System (PWR) (Including Boron Recovery System)	II.6	Detection of Reactor Coolant Leakage Outside Containment	Conforms	Applicable	None.	9.3.4
NSDSRS 9.3.4, Rev 0: Chemical and Volume Control System (PWR) (Including Boron Recovery System)	II.7	Prevention of CVCS Holdup Tank Wall Buckling/Failure; CVCS Venting and Draining	Partially Conforms	Applicable	A portion of this acceptance criterion is applicable but the specific language refers to CVCS designs that are not relevant to the NuScale design. Specifically, the CVCS does not have holdup tanks that are subject to the vacuum conditions that are addressed in subtier Bulletin 80-05. The last sentence of this acceptance criterion is applicable to the NuScale CVCS design, which will include appropriate venting and draining capability.	9.3.4
NSDSRS 9.3.4, Rev 0: Chemical and Volume Control System (PWR) (Including Boron Recovery System)	II.8	ITAAC	Conforms	Applicable	None.	9.3.4 14.3
SRP 9.3.5, Rev 3: Standby Liquid Control System (BWR)	All	Various	Not Applicable	Not Applicable	This SRP section and its acceptance criteria are applicable only to BWRs.	Not Applicable
NSDSRS 9.3.6, Rev 0: Containment Evacuation and Flooding Systems	II.1	GDC 2	Conforms	Applicable	None.	9.3.6
NSDSRS 9.3.6, Rev 0: Containment Evacuation and Flooding Systems	II.2	GDC 60	Conforms	Applicable	None.	9.3.6

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 9.3.6, Rev 0: Containment Evacuation and Flooding Systems	II.3	TMI 10 CFR 50.34(f)	Conforms	Applicable	None.	9.3.6
SRP 9.4.1, Rev 3: Control Room Area Ventilation System	II.1	Protection Against Natural Phenomena	Conforms	Applicable	None.	9.4.1
SRP 9.4.1, Rev 3: Control Room Area Ventilation System	II.2	Environmental and Dynamic Effects	Conforms	Applicable	None.	9.4.1
SRP 9.4.1, Rev 3: Control Room Area Ventilation System	II.3	Sharing of Structures, Systems, and Components	Conforms	Applicable	Operation of the CRVS is part of normal plant operations. All units share the same control room.	9.4.1
SRP 9.4.1, Rev 3: Control Room Area Ventilation System	II.4	Control Room	Conforms	Applicable	None.	9.4.1
SRP 9.4.1, Rev 3: Control Room Area Ventilation System	II.5	Control of Releases of Radioactive Material to the Environment	Conforms	Applicable	This acceptance criterion is applicable except for aspects related to ESF atmosphere cleanup systems. The NuScale control room habitability system neither relies on nor uses emergency filtration to protect operators during accident conditions. Rather, clean air is provided using compressed air tanks.	9.4.1

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 9.4.1, Rev 3: Control Room Area Ventilation System	II.6	Loss of All Alternating Current Power	Conforms	Applicable	The intent of this acceptance criterion and its subtler guidance - to maintain the ability to withstand and recover from a Station Blackout (SBO) lasting a specified minimum duration - is applicable. However, much of the specific language refers to reactor plant designs such as large LWRs, and is not relevant to the NuScale plant design. The NuScale plant design meets the intent of this guidance largely in its passive design and associated reduced reliance on AC power in coping with design basis events. Consistent with Commission policy, this strong coping capability eliminates any significant safety benefit a typical large LWR gains by having an alternate AC power source (e.g., gas turbine generator) for station blackout. Moreover, and specific to this SRP Section 9.4.1 acceptance criterion, the control room habitability system (Section 6.4) relies upon compressed air tanks to pressurize the control room envelope in the event of a SBO. The design of the main control room and the surrounding walls, ceiling, and structure act as a passive heat sink to maintain the environment within acceptable conditions in the event of an SBO.	9.4.1
SRP 9.4.2, Rev 3: Spent Fuel Pool Area Ventilation System	II.1	Compliance with GDC 2	Conforms	Applicable	None.	9.4.2
SRP 9.4.2, Rev 3: Spent Fuel Pool Area Ventilation System	II.2	Compliance with GDC 5	Conforms	Applicable	None.	9.4.2



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 9.4.2, Rev 3: Spent Fuel Pool Area Ventilation System	II.3	Compliance with GDC 60	Conforms	Applicable	This acceptance criterion is applicable except for aspects related to ESF atmosphere cleanup systems (see RG 1.52).	9.4.2
SRP 9.4.2, Rev 3: Spent Fuel Pool Area Ventilation System	II.4	Compliance with GDC 61	Conforms	Applicable	This acceptance criterion is applicable except for aspects related to ESF atmosphere cleanup systems, as described in the comment above for RG 1.52 subtier to Acceptance Criterion II.3.	9.4.2
SRP 9.4.3, Rev 3: Auxiliary and Radwaste Area Ventilation System	II.1	Compliance with GDC 2	Conforms	Applicable	None.	9.4.3
SRP 9.4.3, Rev 3: Auxiliary and Radwaste Area Ventilation System	II.2	Compliance with GDC 5	Conforms	Applicable	None.	9.4.3
SRP 9.4.3, Rev 3: Auxiliary and Radwaste Area Ventilation System	II.3	Compliance with GDC 60	Not Applicable	Applicable	The RWBV system does not filter exhaust. Exhaust is filtered by the RBV system.	9.4.3

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 9.4.4, Rev 3: Turbine Area Ventilation System	All (II.1 thru II.3)	Compliance with GDC 2, GDC 5, and GDC 60	Not Applicable	Not Applicable	This SRP section and its acceptance criteria (II.1 through II.3) are applicable only to LWR designs that rely on the turbine area ventilation system, or portions thereof, to fulfill safety-related or risk-significant functions. The NuScale turbine building ventilation (TBV) system is not relied upon to control airborne radioactivity concentrations in the turbine building and gaseous effluents during normal operations (including anticipated operational occurrences) and after any accidents that result in a radioactive material release. Furthermore, there are no requirements for TBV system performance that are needed to preclude any adverse effect on safety- related functions during all conditions of plant operation.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 9.4.5, Rev 3: Engineered Safety Feature Ventilation System	All	Various	Not Applicable	Not Applicable	This SRP Section addresses ESF ventilation systems designed for fission product removal in a post-design basis accident environment. The NuScale design does not rely on ESF ventilation systems to mitigate the consequences of a design basis accident. Non-safety-related normal ventilation systems provide atmosphere cleanup capability, as necessary, that meets the design, testing, and maintenance guidelines specified in RG 1.140. These systems provide appropriate containment, confinement, and filtering to limit releases of airborne radioactivity to the environment during normal and postulated accident conditions. However, these systems are not required following an accident, and accordingly receive no credit in the determination of the radiological consequences of an accident.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 9.5.1.1, Rev 0: Fire Protection Program	II.1	Fire Protection Probabilistic Risk Assessment (Including Appendix C)	Not Applicable	Applicable	This acceptance criterion (RG 1.174) is directed towards existing reactor licensees that are using PRA to support proposed changes to their fire protection licensing basis. This criterion and its subtier guidance could be applied to the initial development of a risk-informed, performance-based fire protection program pursuant to 10 CFR 50.48(c). However, development and implementation of a risk-informed, performance-based fire protection program would be the responsibility of COL applicants that reference the NuScale design, and that elect to implement the provisions of 10 CFR 50.48(c).	Not Applicable
SRP 9.5.1.1, Rev 0: Fire Protection Program	II.2	Fire Protection Program Considerations for License Renewal (Including Appendix B)	Not Applicable	Not Applicable	This acceptance criterion is applicable only to reactor licensees seeking license renewal.	Not Applicable
SRP 9.5.1.1, Rev 0: Fire Protection Program	II.3	NRC Staff Positions and Guidelines on Fire Protection	Partially Conforms	Applicable	This acceptance criterion is applicable except NuScale will use the current year subtier documents.	9.5.1 Table 9.5.1-2
SRP 9.5.1.1, Rev 0: Fire Protection Program	II.4	Fire Protection for Permanently Shutdown and Decommissioning Reactor Plants	Not Applicable	Not Applicable	This acceptance criterion (RG 1.191) is applicable only to reactor licensees that have submitted the necessary certifications for license termination under 10 CFR 50.82.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 9.5.1.1, Rev 0: Fire Protection Program	II.5	Fire Protection Program for New Reactor Combined License Applications	Partially Conforms	Applicable	This acceptance criterion and its subtier guidance are directed towards COL applicants under 10 CFR 52. Accordingly, COL applicants referencing a certified design would be responsible for implementing this guidance. Consistent with SRP Section 1.0, Appendix A; RG 1.206, Regulatory Position C.III.4; and ESP/DC/COL-ISG-015, the DCA contains COL information items, as appropriate, that describe the information that is deferred to the COL applicant referencing the certified design. Notwithstanding the above, NuScale, as an applicant for a design certification, would consider this guidance to be applicable to the design certification application to the extent necessary to ensure that the COL applicant can satisfy this guidance.	9.5.1 Appendix 9A
SRP 9.5.1.1, Rev 0: Fire Protection Program	II.6	Enhanced Fire Protection Criteria for New Reactor Designs (Including Appendix A)	Partially Conforms	Applicable	The enhanced fire protection criteria for new reactor designs specify passive separation of redundant trains as the preferred approach to ensure safe-shutdown capability. Due to the modular nature and small size of the NuScale power module (NPM), it is not feasible in all instances to provide installed passive separation of redundant trains. When train separation is not feasible, fire protection for redundant shutdown systems is employed to ensure, to the extent practicable, such that one shutdown division will be free of fire damage.	9.5.1 Table 9.5.1-2 Appendix 9A

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 9.5.1.1, Rev 0: Fire Protection Program	II.7	Operational Program and Proposed Implementation Milestones	Not Applicable	Applicable	The information governed by this acceptance criterion is largely site-specific and is the responsibility of the permit/COL applicant. Consistent with SRP Section 1.0, Appendix A; RG 1.206, Regulatory Position C.III.4; and ESP/DC/COL-ISG-015, the DCA contains COL information items that describe the information that is deferred to the COL applicant referencing the certified design. Notwithstanding the above, NuScale considered this guidance to the extent necessary to ensure that the COL applicant can satisfy this guidance.	Not Applicable
SRP 9.5.1.2, Rev 0: Risk-Informed, Performance-Based Fire Protection Program	All	Various	Not Applicable	Applicable	This SRP section and its acceptance criteria (II.1 through II.11) are applicable only to reactor licensees and applicants that are developing or revising a risk-informed, performance-based fire protection program pursuant to 10 CFR 50.48(c). Development and implementation of a risk-informed, performance-based fire protection program would be the responsibility of COL applicants that reference the NuScale design, and that elect to implement the provisions of 10 CFR 50.48(c).	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 9.5.2, Rev 0: Communication Systems	II.1	Emergency Facilities and Equipment	Partially Conforms	Applicable	This acceptance criterion governs site-specific emergency response communication systems that are the responsibility of the COL applicant. Consistent with SRP Section 1.0, Appendix A; RG 1.206, Position C.III.4; and ESP/DC/COL-ISG-015, the DCA contains COL information items, as appropriate, that describe the site-specific information that is deferred to the COL applicant referencing the certified design.	9.5.2
NSDSRS 9.5.2, Rev 0: Communication Systems	II.2	Onsite Technical Support Center and Operational Support Center	Partially Conforms	Applicable	The NuScale standard plant design will include provisions for an onsite technical support center and an onsite operational support center as specified by 10 CFR 50.34(f)(2)(xxv) and this acceptance criterion. However, communication systems serving these facilities in support of emergency response are part of the site-specific design that are the responsibility of the COL applicant. Consistent with SRP Section 1.0, Appendix A; RG 1.206, Position C.III.4; and ESP/DC/COL-ISG-015, the DCA contains COL information items that describe the site-specific information that is deferred to the COL applicant referencing the certified design.	9.5.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSDSRS 9.5.2, Rev 0: Communication Systems	II.3	Emergency Facilities and Equipment for Meeting 10 CFR 52.47(a)(8)	Partially Conforms	Applicable	The NuScale design will include provisions for design-specific emergency facilities (i.e., pertain to design features, facilities, functions, and equipment that are technically relevant to the NuScale standard plant design), consistent with 10 CFR 50.47(a)(8) and this acceptance criterion. However, communication systems and equipment serving these facilities in support of emergency response are part of the site-specific design that are the responsibility of the COL applicant. Consistent with SRP Section 1.0, Appendix A; RG 1.206, Position C.III.4; and ESP/DC/COL-ISG-015, the DCCA contains COL information items that describe the site-specific information that is deferred to the COL applicant referencing the certified design.	9.5.2
NSDSRS 9.5.2, Rev 0: Communication Systems	II.4	Design, Fabrication, Erection, Construction, Testing, and Inspection of SSCs to Meet 10 CFR 50.55a	Partially Conforms	Applicable	The aspects of this acceptance criterion within the scope of the NuScale standard plant design are applicable to the DCA. Aspects related to site-specific design, fabrication, erection, construction, testing, and inspection of SSCs are the responsibility of the COL applicant referencing the certified design.	9.5.2



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 9.5.2, Rev 0: Communication Systems	II.5	ITAAC	Conforms	Applicable	The aspects of this acceptance criterion within the scope of the NuScale standard plant design are applicable to the DCA. Aspects related to site-specific design, fabrication, erection, construction, testing, and inspection of SSCs, and the maintenance of records for those activities throughout the life of the facility, are the responsibility of the COL applicant referencing the certified design.	Ch 14
NSDSRS 9.5.2, Rev 0: Communication Systems	II.6	ITAAC for a COL applicant	Not Applicable	Applicable	COL applicant responsibility to prepare COL specific ITAAC. See Section 14.3 for ITAAC development methodology. Review the communication systems facility design functions as presented in this section in order to assess need for ITAAC.	Not Applicable
NSDSRS 9.5.2, Rev 0: Communication Systems	II.7	Compliance with GDC 1	Conforms	Applicable	Design documents are developed which comply with the requirements of GDC-1 for the standard plant relative to application of quality standards in support of design, fabrication, erection, and testing of communication systems. Applicant must comply with GDC-1 for site-specific scope.	9.5.2
NSDSRS 9.5.2, Rev 0: Communication Systems	II.8	Compliance with GDC 2	Conforms	Applicable	Design documents are developed which comply with the requirements of GDC-2 pertaining to protection from natural phenomena as it relates to communication equipment.	9.5.2
NSDSRS 9.5.2, Rev 0: Communication Systems	II.9	Compliance with GDC 3	Conforms	Applicable	Design documents are developed which comply with the requirements of GDC-3 pertaining to fire protection as it relates to communications.	9.5.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 9.5.2, Rev 0: Communication Systems	II.10	Compliance with GDC 4	Conforms	Applicable	Design documents are developed for the communications systems comply with the requirements of GDC-4 relative to protection from deleterious impact of Environmental and Dynamic Effects.	9.5.2
NSDSRS 9.5.2, Rev 0: Communication Systems	II.11	Compliance with GDC 19	Conforms	Applicable	Design documents are developed which comply with the requirements of GDC-19 relative to ensuring that communication equipment is provided at appropriate locations inside the control room with the capability to support all normal and emergency operations, including intra-plant communications and plant to emergency facilities and offsite communication requirements even in the event of a single failure within a communication subsystem or the loss of the normal power source. Design addresses control room communications from the standpoint that the control room can maintain communications with site and offsite entities during normal and accident conditions.	9.5.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSDSRS 9.5.2, Rev 0: Communication Systems	II.12	Compliance with 10 CFR 73.45(e)(2)(iii), 10 CFR 73.45(g)(4)(i), and 10 CFR 73.45(g)(4)(ii)	Not Applicable	Not Applicable	This acceptance criterion is applicable only to licensees subject to 10 CFR 73.45 and the general performance requirements of 10 CFR 73.20. Licensees referencing the certified design would not be subject to 10 CFR 73.20 since it is not anticipated that a licensee referencing the certified design would possess, use, deliver, transport, export, or import formula quantities of strategic special nuclear material that would not be subject to the exemptions of 10 CFR 73.6 and 10 CFR 73.55(l)(2). Accordingly, the performance requirements of 10 CFR 73.55 would be the applicable requirements for power reactor licensees such as licensees referencing a certified standard design.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 9.5.2, Rev 0: Communication Systems	II.13	Compliance with 10 CFR 73.46(f)	Conforms	Applicable	Much of this acceptance criterion governs site-specific, programmatic aspects of physical security communication systems that are the responsibility of the COL applicant referencing the certified design. Consistent with SRP Section 1.0, Appendix A; RG 1.206, Position C.III.4; and ESP/DC/COL-ISG-015, the DCA contains COL information items that describe the site-specific, programmatic information that is deferred to the COL applicant referencing the certified design. Aspects of this acceptance criterion that are related to the physical design of the power reactor and communication systems within the scope of the certified design are applicable to the DCA.	9.5.2
NSDSRS 9.5.2, Rev 0: Communication Systems	II.14	Compliance with 10 CFR 73.55(e)(9)(vi)(B)	Conforms	Applicable	Much of this acceptance criterion governs site-specific, programmatic aspects of physical security communication systems that are the responsibility of the COL applicant referencing the NuScale design. Aspects of this acceptance criterion that are related to the physical design of the power reactor and communication systems within the scope of the certified design are applicable to the DCA.	13.6.2 (via Security Technical Report)

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 9.5.2, Rev 0: Communication Systems	II.15	Compliance with 10 CFR 73.55(j)	Partially Conforms	Applicable	Design focus pertains to addressing requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage and the communication requirements necessary to afford this protection. Elements of this design fall under the COL applicant and are addressed as part of the facility physical security plan.	13.6
SRP 9.5.3, Rev 3: Lighting Systems	II.1	Integrated Design of the System	Conforms	Applicable	None.	9.5.3
SRP 9.5.3, Rev 3: Lighting Systems	II.2	Emergency Lighting System(s)	Conforms	Applicable	None.	9.5.3
SRP 9.5.3, Rev 3: Lighting Systems	II.3	Lighting Levels	Conforms	Applicable	None.	9.5.3
SRP 9.5.4, Rev 3: Emergency Diesel Engine Fuel Oil Storage and Transfer System	All (II.1 thru II.4)	Compliance with GDC 2, GDC 4, GDC 5, and GDC 17	Not Applicable	Not Applicable	The NuScale plant design does not require or include safety-related emergency diesel generators that would be subject to this SRP section. No AC or DC power is relied upon for the performance of NuScale plant safety functions.	Not Applicable
SRP 9.5.5, Rev 3: Emergency Diesel Engine Cooling Water System	All (II.1 thru II.7)	Compliance with GDC 2, GDC 4, GDC 5, GDC 17, GDC 44, GDC 45, and GDC 46	Not Applicable	Not Applicable	The NuScale plant design does not require or include safety-related emergency diesel generators that would be subject to this SRP section. No AC or DC power is relied upon for the performance of NuScale plant safety functions.	Not Applicable
SRP 9.5.6, Rev 3: Emergency Diesel Engine Starting System	All (II.1 thru II.4)	Compliance with GDC 2, GDC 4, GDC 5, and GDC 17	Not Applicable	Not Applicable	The NuScale plant design does not require or include safety-related emergency diesel generators that would be subject to this SRP section. No AC or DC power is relied upon for the performance of NuScale plant safety functions.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 9.5.7, Rev 3: Emergency Diesel Engine Lubrication System	All (II.1 thru II.4)	Compliance with GDC 2, GDC 4, GDC 5, and GDC 17	Not Applicable	Not Applicable	The NuScale plant design does not require or include safety-related emergency diesel generators that would be subject to this SRP section. No AC or DC power is relied upon for the performance of NuScale plant safety functions.	Not Applicable
SRP 9.5.8, Rev 3: Emergency Diesel Engine Combustion Air Intake and Exhaust System	All (II.1 thru II.4)	Compliance with GDC 2, GDC 4, GDC 5, and GDC 17	Not Applicable	Not Applicable	The NuScale plant design does not require or include safety-related emergency diesel generators that would be subject to this SRP section. No AC or DC power is relied upon for the performance of NuScale plant safety functions.	Not Applicable
SRP 10.2, Rev 3: Turbine Generator	II.1	Protect SSCs important to safety from the effects of turbine missiles with a turbine overspeed protection system (GDC 4)	Partially Conforms	Applicable	The combination of turbine rotor inspections and the low probability of turbine missile generation is sufficient to protect SSCs from the adverse effects of turbine missiles.	10.2.2
SRP 10.2, Rev 3: Turbine Generator	II.2	Inservice Inspection covering valves essential for overspeed protection.	Conforms	Applicable	None.	10.2.2
SRP 10.2, Rev 3: Turbine Generator	II.3	Prevention of Adverse Effects on Safety-Related SSCs in the Turbine Building	Not Applicable	Not Applicable	There are no safety-related SSCs in the turbine building.	Not Applicable
NSDSRS 10.2.3, Rev 0: Turbine Rotor Integrity	II.1	Materials Selection	Conforms	Applicable	None.	10.2.3
NSDSRS 10.2.3, Rev 0: Turbine Rotor Integrity	II.2	Fracture Toughness	Conforms	Applicable	None.	10.2.3
NSDSRS 10.2.3, Rev 0: Turbine Rotor Integrity	II.3	Pre-Service Inspection	Conforms	Applicable	None.	10.2.3
NSDSRS 10.2.3, Rev 0: Turbine Rotor Integrity	II.4	Turbine Rotor Design	Conforms	Applicable	None.	10.2.3
NSDSRS 10.2.3, Rev 0: Turbine Rotor Integrity	II.5	Inservice Inspection	Conforms	Applicable	None.	10.2.3

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 10.2.3, Rev 0: Turbine Rotor Integrity	II.6	10 CFR 52.47(b)(1) ITAAC	Conforms	Applicable	Including ITAAC because it is noted as NSDSRS guidance.	10.2.3
NSDSRS 10.3, Rev 0: Main Steam Supply System	II.1	Protection against natural phenomena (GDC 2)	Conforms	Applicable	The NuScale MSS is not safety-related, but the portion of the system downstream of the MSIVs inside the RXB is considered risk-significant because the secondary MSIVs act as backup to the MSIVs. Functionality is ensured by the design and construction of the MSS to the provisions of RG 1.29, Regulatory Position C.2 and C.3.	10.3.1
NSDSRS 10.3, Rev 0: Main Steam Supply System	II.2	Protection of SSCs important to safety from the effects of turbine missiles (GDC 4)	Conforms	Applicable	The NuScale MSS is not safety-related or risk-significant. Thus, the applicability of GDC 4 to the NuScale MSS reviewed under this acceptance criterion is limited to aspects ensuring that a failure of the nonsafety-related SSC does not result in an adverse effect on a safety-related SSC.	10.3.1
NSDSRS 10.3, Rev 0: Main Steam Supply System	II.3	Shared SSCs important to safety perform required safety functions (GDC 5)	Conforms	Applicable	None.	10.3.1
NSDSRS 10.3, Rev 0: Main Steam Supply System	II.4	MSS is capable of supporting core cooling and/or safe-shutdown (non-DBA) in the event of an SBO (10 CFR 50.63)	Partially Conforms	Applicable	The intent of this acceptance criterion and its subtier guidance is applicable. The NuScale plant design meets the intent of this guidance largely in its passive design and associated reduced reliance on AC power in coping with design basis events.	10.3.1
NSDSRS 10.3, Rev 0: Main Steam Supply System	II.5	Protection of Important-to-Safety SSCs from Tornado Missiles (RG 1.117, Appendix Positions 2 and 4)	Conforms	Applicable	None.	10.3.1
SRP 10.3.6, Rev 3: Steam and Feedwater System Materials	II.1	Materials Selection and Fabrication of Class 2 and 3 Components	Partially Conforms	Applicable	This acceptance criterion is applicable except for reference to subtier RG 1.37, RG 1.37 has been withdrawn by the NRC.	10.3.6

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 10.3.6, Rev 3: Steam and Feedwater System Materials	II.2	Fracture Toughness of Class 2 and 3 Components	Conforms	Applicable	None.	10.3.6
SRP 10.4.1, Rev 3: Main Condensers	II.1	Prevent excessive releases of radioactivity to the environment (GDC 60)	Conforms	Applicable	None.	10.4.1
SRP 10.4.2, Rev 3: Main Condenser Evacuation System	II.1	Prevent excessive releases of radioactivity to the environment (GDC 60)	Conforms	Applicable	None.	10.4.2
SRP 10.4.3, Rev 3: Turbine Gland Seal	II.1	Prevent excessive releases of radioactivity to the environment (GDC 60)	Conforms	Applicable	None.	10.4.3
SRP 10.4.4, Rev 3: Turbine Bypass System	II.1	Piping Failures (GDC 4)	Conforms	Applicable	None.	10.4.4
SRP 10.4.4, Rev 3: Turbine Bypass System	II.2	Residual Heat Removal (GDC 34)	Conforms	Applicable	None.	10.4.4
SRP 10.4.4, Rev 3: Turbine Bypass System	II.3	MSIV Alternate Leakage Path	Not Applicable	Not Applicable	BWR only.	10.4.4
SRP 10.4.5, Rev 3: Circulating Water System	II.1	Flooding of SSCs important to safety (GDC 4)	Conforms	Applicable	None.	10.4.5
SRP 10.4.6, Rev 3: Condensate Cleanup System	II.1	Maintain direct cycle BWR plant water quality to avoid corrosion-induced failure of the reactor coolant pressure boundary (GDC 14)	Not Applicable	Not Applicable	BWR only.	10.4.6
SRP 10.4.6, Rev 3: Condensate Cleanup System	II.2	Maintain indirect cycle PWR water quality to avoid corrosion-induced failure of the reactor coolant pressure boundary (GDC 14)	Conforms	Applicable	In the NuScale SG design, the primary water is outside the steam generator tubes, the secondary water is inside the tubes, and there is no SG blowdown so the secondary chemistry requirements for the NuScale design differ from those outlined in the referenced EPRI report.	10.4.6
NSDSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.1	Seismic Events (GDC 2)	Conforms	Applicable	None.	10.4.7



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.2	Fluid Instabilities (GDC 4)	Partially Conforms	Applicable	The intent of this acceptance criterion and its subtier guidance - to satisfy GDC 4 related to protecting SSCs from fluid flow instability effects such as water hammer - is applicable. However, much of the specific language in the subtier guidance refers to reactor plant designs such as large LWRs, and is not relevant to the NuScale plant design.	10.4.7
NSDSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.3	Sharing of Structures, Systems, and Components (GDC 5)	Conforms	Applicable	None.	10.4.7
NSDSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.4	Heat Removal Capability (GDC 44)	Conforms	Applicable	None.	10.4.7
NSDSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.5	Inspection (GDC 45)	Conforms	Applicable	None.	10.4.7
NSDSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.6	Testing (GDC 46)	Conforms	Applicable	None.	10.4.7
NSDSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.7	Flow Accelerated Corrosion	Conforms	Applicable	None.	10.4.7
SRP 10.4.8, Rev 3: Steam Generator Blowdown System	All	Various	Not Applicable	Not Applicable	The NuScale steam generator design does not use a blowdown system.	Not Applicable
SRP 10.4.9, Rev 3: Auxiliary Feedwater System (PWR)	All	Various	Not Applicable	Not Applicable	The NuScale design neither requires nor uses an auxiliary feedwater system. The NuScale decay heat removal (DHR) system performs some functions similar to an auxiliary feedwater system. However, as compared to an auxiliary feedwater system, the DHR system differs substantially in its design, operation, and relationship to the small break LOCA plant response.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
BTP 10-1, Rev 3: Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants	All	Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity	Not Applicable	Not Applicable	This guidance is applicable only to large PWRs that use Auxiliary Feedwater (AFW) system pumps powered by electrical and steam sources. The NuScale DHR system fulfills a similar function as the AFW system at a large PWR. The NuScale DHR system design does not use pumps, but rather operates via passive natural circulation.	Not Applicable
BTP 10-2, Rev 4: Design Guidelines for Avoiding Water Hammers in Steam Generators	TFSGD B.1 thru B.4	Top-Feed Steam Generator Designs	Not Applicable	Not Applicable	The NuScale plant design does not use a top-feed steam generator design.	Not Applicable
BTP 10-2, Rev 4: Design Guidelines for Avoiding Water Hammers in Steam Generators	PSGD B.1 thru B.4	Preheat Steam Generator Designs	Not Applicable	Not Applicable	The NuScale plant design does not use a preheat steam generator design.	Not Applicable
BTP 10-2, Rev 4: Design Guidelines for Avoiding Water Hammers in Steam Generators	OTSGD B.1	Once-Through Steam Generator Designs - Auxiliary Feedwater Supply	Not Applicable	Not Applicable	This acceptance criterion is applicable only to large PWRs that use a once-through steam generator design. The NuScale plant design does not involve an AFW system as would be found at a typical large LWR, but does include the DHR system that fulfills a substantively similar function as a typical AFW system. However, the NuScale steam generator design precludes potential water hammer issues without providing DHR system water through an externally mounted supply top discharge header as is prescribed by this acceptance criterion.	Not Applicable
BTP 10-2, Rev 4: Design Guidelines for Avoiding Water Hammers in Steam Generators	OTSGD B.2	Once-Through Steam Generator Designs - Tests and Test Procedures	Conforms	Applicable	None.	5.4.1

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 11.1, Rev 0: Coolant Source Terms	II.1	RG 1.110	Partially Conforms	Applicable	See RG 1.110 in Table 1.9-2.	11.1
NSDSRS 11.1, Rev 0: Coolant Source Terms	II.2	RG 1.112	Partially Conforms	Applicable	See RG 1.112 in Table 1.9-2.	11.1
NSDSRS 11.1, Rev 0: Coolant Source Terms	II.3	RG 1.140	Partially Conforms	Applicable	RG 1.140 in Table 1.9-2.	11.1
NSDSRS 11.1, Rev 0: Coolant Source Terms	II.4	DC/COL-ISG-5	Not Applicable	Applicable	The NuScale design is similar to large PWRs in the existing fleet with regards to effluent release calculations. However, the development of an alternate methodology is necessary because the existing PWRGALE code was developed in the 1980s for evaluation of the large PWR reactors of that time and does not appropriately address the NuScale plant design.	11.1
NSDSRS 11.1, Rev 0: Coolant Source Terms	II.5	normal operation and AOO sources of radioactive liquid and gaseous effluents	Conforms	Applicable	None.	11.1
NSDSRS 11.1, Rev 0: Coolant Source Terms	II.6	Release rates should be developed using methods that are consistent with NUREG-0017, PWR-GALE86, or ANSI/ANS 18.1-1999	Partially Conforms	Applicable	The NuScale design is similar to large PWRs in the existing fleet with regards to effluent release calculations. However, the development of an alternate methodology is necessary because the existing PWRGALE code was developed in the 1980s for evaluation of the large PWR reactors of that time and does not appropriately address the NuScale plant design. Some aspects of ANSI/ANS 18.1 are utilized for the coolant source terms.	11.1
NSDSRS 11.1, Rev 0: Coolant Source Terms	II.7	Decontamination factors used to reduce gaseous effluent releases to the environment	Partially Conforms	Applicable	Decontamination factors are consistent with the NuScale Normal Effluent Technical Report.	11.1
NSDSRS 11.1, Rev 0: Coolant Source Terms	II.8	Decontamination factors applied to reduce liquid effluent releases to the environment	Partially Conforms	Applicable	Decontamination factors are consistent with the NuScale Normal Effluent Technical Report.	11.1

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 11.1, Rev 0: Coolant Source Terms	II.9	RWMS system augmentations used in cost-benefit calculations are consistent with the guidance of RG 1.110	Partially Conforms	Applicable	See RG 1.110 in Table 1.9-2.	11.1
NSDSRS 11.1, Rev 0: Coolant Source Terms	II.10	primary and secondary coolant source terms, used in characterizing liquid and gaseous effluents	Conforms	Applicable	None.	11.2 11.3
NSDSRS 11.1, Rev 0: Coolant Source Terms	II.11	If neutron activation products are expected in reactor pool water and secondary coolant	Conforms	Applicable	None.	11.1
NSDSRS 11.1, Rev 0: Coolant Source Terms	II.12	10 CFR 50.34(b)(3), 10 CFR 50.34a, and 10 CFR 52.79(a)(3).	Partially Conforms	Applicable	The NuScale design is similar to large PWRs in the existing fleet with regards to effluent release calculations. However, the development of an alternate methodology is necessary because the existing PWRGALE code was developed in the 1980s for evaluation of the large PWR reactors of that time and does not appropriately address the NuScale plant design.	11.1
NSDSRS 11.1, Rev 0: Coolant Source Terms	II.13	The design basis coolant source term is based on a combination of assumptions of failed fuel fractions	Partially Conforms	Applicable	The design basis coolant source term for NuScale is partially based on a failed fuel fraction much less than 0.25%, which is described in NuScales Normal Effluent Technical Report.	11.1
NSDSRS 11.1, Rev 0: Coolant Source Terms	II.14	calculational technique or any source term parameter	Conforms	Applicable	None.	11.1
NSDSRS 11.2, Rev 0: Liquid Waste Management System	II.1	Capability to Meet Dose Design Objectives	Partially Conforms	Applicable	This acceptance criterion is applicable except for aspects that are related to performance of a site-specific cost-benefit analysis, which is the responsibility of the COL applicant.	11.2.3
NSDSRS 11.2, Rev 0: Liquid Waste Management System	II.2	Design for Anticipated Processing Requirements	Conforms	Applicable	None.	11.2.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 11.2, Rev 0: Liquid Waste Management System	II.3	Seismic Design of Structures Housing Liquid Waste Management System Components	Conforms	Applicable	None.	11.2.2
NSDSRS 11.2, Rev 0: Liquid Waste Management System	II.4	Provisions to Control Leakage and Facilitate Operation and Maintenance	Conforms	Applicable	None.	11.2.2
NSDSRS 11.2, Rev 0: Liquid Waste Management System	II.5	Automatic control features	Conforms	Applicable	None.	11.2 11.5 11.6
NSDSRS 11.2, Rev 0: Liquid Waste Management System	II.6	Exhaust ventilation system	Conforms	Applicable	None.	11.3
NSDSRS 11.2, Rev 0: Liquid Waste Management System	II.7	Criteria for Early Site Permit Applications	Not Applicable	Not Applicable	This acceptance criterion is applicable only to applicants for an early site permit.	Not Applicable
NSDSRS 11.3, Rev 0: Gaseous Waste Management System	II.1	Capability to Meet Dose Design Objectives	Partially Conforms	Applicable	This acceptance criterion is applicable except for aspects that are related to performance of a site-specific cost-benefit analysis, which is the responsibility of the COL applicant.	11.3.1 11.3.2 11.3.3 11.3.4
NSDSRS 11.3, Rev 0: Gaseous Waste Management System	II.2	Design for Anticipated Processing Requirements	Conforms	Applicable	None.	11.3.2
NSDSRS 11.3, Rev 0: Gaseous Waste Management System	II.3	Seismic Design and Quality Group Classification of Components and Structures Housing Gaseous Waste Management System	Conforms	Applicable	None.	11.3.1
NSDSRS 11.3, Rev 0: Gaseous Waste Management System	II.4	Features to Minimize Contamination, Facilitate Decommissioning, and Minimize Generation of Radwaste	Partially Conforms	Applicable	This acceptance criterion is applicable except for aspects that govern site-specific activities that are the responsibility of the COL applicant.	11.3.2
NSDSRS 11.3, Rev 0: Gaseous Waste Management System	II.5	Design, Testing, and Maintenance of HEPA Filters and Charcoal Adsorbers	Conforms	Applicable	None.	11.3.1 11.3.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSDSRS 11.3, Rev 0: Gaseous Waste Management System	II.6	Automatic control features	Conforms	Applicable	None.	11.3.7
NSDSRS 11.3, Rev 0: Gaseous Waste Management System	II.7	Design to Withstand Effects of Hydrogen Explosion	Conforms	Applicable	None.	11.3.2
NSDSRS 11.3, Rev 0: Gaseous Waste Management System	II.8	Postulated Leakage or Failure of a Waste Gas Storage Tank or Offgas Charcoal Delay Bed	Conforms	Applicable	None.	11.3.3
NSDSRS 11.3, Rev 0: Gaseous Waste Management System	II.9	Criteria for Early Site Permit Applications	Not Applicable	Not Applicable	This acceptance criterion is applicable only to applicants for an early site permit.	Not Applicable
NSDSRS 11.3, Rev 0: Gaseous Waste Management System	II.10	Relevant RGs, ISG, and BTP	Partially Applicable	Applicable	As described above.	As listed above.
NSDSRS 11.4, Rev 0: Solid Waste Management System	II.1	Design Parameters Based on Expected Radionuclide Distributions and Concentrations	Conforms	Applicable	None.	Table 11.4-5 thru Table 11.4-9
NSDSRS 11.4, Rev 0: Solid Waste Management System	II.2	Sizing of Processing Equipment	Conforms	Applicable	None.	11.4.2
NSDSRS 11.4, Rev 0: Solid Waste Management System	II.3	Liquid and Wet Waste Stabilization in Accordance with Process Control Program	Partially Conforms	Applicable	This acceptance criterion is applicable except for aspects related to development and implementation of a Process Control Program (PCP), which is the responsibility of the COL applicant.	11.4.2
NSDSRS 11.4, Rev 0: Solid Waste Management System	II.4	Stabilization of Other Forms of Wet Waste in Accordance with Process Control Program	Not Applicable	Applicable	The development and implementation of a PCP is the responsibility of the COL applicant.	11.4
NSDSRS 11.4, Rev 0: Solid Waste Management System	II.5	Design Objectives, Design Criteria, Treatment Methods, Expected Effluent Releases, Monitoring and Control Instrumentation Setpoints	Not Applicable	Applicable	The development and implementation of a PCP and ODCM are the responsibility of the COL applicant.	11.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 11.4, Rev 0: Solid Waste Management System	II.6	Waste Containers, Shipping Casks, and Waste Packaging	Partially Conforms	Applicable	This guidance is applicable to design certification except for site-specific, programmatic and operational aspects that are the responsibility of the COL applicant.	11.4.1 11.4.2
NSDSRS 11.4, Rev 0: Solid Waste Management System	II.7	Onsite Waste Storage Facilities	Partially Conforms	Applicable	This guidance is applicable to design certification except for site-specific, programmatic and operational aspects that are the responsibility of the COL applicant.	11.4.1 11.4.2
NSDSRS 11.4, Rev 0: Solid Waste Management System	II.8	Seismic Design and Quality Group Classification of Components and Structures Housing Solid Waste Management System	Conforms	Applicable	None.	3.8 11.4.1 Table 11.4-1 11.4.2
NSDSRS 11.4, Rev 0: Solid Waste Management System	II.9	Provisions to Control Leakage and Facilitate Operation and Maintenance	Partially Conforms	Applicable	This acceptance criterion is applicable except for aspects that govern site-specific activities that are the responsibility of the COL applicant.	11.4.1 11.4.3 12.3
NSDSRS 11.4, Rev 0: Solid Waste Management System	II.10	Features to Minimize Contamination, Facilitate Decommissioning, and Minimize Generation of Radwaste	Partially Conforms	Applicable	This acceptance criterion is applicable except for aspects that govern site-specific activities that are the responsibility of the COL applicant.	11.4.1 11.4.3 12.3
NSDSRS 11.4, Rev 0: Solid Waste Management System	II.11	Storage Facility Design for Long Term Onsite Storage (Including Appendix 11.4A)	Not Applicable	Applicable	The NuScale design has no long term storage facility for solid radioactive waste. This is a COL applicant responsibility.	11.4.2
NSDSRS 11.4, Rev 0: Solid Waste Management System	II.12	Class A, B, C - Processing and Disposing of Liquid, Wet, and Dry Solid Wastes	Partially Conforms	Applicable	This acceptance criterion governs site-specific, programmatic aspects of the PCP development and implementation that are the responsibility of the COL applicant. NuScale will consider this guidance to be applicable to the extent necessary to ensure that the COL applicant referencing the certified design can satisfy the guidance contained therein.	11.4.2 11.4.3

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSDSRS 11.4, Rev 0: Solid Waste Management System	II.13	Greater than Class C - Processing and Disposing of Liquid, Wet, and Dry Solid Wastes	Partially Conforms	Applicable	This acceptance criterion governs site-specific, programmatic aspects of the PCP development and implementation that are the responsibility of the COL applicant. NuScale will consider this guidance to be applicable to the extent necessary to ensure that the COL applicant referencing the certified design can satisfy the guidance contained therein.	11.4.1 11.4.2
NSDSRS 11.4, Rev 0: Solid Waste Management System	II.14	Processing and Disposing of Mixed Wastes	Partially Conforms	Applicable	This acceptance criterion governs site-specific, programmatic aspects of PCP implementation (specific to mixed waste processing and disposal) that are the responsibility of the COL applicant. NuScale will consider this guidance to be applicable to the extent necessary to ensure that the COL applicant referencing the certified design can satisfy the guidance contained therein.	11.4.2
NSDSRS 11.4, Rev 0: Solid Waste Management System	II.15	All Effluent Releases Associated with Operation of the SWMS	Partially Conforms	Applicable	This acceptance criterion is applicable except for site- specific, programmatic aspects that are the responsibility of the COL applicant.	11.4.2



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 11.4, Rev 0: Solid Waste Management System	II.16	Operational Programs	Not Applicable	Applicable	The information governed by this acceptance criterion is largely site-specific and is the responsibility of the COL applicant. Consistent with SRP Section 1.0, Appendix A; RG 1.206, Regulatory Position C.III.4; and ESP/DC/ COL-ISG-015, the DCA contains COL information items, as appropriate, that describe the information that is deferred to the COL applicant referencing the certified design. Notwithstanding the above, NuScale considered this guidance to be applicable to the DCA to the extent necessary to ensure that the COL applicant can satisfy this guidance.	11.4
NSDSRS 11.4, Rev 0: Solid Waste Management System	II.17	Automatic control features	Conforms	Applicable	The SRWM system does not use automatic controls from radiation detectors.	11.4
NSDSRS 11.4, Rev 0: Solid Waste Management System	II.18	Design of exhaust ventilation systems	Conforms	Applicable	None.	11.4.2
NSDSRS 11.4, Rev 0: Solid Waste Management System	II.19	Seismic design of structures housing SWMS	Conforms	Applicable	None.	11.4.1
NSDSRS 11.5, Rev 0: Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	II.1	Installation of Instrumentation and Monitoring Equipment and Sampling and Analyses of Normal and Potential Effluent Pathways	Partially Conforms	Applicable	This acceptance criterion is applicable except for certain aspects of its subtier guidance (see RG 1.21, RG 1.33, RG 1.97, RG 4.1, RG 4.15, and BTP 7-10).	9.3.2 11.2 11.3 11.5
NSDSRS 11.5, Rev 0: Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	II.2	Instrumentation and Monitoring Equipment and Sampling and Analysis of Radioactive Waste Process Systems (Including Appendix 11.5A)	Partially Conforms	Applicable	This acceptance criterion is applicable except for certain aspects of its subtier guidance (see RG 1.21, RG 1.33, RG 1.97, RG 4.15, RG 4.21 and BTP 7-10). Administrative and procedural controls are COL applicant responsibility.	9.3.2 11.5 12.3.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 11.5, Rev 0: Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	II.3	Provisions for Administrative and Procedural Controls (Including Appendix 11.5A)	Partially Conforms	Applicable	This acceptance criterion is applicable except for certain aspects of its subtier guidance (see RG 1.21, RG 1.33, RG 1.97 and RG 4.15). Administrative and procedural controls are COL applicant responsibility.	9.3.2 11.5 12.3
NSDSRS 11.5, Rev 0: Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	II.4	Monitoring, Sampling, and Analyses of All Identified Gaseous Effluent Release Paths (Including Appendix 11.5A)	Partially Conforms	Applicable	This acceptance criterion is applicable except for certain aspects of its subtier guidance (see RG 1.97 and BTP 7-10). Administrative and procedural controls are COL applicant responsibility.	11.5
NSDSRS 11.5, Rev 0: Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	II.5	Monitoring, Sampling, and Analysis of All Identified Liquid Effluent Release Paths	Partially Conforms	Applicable	Administrative and procedural controls are COL applicant responsibility.	11.5
NSDSRS 11.5, Rev 0: Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	II.6	Operational Programs	Not Applicable	Applicable	The information governed by this acceptance criterion is site-specific and is the responsibility of the COL applicant. Consistent with SRP Section 1.0, Appendix A; RG 1.206, Regulatory Position C.III.4; and ESP/DC/COL-ISG- 015, the DCA contains COL information items that describe the information that is deferred to the COL applicant referencing the certified design. Notwithstanding the above, NuScale considered this guidance to be applicable to the DCA to the extent necessary to ensure that the COL applicant can satisfy this guidance.	Not Applicable
NSDSRS 11.5, Rev 0: Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	II.7	Descriptions of design features and instrumentation used in primary and secondary coolant system leakage detection	Conforms	Applicable	None.	11.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSDSRS 11.6, Rev 0: Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring	II.1	Installation of instrumentation or sampling equipment	Conforms	Applicable	None.	11.5 11.6
NSDSRS 11.6, Rev 0: Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring	II.2	Gaseous and liquid release points should be monitored	Conforms	Applicable	None.	11.5 11.6
NSDSRS 11.6, Rev 0: Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring	II.3	Radiation exposure rates and airborne concentration monitoring locations and sampling points	Conforms	Applicable	None.	11.6 12.3
NSDSRS 11.6, Rev 0: Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring	II.4	Compliance with GDC 63 & 64 via post-TMI action plan items	Partially Conforms	Applicable	This acceptance criterion is applicable except for certain aspects of its subtier regulation 10 CFR 50.34(f)(2)(xxvi) as it relates to testing and operational programs, which are a COL applicant responsibility.	9.3.2 11.5 11.6

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSDSRS 11.6, Rev 0: Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring	II.5	Ensure samples are representative	Conforms	Applicable	None.	9.3.2 11.6
NSDSRS 11.6, Rev 0: Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring	II.6	Describe process used to develop, review, verify, validate and audit digital computer software.	Partially Conforms	Applicable	This acceptance criterion is applicable except for site- specific, programmatic aspects regarding software reviews.	11.6 Ch 17
NSDSRS 11.6, Rev 0: Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring	II.7	RETS/SREC and ODCM established setpoints.	Not Applicable	Applicable	The RETS/SREC and ODCM are COL applicant responsibilities.	Not Applicable
NSDSRS 11.6, Rev 0: Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring	II.8	Compliance with 10 CFR 20.1406 via RG 4.21, NEI 97-06, 08-08A and 07-07.	Partially Conforms	Applicable	See RG 4.21 in Table 1.9-2.	11.5 11.6 12.3.6

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSDSRS 11.6, Rev 0: Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring	II.9	Description of design features and instrumentation used in primary and secondary coolant system leakage detection	Conforms	Applicable	None.	5.2.5 9.3.4 9.3.6 11.5 11.6
NSDSRS 11.6, Rev 0: Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring	II.10	Additional information on operating experience	Conforms	Applicable	None.	11.5 11.6 12.3
NSDSRS 11.6, Rev 0: Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring	II.11	Radiation monitoring and sampling conformance to Tech Specs, Initial Test Program, and ITAAC.	Conforms	Applicable	None.	13.4 14.2 14.3 Ch 16
NSDSRS 11.6, Rev 0: Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring	II.12	Describe the types and ranges of radiation monitoring equipment	Conforms	Applicable	None.	11.5 11.6 12.3

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSDSRS 11.6, Rev 0: Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring	II.13	Reactor fuel storage area monitors	Conforms	Applicable	None.	7.2 11.5 11.6 12.3.4
BTP 11-3, Rev 4: Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water- Cooled Nuclear Power Reactor Plants	B.1	Processing Requirements	Conforms	Applicable	This guidance is applicable except for aspects related to PCP development and implementation that are applicable to COL applicants.	11.4.1 11.4.2
BTP 11-3, Rev 4: Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water- Cooled Nuclear Power Reactor Plants	B.2	Assurance of Complete Stabilization or Dewatering	Not Applicable	Applicable	This guidance is related to PCP development and implementation that are applicable to COL applicants.	Not Applicable
BTP 11-3, Rev 4: Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water- Cooled Nuclear Power Reactor Plants	B.3	Waste Storage	Conforms	Applicable	None.	11.4.1 11.4.2
BTP 11-3, Rev 4: Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water- Cooled Nuclear Power Reactor Plants	B.4	Portable Solid Waste Systems	Partially Conforms	Applicable	This guidance is applicable except for aspects related to control and use of portable solid radwaste processing equipment that are applicable to COL applicants.	11.4.1 11.4.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
BTP 11-3, Rev 4: Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants	B.5	Additional Design Features	Partially Conforms	Applicable	This guidance is applicable except for aspects related to PCP development and implementation that are applicable to COL applicants.	11.4.2
BTP 11-5, Rev 4: Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure	B.1	Waste Gas System Leak or Failure Analysis	Partially Conforms	Applicable	This acceptance criterion is applicable except for aspects that are BWR-specific or are site-specific.	11.3.1 11.3.3
BTP 11-5, Rev 4: Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure	B.2	Staff Method for Analysis	Conforms	Applicable	None.	11.3.3
BTP 11-6, Rev 4: Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	B.1	Failure Mechanism and Radioactivity Releases	Not Applicable	Applicable	COL applicant.	11.2.3
BTP 11-6, Rev. 4: Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	B.2	Mitigating Design Features	Not Applicable	Applicable	COL applicant.	11.2.3
BTP 11-6, Rev 4: Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	B.3	Radioactive Source Term	Not Applicable	Applicable	This acceptance criterion is applicable except for aspects that are BWR specific or are related to site-specific activities that are the responsibility of the COL applicant.	11.2.3
BTP 11-6, Rev 4: Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	B.4	Calculations of Transport Capabilities in Groundwater or Surface Water	Not Applicable	Applicable	The development of representative site parameters under this acceptance criterion (and SRP Section 2.4.13) is site-specific and thus applicable to COL applicant.	Not Applicable
BTP 11-6, Rev 4: Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	B.5	Exposure Scenarios and Acceptance Criteria	Not Applicable	Applicable	The development of representative site parameters under this acceptance criterion is site-specific and thus applicable to COL applicant.	11.2.3

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
BTP 11-6, Rev 4: Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	B.6	SRP Dose Acceptance Criteria	Not Applicable	Applicable	This acceptance criterion is applicable except for aspects that are site-specific that are the responsibility of the COL applicant.	11.2.3
BTP 11-6, (Rev 4): Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	B.7	Specifications on Tank Waste Radioactivity Concentration Levels	Not Applicable	Applicable	This guidance is the responsibility of the COL applicant.	11.2.3
SRP 12.1, Rev 4: Assuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable	II.1	Policy Considerations	Partially Conforms	Applicable	The programmatic and operational aspects of this guidance are not applicable to the DCA. These site-specific aspects are the responsibility of the COL applicant referencing the certified design.	12.1.1
SRP 12.1, Rev 4: Assuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable	II.2	Design Considerations	Conforms	Applicable	None.	12.1.2
SRP 12.1, Rev 4: Assuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable	II.3	Operational Considerations	Not Applicable	Applicable	This guidance governs site-specific operational programs, plans, and procedures that are the responsibility of the COL applicant. Consistent with SRP Section 1.0, Appendix A; RG 1.206, Regulatory Position C.III.4; and ESP/DC/ COL-ISG-015, the DCA contains COL information items that describe the information that is deferred to the COL applicant referencing the certified design.	12.1.3
SRP 12.1, Rev 4: Assuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable	II.4	Radiation Protection Considerations	Not Applicable	Applicable	See comment above for Acceptance Criterion II.3.	12.1.3
NSDSRS 12.2, Rev 0: Radiation Sources	II.1	RG 1.183	Partially Conforms	Applicable	See RG 1.183 in Table 1.9-2.	12.2.1



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 12.2, Rev 0: Radiation Sources	II.2	RG 1.7	Not Applicable	Not Applicable	See RG 1.7 in Table 1.9-2. There is no radiation source created from the determination of gaseous concentrations in containment following an accident (such as sample lines outside containment).	Not Applicable
NSDSRS 12.2, Rev 0: Radiation Sources	II.3	RG 1.112	Partially Conforms	Applicable	See RG 1.112 in Table 1.9-2.	12.2.1
NSDSRS 12.2, Rev 0: Radiation Sources	II.4	NUREG-0737, Task Action Plan Item II.B.2	Conforms	Applicable	None.	12.3 12.4
NSDSRS 12.2, Rev 0: Radiation Sources	II.5	ANSI/ANS Standard 18.1	Conforms	Applicable	None.	11.1
NSDSRS 12.2, Rev 0: Radiation Sources	II.6	Radiation Sources for 10 CFR 50.49 (EQ)	Conforms	Applicable	None.	12.2 CH 3
NSDSRS 12.2, Rev 0: Radiation Sources	II.7	RG 1.143	Partially Conforms	Applicable	See RG 1.143 in Table 1.9-2.	11.2 11.3 11.4 11.6
NSDSRS 12.2, Rev 0: Radiation Sources	II.8	RG 1.26, RG 1.29 and RG 1.117	Conforms	Applicable	None.	3.2
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.1	RG 1.7	Not Applicable	Not Applicable	See RG 1.7 in Table 1.9-2. There is no radiation field created from the determination of gaseous concentrations in containment following an accident (such as sample lines outside containment).	Not Applicable
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.2	RG 1.52	Not Applicable	Not Applicable	See RG 1.52 in Table 1.9-2.	Not Applicable
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.3	RG 1.69	Partially Conforms	Applicable	See RG 1.69 in Table 1.9-2.	12.3.2
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.4	RG 1.97	Partially Conforms	Applicable	See RG 1.97 in Table 1.9-2.	7.2.13 12.3.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.5	RG 1.183	Partially Conforms	Applicable	See RG 1.183 in Table 1.9-2.	12.2
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.6	RG 8.2	Not Applicable	Applicable	See RG 8.2 in Table 1.9-2.	Not Applicable
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.7	RG 8.8	Partially Conforms	Applicable	See RG 8.8 in Table 1.9-2.	12.3.1
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.8	RG 8.10	Not Applicable	Applicable	See RG 8.10 in Table 1.9-2.	Not Applicable
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.9	RG 8.15	Not Applicable	Applicable	See RG 8.15 in Table 1.9-2.	Not Applicable
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.10	RG 8.19	Conforms	Applicable	None.	12.4
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.11	RG 8.25	Not Applicable	Applicable	See RG 8.25 in Table 1.9-2.	12.3.4
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.12	RG 8.38	Partially Conforms	Applicable	See RG 8.38 in Table 1.9-2.	12.3.1
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.13	ANSI/ANS/HPSSC-6.8.1-1981	Conforms	Applicable	None.	12.3.4
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.14	ANSI/HPS N13.1-2011	Conforms	Applicable	None.	12.3.4
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.15	ANSI/ANS-6.4-2006	Conforms	Applicable	None.	12.3.2
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.16	Memo from Larry W. Camper to David B. Matthews and Elmo E. Collins dated 10-10-2006	Partially Conforms	Applicable	The portion of this guidance that pertains to the design phase is applicable to the DCA.	12.3.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.17	RG 1.140	Partially Conforms	Applicable	See RG 1.140 in Table 1.9-2.	12.3.3
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.18	RG 1.89	Partially Conforms	Applicable	See RG 1.89 in Table 1.9-2.	3.11
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.19	RG 4.21	Partially Conforms	Applicable	See RG 4.21 in Table 1.9-2.	12.3.6
NSDSRS 12.3-12.4, Radiation Protection Design Features	II.20	RG1.45	Partially Conforms	Applicable	See RG 1.45 in Table 1.9-2.	5.2.5
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.21	NEI 97-06	Conforms	Applicable	None.	CH 5
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.22	RG 1.143	Partially Conforms	Applicable	See RG 1.143 in Table 1.9-2.	CH 11
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.23	BTP 11-3 and SECY-94-198	Partially Conforms	Applicable	These guidance documents are not applicable to the DCA so far as they address the addition of supplemental extended LLW storage and the development of a PCP. This is a COL applicant responsibility.	11.4
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.24	RG 1.97	Partially Conforms	Applicable	See RG 1.97 in Table 1.9-2.	12.3.4
NSDSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.25	RG 1.12	Partially Conforms	Applicable	See RG 1.12 in Table 1.9-2.	12.3.1
NSDSRS 12.5, Rev 0: Operational Radiation Protection Program	All	Various	Not Applicable	Applicable	This guidance governs operational programs, procedures, facilities and organization that are site-specific, and are the responsibility of the COL applicant referencing the certified design.	12.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 13.1.1, Rev 5: Management and Technical Support Organization	All	General and Specific Requirements	Not Applicable	Applicable	COL applicant.	13.1.1
SRP 13.1.2 - 13.1.3, Rev 6: Operating Organization	All	Operating Organization	Not Applicable	Applicable	COL applicant.	13.1.2 13.1.3
SRP 13.2.1, Rev 3: Reactor Operator Requalification Program; Reactor Operator Training	All	General and Specific Requirements	Not Applicable	Applicable	COL applicant.	13.2
SRP 13.2.2, Rev 3: Non- Licensed Plant Staff Training	All	Various	Not Applicable	Applicable	COL applicant.	13.2
SRP 13.3, Rev 3: Emergency Planning	II.1	Meeting the Standards of 10 CFR 50.47(b); Conduct of Full Participation Exercise per 10 CFR 50, Appendix E	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.2	Onsite and Offsite Emergency Response Plans	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.3	Emergency Classification and Action Level Scheme	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.4	Meteorological Criteria	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.5	Upgrading Emergency Response Facilities	Not Applicable	Not Applicable	There are no proposed changes to existing emergency response facilities.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.6	Alerting and Notifications	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.7	Protective Action Recommendations	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.8	Alternatives to NUREG-0654/ FEMA-REP-1, Rev 1,	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.9	State, Tribal, and Local Government Planning and Preparedness	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.10	Emergency Planning Zones	Not Applicable	Applicable	COL applicant.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 13.3, Rev 3: Emergency Planning	II.11	Evacuation Time Estimates	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.12	Emergency Response Data System	Partially Conforms	Applicable	COL applicant.	13.3
SRP 13.3, Rev 3: Emergency Planning	II.13	Acceptability of Emergency Plans	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.14	Offsite Emergency Planning When Local Governments Decline to Participate	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.15	Early Site Permit Criteria - Physical Characteristics Unique to Proposed Site	Not Applicable	Not Applicable	Early site permit.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.16	Early Site Permit Criteria - Preliminary Analysis of Evacuation Times	Not Applicable	Not Applicable	Early site permit.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.17	Physical Characteristics Unique to Proposed Site	Not Applicable	Not Applicable	Early site permit.	13.3
SRP 13.3, Rev 3: Emergency Planning	II.18	Copies of Letters of Agreement or Other Certifications	Not Applicable	Not Applicable	Early site permit.	13.3
SRP 13.3, Rev 3: Emergency Planning	II.19	Emergency Preparedness Information and Plans Associated with Early Site Permit Application	Not Applicable	Not Applicable	Early site permit.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.20	Complete and Integrated Emergency Plans Associated with Early Site Permit Application	Not Applicable	Not Applicable	Early site permit.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.21	ITAAC Associated with Early Site Permit Application	Not Applicable	Not Applicable	Early site permit.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.22	ITAAC Associated with Design Certification Application	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.23	ITAAC Associated with Combined License Application	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.24	Generic Emergency Planning ITAAC	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.25	Design and Implementation of Emergency Response Facilities	Partially Conforms	Applicable	COL applicant.	13.3
SRP 13.3, Rev 3: Emergency Planning	II.26	Safety Parameter Display System	Not Applicable	Applicable	COL applicant (safety display and indication system).	13.3

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 13.3, Rev 3: Emergency Planning	II.27	Reactor Coolant System and Containment Sampling	Partially Conforms	Applicable	COL applicant.	9.3.2
SRP 13.3, Rev 3: Emergency Planning	II.28	Containment Monitoring and Continuous Sampling from Potential Accident Release Points	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.29	NRC Notifications and Communications	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.30	Generic Communications and Commission Orders Pertaining to Emergency Planning	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.31	Operational Programs	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.4, Rev 3: Operational Programs	Not Applicable	Various (Including Attachment, Sample FSAR Table 13.4-x)	Not Applicable	Not Applicable	There are no specific requirements for this SRP section.	Not Applicable
SRP 13.5.1.1, Rev 1: Administrative Procedures - General	All	Various	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.5.1.2, Draft Rev 0: Administrative Procedures - Initial Test Program	All	Various	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.5.2.1, Rev 2: Operating and Emergency Operating Procedures	All	Various	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.5.2.2, Draft Rev 0: Maintenance and Other Operating Procedures	All	Various	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.6.1, Rev 1: Physical Security - Combined License and Operating Reactors	All	Various	Not Applicable	Applicable	COL applicant.	Not Applicable
SRP 13.6.2, Rev 2: Physical Security - Design Certification	All	Various	Conforms	Applicable	Applicable for the physical security elements within the certified design boundary of the NuScale plant.	13.6.2 (via Security Technical Report)
SRP 13.6.3, Rev 1: Physical Security - Early Site Permit	All	Various	Not Applicable	Not Applicable	ESP applicant.	Not Applicable
SRP 13.6.4, Rev 1: Access Authorization	II (no number)	10 CFR 73.56	Not Applicable	Applicable	COL applicant.	13.6.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 13.6.6, Rev 0: Cyber Security Plan	All	Various	Not Applicable	Applicable	COL applicant.	13.6.3
SRP 13.7.1, Rev 0: Fitness for Duty (Operational)	All	Various	Not Applicable	Applicable	COL applicant.	13.7.1
SRP 13.7.2, Rev 0: Fitness for Duty (Construction)	All	Various	Not Applicable	Applicable	COL applicant.	13.7.2
NSDSRS 14.2, Rev 0: Initial Plant Test Program - Design Certification and New COL applicants	II.1	Summary of Test Program and Objectives	Conforms	Applicable	None.	14.2
NSDSRS 14.2, Rev 0: Initial Plant Test Program - Design Certification and New COL applicants	II.2	Test Programs Conformance with Regulatory Guides	Conforms	Applicable	None.	14.2
NSDSRS 14.2, Rev 0: Initial Plant Test Program - Design Certification and New COL applicants	II.3	Initial Test Program Administrative Procedures	Partially Conforms	Applicable	Subheading DC Applicant, Items A through D, are applicable to the DCA. Subheading COL/OL applicants, Items A through H, are applicable only to COL applicant.	14.2
NSDSRS 14.2, Rev 0: Initial Plant Test Program - Design Certification and New COL applicants	II.4	Initial Startup Tests	Partially Conforms	Applicable	Subheading DC Applicant, Item A, is applicable to the DCA. Subheading COL/OL applicants, Items A and B, are applicable to COL applicants.	14.2
NSDSRS 14.2, Rev 0: Initial Plant Test Program - Design Certification and New COL applicants	II.5	Individual Test Descriptions/ Abstracts	Conforms	Applicable	None.	14.2
NSDSRS 14.2, Rev 0: Initial Plant Test Program - Design Certification and New COL applicants	II.6	Initial Test Program Acceptance Criteria	Partially Conforms	Applicable	Subheading DC Applicant, Items A through C, are applicable to the DCA. Subheading COL/OL applicants, Items A through C, are applicable to COL applicants.	14.2
SRP 14.2.1, (August 2006): Generic Guidelines for Extended Power Uprate Testing Programs	All	Various	Not Applicable	Not Applicable	This SRP section is applicable only to extended power uprate license amendment requests.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 14.3, (March 2007): Inspections, Tests, Analyses, and Acceptance Criteria	II.1	Acceptability of the Scope of ITAAC	Partially Conforms	Applicable	A portion of this acceptance criterion is applicable to COL applicants.	14.3
SRP 14.3, (March 2007): Inspections, Tests, Analyses, and Acceptance Criteria	II.2	Specific Acceptance Criteria for ITAAC Specified in SRP Section 14.3	Conforms	Applicable	None.	14.3
SRP 14.3.2, Rev 0: Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria	All	Various	Not Applicable	Applicable	Methodology for developing ITAAC is provided in SRP 14.3.	Not Applicable
SRP 14.3.3, (March 2007): Piping Systems and Components - Inspections, Tests, Analyses, and Acceptance Criteria	All	Various	Not Applicable	Applicable	Methodology for developing ITAAC is provided in SRP 14.3.	Not Applicable
SRP 14.3.4, Rev 0: Reactor Systems - Inspections, Tests, Analyses, and Acceptance Criteria	All	Various	Not Applicable	Applicable	Methodology for developing ITAAC is provided in SRP 14.3.	Not Applicable
NSDSRS 14.3.5, Rev 0: Instrumentation and Controls - Inspections, Tests, Analyses, and Acceptance Criteria	All	Various	Not Applicable	Applicable	Methodology for developing ITAAC is provided in SRP 14.3.	Not Applicable
SRP 14.3.6, Rev 0: Electrical Systems - Inspections, Tests, Analyses, and Acceptance Criteria	All	Various	Not Applicable	Applicable	Methodology for developing ITAAC is provided in SRP 14.3.	Not Applicable
SRP 14.3.7, Rev 0: Plant Systems - Inspections, Tests, Analyses, and Acceptance Criteria	All	Various	Not Applicable	Applicable	Methodology for developing ITAAC is provided in SRP 14.3.	Not Applicable



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 14.3.8, Rev 0: Radiation Protection - Inspections, Tests, Analyses, and Acceptance Criteria	All	Various	Not Applicable	Applicable	Methodology for developing ITAAC is provided in SRP 14.3.	Not Applicable
SRP 14.3.9, (March 2007): Human Factors Engineering - Inspections, Tests, Analyses, and Acceptance Criteria	All	Various	Not Applicable	Applicable	Methodology for developing ITAAC is provided in SRP 14.3.	Not Applicable
SRP 14.3.10, (March 2007): Emergency Planning - Inspections, Tests, Analyses, and Acceptance Criteria	All	Various	Not Applicable	Applicable	Methodology for developing ITAAC is provided in SRP 14.3.	Not Applicable
SRP 14.3.11, (March 2007): Containment Systems - Inspections, Tests, Analyses, and Acceptance Criteria	All	Various	Not Applicable	Applicable	Methodology for developing ITAAC is provided in SRP 14.3.	Not Applicable
SRP 14.3.12, Rev 1: Physical Security Hardware - Inspections, Tests, Analyses, and Acceptance Criteria	All	Various	Not Applicable	Applicable	Methodology for developing ITAAC is provided in SRP 14.3.	Not Applicable
NSDSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.1	Categorization of Transients and Accidents	Conforms	Applicable	None.	15.0
NSDSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.2	Categorization According to Frequency of Occurrence	Partially Conforms	Applicable	Events that have been historically classified as AOOs are not analyzed for frequency of occurrence. Some events that have an IE frequency are also deterministically classified as AOOs.	15.0
NSDSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.3	Categorization According to Type	Conforms	Applicable	None.	15.0

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.4.A	Analysis Acceptance Criteria for AOOs	Conforms	Applicable	None.	15.0.0
NSDSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.4.B	Analysis Acceptance Criteria for IEs and Postulated Accidents	Partially Conforms	Applicable	The guidance is applicable except for 4.B.ii and 4.B.iv. CHF, not DNBR is used to determine the thermal margin for the fuel cladding. LOCA acceptance criteria uses an acceptance criterion that is more restrictive than the temperature limit of 2,200°F.	15.0.0
NSDSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.5	Plant Characteristics Considered in the Safety Evaluation	Conforms	Applicable	None.	15.0
NSDSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.6	Assumed Protection and Safety Systems Actions	Conforms	Applicable	None.	15.0
NSDSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.7	Evaluation of Individual Initiating Events	Conforms	Applicable	None.	15.0
NSDSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.8.A	Identification of Causes and Frequency Classification	Conforms	Applicable	None.	15.0
NSDSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.8.B	Sequence of Events and Systems Operation	Partially Conforms	Applicable	This acceptance criterion is applicable except for Item B.vi, which is applicable to COL applicants. Specifically, development and implementation of emergency operating procedures and emergency response procedures is the responsibility of the COL applicant referencing the NuScale design.	15.0
NSDSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.8.C	Core, System, and Barrier Performance	Partially Conforms	Applicable	The guidance is applicable except for aspects that are BWR-specific. NuScale evaluates critical heat flux (CHF), which is more applicable to the NuScale design than DNBR.	15.0.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 15.0.1, Rev 0: Radiological Consequence Analyses Using Alternative Source Terms	II (No number)	First full paragraph and 6 bullets on Page 15.0.1-6, Compliance with Specific Provisions of NUREG-0737	Not Applicable	Not Applicable	The NuScale design utilizes a modified version of the alternative source term (AST) methodology to evaluate radiological consequences of accidents.	15.0.3
SRP 15.0.1, Rev 0: Radiological Consequence Analyses Using Alternative Source Terms	II (No number)	Last paragraph on Page 15.0.1-6 and Table 1, Exposure Criteria for Radiological Consequences of Design Basis Accident	Not Applicable	Not Applicable	The NuScale design utilizes a modified version of the AST methodology to evaluate radiological consequences of accidents.	15.0.3
SRP 15.0.2, (March 2007): Review of Transient and Accident Analysis Methods	II.1	Evaluation Model	Conforms	Applicable	NuScale ECCS evaluation models for LOCAs only address technically relevant features required by Appendix K.	15.0.2
SRP 15.0.2, (March 2007): Review of Transient and Accident Analysis Methods	II.2	Accident Scenario Identification Process	Conforms	Applicable	None.	15.0.2
SRP 15.0.2, (March 2007): Review of Transient and Accident Analysis Methods	II.3	Code Assessment	Conforms	Applicable	NuScale ECCS evaluation models for LOCAs only assess the technically relevant features required by Appendix K and TMI Action Item II.K3.30.	15.0.2
SRP 15.0.2, (March 2007): Review of Transient and Accident Analysis Methods	II.4	Uncertainty Analysis	Conforms	Applicable	Non-LOCA methods use sensitivity analyses or bounding values to determine input parameters.	15.0.2
SRP 15.0.2, (March 2007): Review of Transient and Accident Analysis Methods	II.5	Quality Assurance Plan	Conforms	Applicable	None.	15.0.2
NSDSRS 15.0.3, Rev 0: Design Basis Accident Radiological Consequence Analyses for the NuScale SMR Design	II.1	Offsite Radiological Consequences of Postulated DBAs (includes Table 1)	Conforms	Applicable	None.	15.0.3

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 15.0.3, Rev 0: Design Basis Accident Radiological Consequence Analyses for the NuScale SMR Design	II.2	Control Room Radiological Habitability	Conforms	Applicable	None.	6.4 9.4 13.3 15.0.3
NSDSRS 15.0.3, Rev 0: Design Basis Accident Radiological Consequence Analyses for the NuScale SMR Design	II.3	Technical Support Center Radiological Habitability	Partially Conforms	Applicable	Dose acceptance criterion met for TSC when AC power is available. TSC function transferred to MCR when AC is not available.	15.0.3
NSDSRS 15.1.1-15.1.4, Rev 0: Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of the Turbine Bypass System or Inadvertent Operation of the Decay Heat Removal System	II.1 Basic Objective	Identify Limiting Increase in Heat Removal Events	Conforms	Applicable	None.	15.1.1-15.1.4
NSDSRS 15.1.1-15.1.4, Rev 0: Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of the Turbine Bypass System or Inadvertent Operation of the Decay Heat Removal System	II.2 Basic Objective	Verify Fuel Damage and System Pressure Criteria are Met for Limiting Event.	Conforms	Applicable	None.	15.1.1-15.1.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 15.1.1-15.1.4, Rev 0: Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of the Turbine Bypass System or Inadvertent Operation of the Decay Heat Removal System	II.1 Specific Criterion	System Pressure	Conforms	Applicable	None.	15.1.1-15.1.4
NSDSRS 15.1.1-15.1.4, Rev 0: Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of the Turbine Bypass System or Inadvertent Operation of the Decay Heat Removal System	II.2 Specific Criterion	MCHFR Remains Above 95/95 Limit	Conforms	Applicable	NuScale has determined that critical heat flux more accurately describes plant phenomena than departure from nucleate boiling.	15.1.1-15.1.4
NSDSRS 15.1.1-15.1.4, Rev 0: Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of the Turbine Bypass System or Inadvertent Operation of the Decay Heat Removal System	II.3 Specific Criterion	AOOs Should Not Generate More Serious Condition	Conforms	Applicable	NuScale events are classified by AOO, IE, accident, and special event; but will conform with SRP requirement that incidents of moderate frequency should not generate a more serious plant condition without other faults occurring independently.	15.1.1-15.1.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 15.1.1-15.1.4, Rev 0: Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of the Turbine Bypass System or Inadvertent Operation of the Decay Heat Removal System	II.4 Specific Criterion	Instrument Spans and Setpoints use RG 1.105	Conforms	Applicable	None.	15.1.1-15.1.4
NSDSRS 15.1.1-15.1.4, Rev 0: Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of the Turbine Bypass System or Inadvertent Operation of the Decay Heat Removal System	II.5 Specific Criterion	Identify Limiting Single Failure	Conforms	Applicable	None.	15.1.1-15.1.4
NSDSRS 15.1.1-15.1.4, Rev 0: Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of the Turbine Bypass System or Inadvertent Operation of the Decay Heat Removal System	II.1 Analytical Parameters	Initial Power Level is 102%	Conforms	Applicable	None.	15.1.1-15.1.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSDSRS 15.1.1-15.1.4, Rev 0: Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of the Turbine Bypass System or Inadvertent Operation of the Decay Heat Removal System	II.2 Analytical Parameters	Conservative Scram Used	Conforms	Applicable	None.	15.1.1-15.1.4
NSDSRS 15.1.1-15.1.4, Rev 0: Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of the Turbine Bypass System or Inadvertent Operation of the Decay Heat Removal System	II.3 Analytical Parameters	Core Burnup	Conforms	Applicable	None.	15.1.1-15.1.4
NSDSRS 15.1.1-15.1.4, Rev 0: Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of the Turbine Bypass System or Inadvertent Operation of the Decay Heat Removal System	II.4 Analytical Parameters	Setpoint Inaccuracies use guidance in RG 1.105	Conforms	Applicable	None.	15.1.1-15.1.4
NSDSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.1 Specific Criteria	Reactor Coolant and Main Steam System Pressure	Conforms	Applicable	None.	15.1.5
NSDSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.2 Specific Criteria	Evaluation of Core Damage Potential	Conforms	Applicable	NuScale has determined that critical heat flux more accurately describes plant phenomena than departure from nucleate boiling.	15.1.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.3 Specific Criteria	Radiological Criteria for Steam Line Breaks	Conforms	Applicable	None.	15.1.5
NSDSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.4 Specific Criteria	Safety-Related Classification and Auto-Initiation of Decay Heat Removal System	Conforms	Applicable	None.	15.1.5
NSDSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.1 Assumptions	Initial Power Level and Plant Operating Mode	Conforms	Applicable	None.	15.1.5
NSDSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.2 Assumptions	Loss of Offsite Power	Conforms	Applicable	None.	15.1.5
NSDSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.3 Assumptions	Postulated Steam Line Break Effects	Conforms	Applicable	None.	15.1.5
NSDSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.4 Assumptions	Worst Case Failure of Single Active Component	Conforms	Applicable	None.	15.1.5
NSDSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.5 Assumptions	Maximum-Worth Rod Fully Withdrawn	Conforms	Applicable	None.	15.1.5
NSDSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.6 Assumptions	Core Burnup	Conforms	Applicable	None.	15.1.5
NSDSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.7 Assumptions	Initial Core Flow	Conforms	Applicable	NuScale has determined that critical heat flux more accurately describes plant phenomena than departure from nucleate boiling.	15.1.5



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSDSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.8 Assumptions	Postulated Failure of Non-Seismic Main Steam Line	Conforms	Applicable	None.	15.1.5
NSDSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.9 Assumptions	Postulated Failure of Seismic Main Steam Line	Conforms	Applicable	None.	15.1.5
NSDSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.10 Assumptions	Limiting Consequence Assessment When Operator Action is Credited	Not Applicable	Not Applicable	Operator Action is not required to mitigate the consequences of a SLB.	Not Applicable
SRP 15.1.5.A, Rev 2: Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR	All	Various	Partially Conforms	Applicable	Per SRP Section 15.0.3, Section I, Areas of Review, Item 10 under subheading Review Interfaces, for the review of design certification applications, SRP Section 15.0.3 supersedes the radiological analyses, assumptions, acceptance criteria, and methodologies identified in this SRP Section 15.1.5, Appendix A. Provisions related to the nonradiological analyses aspects of this SRP Section 15.1.5, Appendix A, remain applicable to the DCA.	15.0.3
SRP 15.1.5.A, Rev 2: Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR	II (No number)	First full paragraph and Items 1 and 2 on Page 15.1.5-11, Exposure Guidelines for Calculated Doses	Partially Conforms	Applicable	The part of this guidance specifying the calculation of radiological consequences of a postulated main steam line break outside containment is applicable to the DCA. However, per SRP Section 15.0.3, Section I, Areas of Review, Item 10.A under subheading Review Interfaces, the part of this acceptance criterion that specifies radiological acceptance criteria is superseded by SRP Section 15.0.3.	15.0.3

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.1.5.A, Rev 2: Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR	II (No number)	First full paragraph following Items 1 and 2 on Page 15.1.5-11, Methodology and Assumptions for Calculating Radiological Consequences	Not Applicable	Not Applicable	This acceptance criterion specifies radiological analysis methodology and assumptions that are superseded by SRP Section 15.0.3.	Not Applicable
SRP 15.1.5.A, Rev 2: Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR	II (No number)	Second full paragraph following Items 1 and 2 on Page 15.1.5-11, Technical Specifications for Assumed Iodine Activity and Primary-to-Secondary Leak Rate	Partially Conforms	Applicable	The part of this guidance related to the required technical specification for primary and secondary coolant iodine activity and primary-to-secondary leak rate is applicable to the DCA. However, per SRP Section 15.0.3, Section I, Areas of Review, Item 10.A under subheading Review Interfaces, the part of this acceptance criterion that specifies radiological acceptance criteria is superseded by SRP Section 15.0.3.	15.0.3
NSDSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.1 Specific Criteria	Reactor Coolant Pressure	Conforms	Applicable	None.	15.1.6
NSDSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.2 Specific Criteria	Cladding Integrity	Conforms	Applicable	None.	15.1.6
NSDSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.3 Specific Criteria	AOOs Should Not Generate More Serious Condition	Conforms	Applicable	None.	15.1.6
NSDSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.4 Specific Criteria	Instruments Spans and Setpoints use RG 1.105	Conforms	Applicable	None.	15.1.6
NSDSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.5 Specific Criteria	Identify Limiting Single Failure	Conforms	Applicable	None.	15.1.6
NSDSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.1 Analytical Parameters	Initial Power Level is 102%	Conforms	Applicable	None.	15.1.6
NSDSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.2 Analytical Parameters	Conservative Scram Used	Conforms	Applicable	None.	15.1.6
NSDSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.3 Analytical Parameters	Core Burnup	Conforms	Applicable	None.	15.1.6
NSDSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.4 Analytical Parameters	Maximize Heat Transfer from RCS to Containment and Reactor Pool	Conforms	Applicable	None.	15.1.6
NSDSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.5 Analytical Parameters	Setpoint Inaccuracies use Guidance in RG 1.105	Conforms	Applicable	None.	15.1.6

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSDSRS 15.2.1-15.2.5, Rev 0: Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)	II.1	Basic Objectives - Initiating Events	Conforms	Applicable	None.	15.2.1-15.2.5
NSDSRS 15.2.1-15.2.5, Rev 0: Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)	II.2	Specific Criteria for Events of Moderate Frequency	Partially Conforms	Applicable	NuScale does not have a steam pressure regulator.	15.2.1-15.2.5
NSDSRS 15.2.1-15.2.5, Rev 0: Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)	II.2.A	Reactor Coolant System and Main Steam System Pressures	Conforms	Applicable	None.	15.2.1-15.2.5
NSDSRS 15.2.1-15.2.5, Rev 0: Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)	II.2.B	Fuel Cladding Integrity	Conforms	Applicable	NuScale does not have a steam pressure regulator.	15.2.1-15.2.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSDSRS 15.2.1-15.2.5, Rev 0: Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)	II.2.C	Incidents of Moderate Frequency	Conforms	Applicable	None.	15.2.1-15.2.5
NSDSRS 15.2.1-15.2.5, Rev 0: Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)	II.2.D	Instrument Setpoints - Impact on Plant Response	Conforms	Applicable	None.	15.2.1-15.2.5
NSDSRS 15.2.1-15.2.5, Rev 0: Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)	II.2.E	Most Limiting Plant System Single Failure	Conforms	Applicable	None.	15.2.1-15.2.5
NSDSRS 15.2.1-15.2.5, Rev 0: Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)	II.2.F	Performance of Nonsafety-Related Systems and Single Failures of Active and Passive Systems	Conforms	Applicable	None.	15.2.1-15.2.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSDSRS 15.2.1-15.2.5, Rev 0: Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)	II.3	Analytical Model	Conforms	Applicable	None.	15.2.1-15.2.5
NSDSRS 15.2.1-15.2.5, Rev 0: Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)	II.3.A	Values of Parameters Used in Analytical Model - Initial Power Level and Modes of Operation	Conforms	Applicable	None.	15.2.1-15.2.5
NSDSRS 15.2.1-15.2.5, Rev 0: Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)	II.3.B	Values of Parameters Used in Analytical Model - Scram Characteristics	Conforms	Applicable	None.	15.2.1-15.2.5
NSDSRS 15.2.1-15.2.5, Rev 0: Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)	II.3.C	Values of Parameters Used in Analytical Model - Core Burnup	Conforms	Applicable	None.	15.2.1-15.2.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 15.2.1-15.2.5, Rev 0: Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)	II.3.D	Values of Parameters Used in Analytical Model - Instrumentation Setpoints for Mitigating System Actuation	Conforms	Applicable	None.	15.2.1-15.2.5
NSDSRS 15.2.6, Rev 0: Loss of Nonemergency AC Power to the Station Auxiliaries	II.1	Reactor Coolant and Main Steam System Pressures	Conforms	Applicable	None.	15.2.6
NSDSRS 15.2.6, Rev 0: Loss of Nonemergency AC Power to the Station Auxiliaries	II.2	Fuel Cladding Integrity	Conforms	Applicable	None.	15.2.6
NSDSRS 15.2.6, Rev 0: Loss of Nonemergency AC Power to the Station Auxiliaries	II.3	Incidents of Moderate Frequency	Conforms	Applicable	NuScale categorizes events as AOO and IE.	15.2.6
NSDSRS 15.2.6, Rev 0: Loss of Nonemergency AC Power to the Station Auxiliaries	II.4	Requirements of GDC 10 and GDC 15	Conforms	Applicable	None.	15.2.6
NSDSRS 15.2.6, Rev 0: Loss of Nonemergency AC Power to the Station Auxiliaries	II.5	Most Limiting Plant System Single Failure	Conforms	Applicable	None.	15.2.6
NSDSRS 15.2.6, Rev 0: Loss of Nonemergency AC Power to the Station Auxiliaries	II.5 A-D	Analysis of Loss of AC Power - Analytical Model and Methods, conservative assumptions and RG 1.105	Conforms	Applicable	None.	15.2.6
NSDSRS 15.2.7, Rev 0: Loss of Normal Feedwater Flow	II.1	Fuel and System Pressure Parameters met	Conforms	Applicable	None.	15.2.7
NSDSRS 15.2.7, Rev 0: Loss of Normal Feedwater Flow	II.2	Events of Moderate Frequency	Conforms	Applicable	NuScale categorizes events as AOO and IE.	15.2.7
NSDSRS 15.2.7, Rev 0: Loss of Normal Feedwater Flow	II.3	Analytical Model and Methods	Conforms	Applicable	None.	15.2.7

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 15.2.8, Rev 0: Feedwater System Pipe Break Inside and Outside Containment (PWR)	II.1	Reactor Coolant System and Main Steam System Pressures	Conforms	Applicable	None.	15.2.8
NSDSRS 15.2.8, Rev 0: Feedwater System Pipe Break Inside and Outside Containment (PWR)	II.2	Evaluation of Core Damage Potential	Conforms	Applicable	For slower reactivity insertions, NuScale uses a heat generation rate limit to ensure that fuel centerline melting limits are met.	15.2.8
NSDSRS 15.2.8, Rev 0: Feedwater System Pipe Break Inside and Outside Containment (PWR)	II.3	Calculated Site Boundary Doses	Conforms	Applicable	None.	15.2.8
NSDSRS 15.2.8, Rev 0: Feedwater System Pipe Break Inside and Outside Containment (PWR)	II.4	DHRS must be safety grade and automatically initiated when required.	Conforms	Applicable	None.	15.2.8
NSDSRS 15.2.8, Rev 0: Feedwater System Pipe Break Inside and Outside Containment (PWR)	II.5	Assumptions for Initial Plant Conditions and Postulated Failures	Conforms	Applicable	None.	15.2.8
SRP 15.3.1-15.3.2, Rev 2: Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	All	Various	Not Applicable	Not Applicable	Applicable only to LWR designs that rely on forced reactor coolant flow for core cooling. The NuScale design uses passive natural circulation of the primary coolant, eliminating the need for reactor coolant pumps.	Not Applicable
SRP 15.3.3-15.3.4, Rev 2: Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	All	Various	Not Applicable	Not Applicable	Section 15.3.3 - 15.3.4 are applicable only to LWR designs that rely on forced reactor coolant flow for core cooling. The NuScale design uses passive natural circulation of the primary coolant, eliminating the need for reactor coolant pumps.	Not Applicable
SRP 15.4.1, Rev 3: Uncontrolled Control Rod Assembly Withdrawal From a Subcritical or Low Power Startup Condition	II.1.A	Thermal Margin Limits	Conforms	Applicable	CHF is more appropriate terminology for NuScale phenomena than DNB.	15.4.1

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.4.1, Rev 3: Uncontrolled Control Rod Assembly Withdrawal From a Subcritical or Low Power Startup Condition	II.1.B	Fuel Centerline Temperatures	Conforms	Applicable	For slower reactivity insertions, NuScale uses a heat generation rate limit to ensure that fuel centerline melting limits are met.	15.4.1
SRP 15.4.1, Rev 3: Uncontrolled Control Rod Assembly Withdrawal From a Subcritical or Low Power Startup Condition	II.1.C	Uniform Cladding Strain	Not Applicable	Not Applicable	The SRP states that this criterion applies to BWRs. NuScale uses the 95/95 MCHFR approach to ensure no cladding or fuel failures.	15.4.1
SRP 15.4.2, Rev 3: Uncontrolled Control Rod Assembly Withdrawal at Power	II.1.A	Thermal Margin Limits	Conforms	Applicable	CHF is more appropriate terminology for NuScale phenomenon than DNB.	15.4.2
SRP 15.4.2, Rev 3: Uncontrolled Control Rod Assembly Withdrawal at Power	II.1.B	Fuel Centerline Temperatures	Conforms	Applicable	For slower reactivity insertions, NuScale uses a heat generation rate limit to ensure that fuel centerline melting limits are met.	15.4.2
SRP 15.4.2, Rev 3: Uncontrolled Control Rod Assembly Withdrawal at Power	II.1.C	Uniform Cladding Strain	Not Applicable	Not Applicable	The SRP states that this criterion applies to BWRs. NuScale uses the 95/95 MCHFR approach to ensure no cladding or fuel failures.	15.4.2
SRP 15.4.3, Rev 3: Control Rod Misoperation (System Malfunction or Operator Error)	II.1	Thermal Margin Limits	Conforms	Applicable	CHF is more appropriate terminology for NuScale phenomenon than DNB.	15.4.3
SRP 15.4.3, Rev 3: Control Rod Misoperation (System Malfunction or Operator Error)	II.2	Fuel Centerline Temperatures	Conforms	Applicable	For slower reactivity insertions, NuScale uses a heat generation rate limit to ensure that fuel centerline melting limits are met.	15.4.3
SRP 15.4.3, Rev 3: Control Rod Misoperation (System Malfunction or Operator Error)	II.3	Uniform Cladding Strain	Conforms	Applicable	None.	15.4.3



Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.4.4-15.4.5, Rev 2: Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	II.A	RCS and MSS Pressures	Not Applicable	Not Applicable	This guidance is not applicable because the specific language refers to PWR designs that use forced reactor coolant flow and have reactor coolant loops and pumps. The NuScale reactor design differs from that of typical PWRs in that the NuScale design does not require or include reactor coolant pumps. Thus, the specific AOOs that are addressed in this SRP section are not applicable to the NuScale design. The potential for a postulated startup reactivity accident (e.g., initiated by abnormal startup sequence) has been identified as an event requiring consideration for the NuScale reactor design. The guidance for these AOOs is available in NSDSRS 15.4.6 - Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR).	15.4.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.4.4-15.4.5, Rev 2: Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	II.B	Fuel Thermal Limits	Not Applicable	Not Applicable	This guidance is not applicable because the specific language refers to PWR designs that use forced reactor coolant flow and have reactor coolant loops and pumps. The NuScale reactor design differs from that of typical PWRs in that the NuScale design does not require or include reactor coolant pumps. Thus, the specific AOOs that are addressed in this SRP section are not applicable to the NuScale design. The potential for a postulated startup reactivity accident (e.g., initiated by abnormal startup sequence) has been identified as an event requiring consideration for the NuScale reactor design. The guidance for these AOOs is available in NSDSRS 15.4.6 - Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR).	15.4.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.4.4-15.4.5, Rev 2: Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	II.C	Events of Moderate Frequency	Not Applicable	Not Applicable	This guidance is not applicable because the specific language refers to PWR designs that use forced reactor coolant flow and have reactor coolant loops and pumps. The NuScale reactor design differs from that of typical PWRs in that the NuScale design does not require or include reactor coolant pumps. Thus, the specific AOOs that are addressed in this SRP section are not applicable to the NuScale design. The potential for a postulated startup reactivity accident (e.g., initiated by abnormal startup sequence) has been identified as an event requiring consideration for the NuScale reactor design. The guidance for these AOOs is available in NSDSRS 15.4.6 - Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR).	15.4.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.4.4-15.4.5, Rev 2: Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	II.D	Instrument Setpoints	Not Applicable	Not Applicable	This guidance is not applicable because the specific language refers to PWR designs that use forced reactor coolant flow and have reactor coolant loops and pumps. The NuScale reactor design differs from that of typical PWRs in that the NuScale design does not require or include reactor coolant pumps. Thus, the specific AOs that are addressed in this SRP section are not applicable to the NuScale design. The potential for a postulated startup reactivity accident (e.g., initiated by abnormal startup sequence) has been identified as an event requiring consideration for the NuScale reactor design. The guidance for these AOs is available in NSDSRS 15.4.6 - Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR).	15.4.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.4.4-15.4.5, Rev 2: Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	II.E	Single Failure	Not Applicable	Not Applicable	This guidance is not applicable because the specific language refers to PWR designs that use forced reactor coolant flow and have reactor coolant loops and pumps. The NuScale reactor design differs from that of typical PWRs in that the NuScale design does not require or include reactor coolant pumps. Thus, the specific AOOs that are addressed in this SRP section are not applicable to the NuScale design. The potential for a postulated startup reactivity accident (e.g., initiated by abnormal startup sequence) has been identified as an event requiring consideration for the NuScale reactor design. The guidance for these AOOs is available in NSDSRS 15.4.6 - Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR).	15.4.4

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.4.4-15.4.5, Rev 2: Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	II.F	Non-Safety Systems	Not Applicable	Not Applicable	This guidance is not applicable because the specific language refers to PWR designs that use forced reactor coolant flow and have reactor coolant loops and pumps. The NuScale reactor design differs from that of typical PWRs in that the NuScale design does not require or include reactor coolant pumps. Thus, the specific AOOs that are addressed in this SRP section are not applicable to the NuScale design. The potential for a postulated startup reactivity accident (e.g., initiated by abnormal startup sequence) has been identified as an event requiring consideration for the NuScale reactor design. The guidance for these AOOs is available in NSDSRS 15.4.6 - Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR).	15.4.4
SRP 15.4.6, Rev 2: Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR)	II.1	Reactor Coolant and Main Steam System Pressures	Conforms	Applicable	None.	15.4.6
SRP 15.4.6, Rev 2: Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR)	II.2	Fuel Cladding Integrity	Conforms	Applicable	CHF is more appropriate terminology for NuScale phenomenon.	15.4.6
SRP 15.4.6, Rev 2: Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR)	II.3	Incidents of Moderate Frequency	Conforms	Applicable	None.	15.4.6

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.4.6, Rev 2: Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR)	II.4	Minimum Time Intervals for Required Operator Actions	Not Applicable	Not Applicable	Operator Action is not required to mitigate an inadvertent boron dilution event.	15.4.6
SRP 15.4.6, Rev 2: Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR)	II.5	Analysis Model, Methods, and Assumptions	Conforms	Applicable	None.	15.4.6
SRP 15.4.7, Rev 2: Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	II.1	Provision in Plant Operating Procedures Requiring Instrumentation to Detect Fuel Loading Errors	Conforms	Applicable	None.	15.4.7
SRP 15.4.7, Rev 2: Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	II.2	Offsite Radiological Consequences	Conforms	Applicable	No fuel failure is expected.	15.4.7
SRP 15.4.8, Rev 3: Spectrum of Rod Ejection Accidents (PWR)	II.1	Availability of Monitoring Instrumentation	Conforms	Applicable	None.	15.4.8
SRP 15.4.8, Rev 3: Spectrum of Rod Ejection Accidents (PWR)	II.2	Effects of Postulated Reactivity Accidents	Conforms	Applicable	None.	15.4.8
SRP 15.4.8, Rev 3: Spectrum of Rod Ejection Accidents (PWR)	II.3	Radiation Dose Limits	Conforms	Applicable	None.	15.4.8

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.4.8.A, Rev 1: Radiological Consequences of a Control Rod Ejection Accident (PWR)	All	Various	Partially Conforms	Applicable	Per SRP Section 15.0.3, Section I, Areas of Review, Item 10 under subheading Review Interfaces, for the review of design certification applications, SRP Section 15.0.3 supersedes the radiological analyses, assumptions, acceptance criteria, and methodologies identified in SRP Section 15.4.8, Appendix A. Provisions related to the nonradiological analyses aspects of this SRP Section 15.4.8, Appendix A, remain applicable to the DCA.	15.0.3
SRP 15.4.8.A, Rev 1: Radiological Consequences of a Control Rod Ejection Accident (PWR)	II (No number)	First paragraph of Section II (bottom of Page 15.4.8-5 and top of Page 15.4.8-6) - Acceptability of Site and Dose Mitigating ESF	Partially Conforms	Applicable	The part of this guidance specifying the calculation of radiological consequences of a postulated control rod ejection accident is applicable to the DCA. However, per SRP Section 15.0.3, Section I, Areas of Review, Item 10.C under subheading Review Interfaces, the part of this acceptance criterion that specifies radiological acceptance criteria is superseded by SRP Section 15.0.3.	15.0.3
SRP 15.4.8.A, Rev 1: Radiological Consequences of a Control Rod Ejection Accident (PWR)	II (No number)	First full paragraph on Page 15.4.8-6) - Technical Specification for Primary-to-Secondary Leak Rate	Partially Conforms	Applicable	The part of this guidance related to the required technical specification for primary-to-secondary leak rate is applicable to the DCA. However, per SRP Section 15.0.3, Section I, Areas of Review, Item 10.C under subheading Review Interfaces, the part of this acceptance criterion that specifies radiological acceptance criteria is superseded by SRP Section 15.0.3.	15.0.3



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.4.8.A, Rev 1: Radiological Consequences of a Control Rod Ejection Accident (PWR)	II (No number)	Second full paragraph on Page 15.4.8-6) - Dose Model	Not Applicable	Not Applicable	Per SRP Section 15.0.3, Section I, Areas of Review, Item 10.C under subheading Review Interfaces, this acceptance criterion specifies radiological acceptance criteria and assumptions that are superseded by SRP Section 15.0.3.	Not Applicable
SRP 15.4.9, Rev 3: Spectrum of Rod Drop Accidents (BWR)	All	-	Not Applicable	Not Applicable	This SRP section and its acceptance criteria (II.1 through II.3) are applicable only to BWRs.	Not Applicable
SRP 15.4.9.A, Draft Rev 3: Radiological Consequences of Control Rod Drop Accident (BWR)	All	-	Not Applicable	Not Applicable	This SRP section and its acceptance criteria are applicable only to BWRs.	Not Applicable
NSDSRS 15.5.1-15.5.2, Rev 0: Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory	II.1	The frequency classification for this event is an AOO.	Conforms	Applicable	Acceptance Criteria for AOOs are specified in Section 15.0.	15.0 15.5.1-15.5.2
NSDSRS 15.5.1-15.5.2, Rev 0: Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory	II.2	The sequence of events, from initiation until a stabilized condition is reached including assumptions for equipment that operates, fails to operate or requires operator action.	Conforms	Applicable	None.	15.5.1-15.5.2
NSDSRS 15.5.1-15.5.2, Rev 0: Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory	II.3	Evaluation Model must be an approved model or be justified.	Conforms	Applicable	None.	15.5.1-15.5.2
NSDSRS 15.5.1-15.5.2, Rev 0: Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory	II.4.A	Input Parameters and Initial Conditions - Initial Power Level	Conforms	Applicable	None.	15.5.1-15.5.2

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 15.5.1-15.5.2, Rev 0: Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory	II.4.B	Input Parameters and Initial Conditions - Scram Characteristics	Conforms	Applicable	None.	15.5.1-15.5.2
NSDSRS 15.5.1-15.5.2, Rev 0: Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory	II.4.C	Input Parameters and Initial Conditions - Core Burnup	Conforms	Applicable	None.	15.5.1-15.5.2
SRP 15.6.1, Rev 2: Inadvertent Opening of a PWR Pressurizer Relief Valve or a BWR Pressure Relief Valve	All	Various	Partially Conforms	Applicable	This guidance is only applicable to LWRs that are designed with power-operated pressurizer relief valves. The NuScale design does not use power-operated relief valves (PORVs), which have the potential to open inadvertently. Rather, the NuScale design uses springloaded ASME code safety relief valves, which do not have the PORVs vulnerability to inadvertent operation. However, a mechanical failure of the reactor safety valve (RSV) is bounded by an inadvertent ECCS valve actuation, analyzed in Section 15.6.6.	15.6.1 15.6.6

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.6.2, Rev 2: Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	II (No Number)	Penultimate paragraph of Section II on Page 15.6.22 - Acceptability of Site and Dose Mitigating ESF Systems	Partially Conforms	Applicable	The part of this guidance specifying the calculation of radiological consequences of a postulated failure outside containment of a small reactor coolant line is applicable to the DCA. However, per SRP Section 15.0.3, Section I, Areas of Review, Item 10.E under subheading Review Interfaces, the part of this acceptance criterion that specifies radiological acceptance criteria is superseded by SRP Section 15.0.3.	15.0.3 15.6.2
SRP 15.6.2, Rev 2: Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	II (No Number)	Last paragraph of Section II on Page 15.6.22 - Plant-Specific Technical Specifications for Primary Coolant System Iodine Activity	Partially Conforms	Applicable	The part of this guidance related to the required technical specification for primary coolant iodine activity is applicable to the DCA. However, per SRP Section 15.0.3, Section I, Areas of Review, Item 10.E under subheading Review Interfaces, the part of this acceptance criterion that specifies radiological acceptance criteria is superseded by SRP Section 15.0.3.	15.0.3 15.6.2
SRP 15.6.3, Rev 2: Radiological Consequences of Steam Generator Tube Failure (PWR)	II (No Number)	First paragraph and Items (1) and (2) of Section II on Page 15.6.32 - Acceptability of Site and Dose Mitigating ESF Systems	Partially Conforms	Applicable	The part of this guidance specifying the calculation of radiological consequences of a postulated steam generator tube failure is applicable to the DCA. However, per SRP Section 15.0.3, Section I, Areas of Review, Item 10.F under subheading Review Interfaces, the part of this acceptance criterion that specifies radiological acceptance criteria is superseded by SRP Section 15.0.3.	15.0.3 15.6.3
SRP 15.6.3, Rev 2: Radiological Consequences of Steam Generator Tube Failure (PWR)	II (No Number)	First sentence of the last paragraph of Section II on Page 15.6.32 - Methodology and Assumptions for Calculating Radiological Consequences	Not Applicable	Not Applicable	This acceptance criterion specifies radiological analysis methodology and assumptions that are superseded by SRP Section 15.0.3.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.6.3, Rev 2: Radiological Consequences of Steam Generator Tube Failure (PWR)	II (No Number)	Last two sentences of the last paragraph of Section II on Page 15.6.32 - Plant-Specific Technical Specifications for Primary and Secondary Coolant System Iodine Activity	Partially Conforms	Applicable	The part of this guidance related to the required technical specification for primary and secondary coolant iodine activity is applicable to the DCA. However, per SRP Section 15.0.3, Section I, Areas of Review, Item 10.F under subheading Review Interfaces, the part of this acceptance criterion that specifies radiological acceptance criteria is superseded by SRP Section 15.0.3.	15.0.3 15.6.3
SRP 15.6.4, Rev 2: Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)	All	-	Not Applicable	Not Applicable	This SRP section and its acceptance criteria are applicable only to BWRs.	Not Applicable
NSDSRS 15.6.5, Rev 0: Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	II.1	Evaluation of ECCS Performance	Partially Conforms	Applicable	The features of Appendix K requirements that are technically relevant to the NuScale design are included in the Appendix K analysis of small break LOCAs.	15.6.5
NSDSRS 15.6.5, Rev 0: Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	II.2	Radiological Consequences of Most Severe LOCA	Conforms	Applicable	Per SRP Section 15.0.3, Section I, Areas of Review, Item 10 under subheading Review Interfaces, for the review of design certification applications, SRP Section 15.0.3 supersedes the radiological analyses, assumptions, acceptance criteria, and methodologies identified in this SRP Section 15.6.5. It appears that the radiological acceptance criteria reflected in this acceptance criterion are consistent with those in SRP Section 15.0.3. If any differences are identified, the criteria of SRP Section 15.0.3 take precedence.	15.6.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 15.6.5, Rev 0: Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	II.3	TMI Action Plan Requirements	Conforms	Applicable	None.	15.6.5
NSDSRS 15.6.5, Rev 0: Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	II.4	Programmatic Requirements	Conforms	Applicable	None.	15.6.5
SRP 15.6.5.A, Rev 1: Radiological Consequences of a Design Basis Loss-of-Coolant Accident Including Containment Leakage Contribution	II.1	Calculated Doses and Containment Leakage Contribution	Partially Conforms	Applicable	The part of this guidance specifying the calculation of radiological consequences of a hypothetical LOCA is applicable to the DCA. However, per SRP Section 15.0.3, Section I, Areas of Review, Item 10.H under subheading Review Interfaces, the part of this acceptance criterion that specifies radiological acceptance criteria is superseded by SRP Section 15.0.3. A portion of this guidance is applicable only to BWRs.	15.0.3 15.6.5
SRP 15.6.5.A, Rev 1: Radiological Consequences of a Design Basis Loss-of-Coolant Accident Including Containment Leakage Contribution	II.2	Model for and Calculation of Post- LOCA Containment Leakage Contribution	Partially Conforms	Applicable	The part of this guidance specifying the calculation of post LOCA containment leakage contribution is applicable to the DCA. However, per SRP Section 15.0.3, Section I, Areas of Review, Item 10.H under subheading Review Interfaces, the part of this Acceptance Criterion that specifies radiological acceptance criteria and analysis model is superseded by SRP Section 15.0.3. A portion of this guidance is applicable only to BWRs.	15.0.3 15.6.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.6.5.B, Rev 1: Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Engineered Safety Feature Components Outside Containment	II.1	ESF System Leakage Assumptions	Conforms	Applicable	None.	15.0.3 15.6.5
SRP 15.6.5.B, Rev 1: Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Engineered Safety Feature Components Outside Containment	II.2	Calculation of Radiological Consequences	Partially Conforms	Applicable	The part of this guidance specifying the calculation of radiological consequences of postulated leakage is applicable to the DCA. However, per SRP Section 15.0.3, Section I, Areas of Review, Item 10.H under subheading Review Interfaces, the part of this acceptance criterion that specifies radiological analyses, assumptions, acceptance criteria, and methodologies is superseded by SRP Section 15.0.3.	15.0.3 15.6.5
SRP 15.6.5.B, Rev 1: Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Engineered Safety Feature Components Outside Containment	II.3	Combining Radiological Consequences	Partially Conforms	Applicable	The part of this guidance specifying that radiological consequences from ESF component leakage should be combined with consequences from other fission product release paths is applicable to the DCA. However, per SRP Section 15.0.3, Section I, Areas of Review, Item 10.H under subheading Review Interfaces, the part of this acceptance criterion that specifies radiological acceptance criteria is superseded by SRP Section 15.0.3. A portion of this guidance is applicable only to BWRs.	15.0.3 15.6.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.6.5.D, Rev 1: Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Main Steam Isolation Valve Leakage Control System (BWR)	All	Various	Not Applicable	Not Applicable	This SRP section and its acceptance criteria are applicable only to BWRs.	Not Applicable
NSDSRS 15.6.6, Rev 0: Inadvertent Operation of ECCS	II.1	RCS pressure below 110 percent design value.	Conforms	Applicable	None.	15.6.6
NSDSRS 15.6.6, Rev 0: Inadvertent Operation of ECCS	II.2	Maintain minimum DNBR.	Conforms	Applicable	NuScale evaluated CHF as it is more appropriate than DNBR for the NuScale design.	15.6.6
NSDSRS 15.6.6, Rev 0: Inadvertent Operation of ECCS	II.3	An AOO should not develop more serious plant condition without other faults occurring independently.	Conforms	Applicable	None.	15.6.6
SRP 15.7.3, Rev 2: Radioactive Release from a Subsystem or Component	All	Various	Partially Conforms	Applicable	The technical content has been relocated to Branch Technical position 11-6, which is referenced in Section 11.2.	11.2
SRP 15.7.4, Rev 1: Radiological Consequences of Fuel Handling Accidents	II.1	Acceptability of Site and Dose Mitigating ESF Systems	Not Applicable	Not Applicable	This acceptance criterion specifies radiological analysis acceptance criteria that are superseded by SRP Section 15.0.3.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.7.4, Rev 1: Radiological Consequences of Fuel Handling Accidents	II.2	Radioactivity Control Features of Fuel Storage and Handling Systems	Partially Conforms	Partially Conforms	The portion of this acceptance criterion related to fuel storage and handling systems inside the Fuel Building is applicable to those systems inside the NuScale Reactor Building. The portion of this acceptance criterion related to fuel storage and handling systems inside containment is applicable only to large LWR designs that incorporate a containment building housing numerous plant SSCs. The NuScale design does not use a containment building. Rather, each NPM has its own compact steel containment vessel. This containment vessel does not contain fuel storage and handling systems. Thus, the portion of this acceptance criterion related to fuel storage and handling systems inside containment is not applicable.	15.7.4
SRP 15.7.4, Rev 1: Radiological Consequences of Fuel Handling Accidents	II.3	Dose Model and Modeling Assumptions	Not Applicable	Not Applicable	This acceptance criterion specifies radiological analysis methodology and assumptions that are superseded by SRP Section 15.0.3.	Not Applicable



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 15.7.4, Rev 1: Radiological Consequences of Fuel Handling Accidents	II.4	ESF Grade Atmosphere Clean-Up System in Spent Fuel Storage Area	Not Applicable	Not Applicable	The NuScale design does not rely on ESF ventilation systems to mitigate the consequences of a design basis accident. Non-safety-related normal ventilation systems provide atmosphere cleanup capability, as necessary, that meets the design, testing, and maintenance guidelines specified in RG 1.140. These systems provide appropriate containment, confinement, and filtering to limit releases of airborne radioactivity to the environment during normal operations, anticipated operational occurrences, and postulated accident conditions. However, these systems are not required following an accident, and accordingly receive no credit in the determination of the radiological consequences of an accident.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.7.4, Rev 1: Radiological Consequences of Fuel Handling Accidents	II.5	Radiation Detection in Containment	Partially Conforms	Applicable	The intent of this acceptance criterion is applicable but the specific language refers to LWR designs that incorporate a containment building within which fuel handling operations are performed. The NuScale design does not use a containment building. Rather, each NPM has its own compact steel containment vessel immediately surrounding the reactor vessel. The containment design provisions of this guidance, specifically directed towards fuel handling operations inside containment, have no relevance to the NuScale containment vessel design. However, the intent of this acceptance criterion is appropriate to apply to the NuScale Reactor Building, where the operating NPMs reside in the reactor pool and fuel handling operations are performed.	15.7.4
SRP 15.7.5, Rev 2: Spent Fuel Cask Drop Accidents	All	Various	Partially Conforms	Applicable	One of the principal functions of the NuScale reactor building crane (RBC) is to move the spent fuel casks in the Reactor Building Refueling Area. The RBC system design conforms to the single-failure-proof guidelines of NUREG-0612 so that any credible failure of a single component will not result in the loss of capability to stop and hold a critical load. The use of this single-failure-proof crane precludes the need to perform load drop evaluations and as a result no accident analysis has been performed to assess radiological consequences of a spent fuel cask drop accident or a NPM drop accident.	15.7.5 15.7.6

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.7.5, Rev 2: Spent Fuel Cask Drop Accidents	II.1	Acceptability of Site and Dose Mitigating ESF Systems	Not Applicable	Not Applicable	The RBC system design conforms to the single-failure-proof guidelines of NUREG-0612 so that any credible failure of a single component will not result in the loss of capability to stop and hold a critical load. The use of this single-failure-proof crane precludes the need to perform load drop evaluations and as a result no accident analysis has been performed to assess radiological consequences of a spent fuel cask drop accident or a NPM drop accident.	Not Applicable
SRP 15.7.5, Rev 2: Spent Fuel Cask Drop Accidents	II.2	Radioactivity Control Features of Fuel Storage and Handling Systems	Not Applicable	Not Applicable	The RBC system design conforms to the single-failure-proof guidelines of NUREG-0612 so that any credible failure of a single component will not result in the loss of capability to stop and hold a critical load. The use of this single-failure-proof crane precludes the need to perform load drop evaluations and as a result no accident analysis has been performed to assess radiological consequences of a spent fuel cask drop accident or a NPM drop accident.	Not Applicable
SRP 15.7.5, Rev 2: Spent Fuel Cask Drop Accidents	II.3	Dose Model and Modeling Assumptions	Not Applicable	Not Applicable	The RBC system design conforms to the single-failure-proof guidelines of NUREG-0612 so that any credible failure of a single component will not result in the loss of capability to stop and hold a critical load. The use of this single-failure-proof crane precludes the need to perform load drop evaluations and as a result no accident analysis has been performed to assess radiological consequences of a spent fuel cask drop accident or a NPM drop accident.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.7.5, Rev 2: Spent Fuel Cask Drop Accidents	II.4	ESF Grade Atmosphere Clean-Up System in Spent Fuel Storage Area	Not Applicable	Not Applicable	The RBC system design conforms to the single-failure-proof guidelines of NUREG-0612 so that any credible failure of a single component will not result in the loss of capability to stop and hold a critical load. The use of this single-failure-proof crane precludes the need to perform load drop evaluations and as a result no accident analysis has been performed to assess radiological consequences of a spent fuel cask drop accident or a NPM drop accident.	Not Applicable
SRP 15.7.5, Rev 2: Spent Fuel Cask Drop Accidents	II.5	Plant Design Features Eliminating Need for Calculation	Partially Conforms	Applicable	The RBC system design conforms to the single-failure-proof guidelines of NUREG-0612 so that any credible failure of a single component will not result in the loss of capability to stop and hold a critical load. The use of this single-failure-proof crane precludes the need to perform load drop evaluations and as a result no accident analysis has been performed to assess radiological consequences of a spent fuel cask drop accident or a NPM drop accident.	15.7.5 15.7.6
SRP 15.8, Rev 2: Anticipated Transients without Scram	II.1	Acceptance Criteria for Boiling Water Reactors (BWRs)	Not Applicable	Not Applicable	This guidance is only applicable to BWRs.	Not Applicable
SRP 15.8, Rev 2: Anticipated Transients without Scram	II.2	Acceptance Criteria for Pressurized Water Reactors (PWRs)	Not Applicable	Not Applicable	NuScale is characterized as an evolutionary plant (See the acceptance criteria in II.3).	Not Applicable
SRP 15.8, Rev 2: Anticipated Transients without Scram	II.3.A.i	Provide a diverse scram system	Partially Conforms	Partially Applicable	The NuScale design relies on diversity within the MPS to reduce the risk associated with ATWS events. Internal diversity within the MPS is a simpler approach and meets the intent of the diverse scram elements of the ATWS Rule.	15.8

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 15.8, Rev 2: Anticipated Transients without Scram	II.3.A.ii	or, Demonstrate that the ATWS event consequences are acceptable	Not Applicable	Not Applicable	As discussed in the comment above for Acceptance Criteria II.3.A.i, the NuScale design relies on diversity within the RPS to reduce the risk associated with ATWS events.	Not Applicable
SRP 15.8, Rev 2: Anticipated Transients without Scram	II.3.B	Required Equipment Does Not Apply to Design	Conforms	Applicable	As discussed above in the comment for Acceptance Criteria II.2, the design features required by 10 CFR 50.62(C)(1) either do not apply to the NuScale design or are not required to reduce the risk from ATWS events. Internal diversity within the MPS is a simpler approach to addressing the diverse scram elements of the ATWS Rule and acceptance criteria II.3.A.ii. and II.3.C(2) for evolutionary plants.	15.8
SRP 15.8, Rev 2: Anticipated Transients without Scram	II.3.C	Analysis Demonstrating the Failure Probability of Failing the ATWS Success Criteria is Sufficiently Small	Partially Conforms	Partially Applicable	NuScale will conform to the second criterion option of reducing the probability of a failure to scram. However, this is achieved with a diverse RPS instead of a diverse scram system as discussed above.	15.8
NSDSRS 15.9.A, Rev 0: Thermal Hydraulic Stability Review Responsibilities	II.1	No requirements	-	-	None.	-
NSDSRS 15.9.A, Rev 0: Thermal Hydraulic Stability Review Responsibilities	II.2	Meeting Requirements of GDC 12	Conforms	Applicable	None.	4.4.7
NSDSRS 15.9.A, Rev 0: Thermal Hydraulic Stability Review Responsibilities	II.3	Detect and suppress system criteria for demonstrating acceptable consequences of stability	Not Applicable	Not Applicable	Reactor trip signals prevent violation of CHF limits before flow instabilities can develop.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 15.9.A, Rev 0: Thermal Hydraulic Stability Review Responsibilities	II.4	Detect and Suppress Method: Exclusion zone and buffer region methodology	Not Applicable	Not Applicable	Exclusion zone option is not used in the NuScale design. Reactor trip signals that prevent violation of CHF limits before unstable flow oscillations can develop. Protective action occurs prior to development of oscillation.	4.4.7
NSDSRS 15.9.A, Rev 0: Thermal Hydraulic Stability Review Responsibilities	II.5	Detect and Suppress Method Trip of reactor before SAFDL violation	Partially Conforms	Applicable	Existing reactor trip signals provide an exclusion zone that prevents violation of SAFDL limits from other causes occur before parameters indicating flow instabilities are present.	4.4.7
NSDSRS 15.9.A, Rev 0: Thermal Hydraulic Stability Review Responsibilities	II.6	Backup options if licensing solutions declared inoperable	Not Applicable	Not Applicable	Detect and Suppress options are not employed. Existing Tech Specs for RTS provide controls on allowable unavailabilitys of protective trips. Backup options are not required.	4.4.7
NSDSRS 15.9.A, Rev 0: Thermal Hydraulic Stability Review Responsibilities	II.7	Criteria to determine the acceptability of the D&S System compliance with the requirements of GDC 20	Partially Conforms	Partially Applicable	RTS system trips reactor prior to conditions that could initiate flow instabilities. Stabilities are not detected and suppressed.	4.4.7
NSDSRS 15.9.A, Rev 0: Thermal Hydraulic Stability Review Responsibilities	II.8	Detect and Suppress system to monitor process variables and systems.	Not Applicable	Not Applicable	RTS system trips reactor prior to conditions that could initiate flow instabilities. Stabilities are not detected and suppressed.	15.9 4.4.7
NSDSRS 15.9.A, Rev 0: Thermal Hydraulic Stability Review Responsibilities	II.9	Stability-related instrumentation functionality should be demonstrated by analysis.	Conforms	Applicable	Reactor trip signals prevent violation of CHF limits before flow instabilities can develop. No unique monitoring is required to detect hydraulic instabilities.	4.4.7 15.9.A 7.2
NSDSRS 15.9.A, Rev 0: Thermal Hydraulic Stability Review Responsibilities	II.10	Ensure plant is free from other instability modes that could violate SAFDLs	Conforms	Applicable	None.	4.4.7 15.9.A
NSDSRS 15.9.A, Rev 0: Thermal Hydraulic Stability Review Responsibilities	II.11	D&S System extremely high probability of functioning in the event of an AOO.	Partially Conforms	Applicable	RTS system is used instead of a D&S. RTS occurs prior to conditions that could initiate instabilities.	4.4.7 4.4.6

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
NSDSRS 16.0, Rev. 0: Technical Specifications	All (No Number)	Acceptance Criteria for Technical Specifications	Partially Conforms	Applicable	<p>This NSDSRS section and its acceptance criteria is applicable but much of the specific language refers to existing LWR technical specifications or to plant-specific technical specifications to be developed by a COL applicant. For the latter, the DCA contains COL information items, as appropriate, that describe the required development of plant-specific technical specifications that is deferred to the COL applicant referencing the NuScale design.</p> <p>Notwithstanding the above, pursuant to 10 CFR 52.47(a)(11) and consistent with NSDSRS 16.0, the DCA contains proposed technical specifications that are prepared in accordance with 10 CFR 50.36 and 10 CFR 50.36a. The improved standard technical specification guidance for LWRs specified in this NSDSRS - NUREGs-1430 through -1434, and NUREG-2194 - were utilized to the extent appropriate and practicable. Additionally, the Technical Specifications Task Force "Writer's Guide for Plant-Specific Improved Technical Specifications," TSTF-GG-05-01, Revision 1, August 2010 was utilized in drafting the specifications.</p> <p>There are a number of technical and editorial differences between the NuScale proposed technical specifications and those presented in the improved standard technical specifications. Consistent with this NSDSRS 16.0, technical justification for such differences is provided.</p>	Ch 16

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 16.1, Rev 1: Risk-Informed Decision Making: Technical Specifications	II.1	Traditional Engineering Guidelines	Partially Conforms	Applicable	This guidance is directed toward revisions being made to existing TS, presumably including deviation from generic or any applicable standard TS. The discussion provided was considered in the development of the NuScale TS, however the specific content is not applicable to development of new generic TS as a part of a DCA.	16.1.1
SRP 16.1, Rev 1: Risk-Informed Decision Making: Technical Specifications	II.2	Probabilistic Guidelines	Partially Conforms	Applicable	This guidance is directed toward revisions being made to existing TS, presumably including deviation from generic or any applicable standard TS. The discussion provided was considered in the development of the NuScale TS, however the specific content is not applicable to development of new generic TS as a part of a DCA.	16.1.1
SRP 17.1, Rev 2: Quality Assurance During the Design and Construction Phases	All	Various	Not Applicable	Not Applicable	This guidance is applicable only to existing NRC-approved QA Programs that are based on ANSI N45.2 and its daughter standards. The NuScale QA Program Description (QAPD) is based on NQA-1-2008 and the NQA-1a-2009 addenda, as endorsed in RG 1.28, Rev 4. Since the issuance of SRP Section 17.1, the NRC has issued SRP Section 17.5 (based on NQA1) for the review of QAPDs for new reactor applicants - including applicants for design certification - under 10 CFR 52. Accordingly, SRP Section 17.5 (rather than SRP Section 17.1) is the appropriate guidance to be applied to the NuScale QAPD.	Not Applicable



**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 17.2, Rev 2: Quality Assurance During the Operations Phase	All	Various	Not Applicable	Not Applicable	This guidance is applicable only to existing NRC-approved operational QA Programs that are based on ANSI N45.2 and its daughter standards.	Not Applicable
SRP 17.3, Rev 0: Quality Assurance Program Description	All	Various	Not Applicable	Not Applicable	This guidance is applicable only to existing NRC-approved QA Programs. Since the issuance of this SRP section, the NRC has issued SRP Section 17.5 for the review of QAPDs for new reactor applicants - including applicants for design certification - under 10 CFR 52. Accordingly, SRP Section 17.5 (rather than SRP Section 17.3) is the appropriate guidance to be applied to the QAPD incorporated into the DCA.	Not Applicable
SRP 17.4, Rev 1: Reliability Assurance Program (RAP)	II.A	Design Certification	Conforms	Applicable	None.	17.4
SRP 17.4, Rev 1: Reliability Assurance Program (RAP)	II.B	COL Applicant	Not Applicable	Applicable	This acceptance criterion is applicable only to COL applicants.	Not Applicable
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.A	Organization	Partially Conforms	Applicable	The onsite,offsite, operational, and maintenance organizational elements of Item II.A.3 are the responsibility of the COL applicant referencing the certified design.	17.5
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.B	Quality Assurance Program	Partially Conforms	Applicable	The provisions for site-specific and operational phase of the quality assurance program are not applicable to the NuScale QA program to be applied during the design certification phase, and are to be addressed within the operational QA program developed and maintained by the COL applicant referencing the certified design.	17.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.C	Design Control and Verification	Conforms	Applicable	None.	17.5
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.D	Procurement Document Control	Conforms	Applicable	None.	17.5
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.E	Instructions, Procedures, and Drawings (Controlled Documents)	Conforms	Applicable	None.	17.5
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.F	Document Control	Partially Conforms	Applicable	The site-specific and operational provisions of document control are the responsibility of the COL applicant referencing the certified design.	17.5
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.G	Control of Purchased Material, Equipment, and Services	Conforms	Applicable	None.	17.5
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.H	Identification and Control of Materials, Parts, and Components	Not Applicable	Applicable	This acceptance criterion governs activities that are the responsibility of the COL applicant referencing the certified design.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.I	Control of Special Processes	Not Applicable	Applicable	This acceptance criterion governs activities that are the responsibility of the COL applicant referencing the certified design.	Not Applicable
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.J	Inspection	Partially Conforms	Applicable	The provisions specific to inservice, modification, etc. are the responsibility of the COL applicant referencing the certified design.	17.5
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.K	Test Control	Conforms	Applicable	None.	17.5
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.L	Control of Measuring and Test Equipment	Conforms	Applicable	None.	17.5
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.M	Handling, Storage, and Shipping	Not Applicable	Applicable	This acceptance criterion governs activities that are the responsibility of the COL applicant referencing the certified design.	Not Applicable
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.N	Inspection, Test, and Operating Status	Not Applicable	Applicable	This acceptance criterion governs activities that are the responsibility of the COL applicant referencing the certified design.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.O	Nonconforming Materials, Parts, or Components	Conforms	Applicable	None.	17.5
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.P	Corrective Action	Conforms	Applicable	None.	17.5
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.Q	Quality Assurance Records	Conforms	Applicable	None.	17.5
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.R	Audits	Conforms	Applicable	None.	17.5
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.S	Training and Qualification Criteria	Conforms	Applicable	None.	17.5
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.T	Training and Qualification - Inspection and Test	Conforms	Applicable	None.	17.5

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	COL Applicability	Comments	Section
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.U	Nonsafety-Related SSC Quality Controls	Conforms	Applicable	None.	17.5
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.V	Quality Assurance Program Commitments	Conforms	Applicable	None.	17.5
SRP 17.6, Rev 2: Maintenance Rule	All	Various	Not Applicable	Applicable	This SRP section and its acceptance criteria govern a site-specific operational program that is the responsibility of the COL applicant. Consistent with SRP Section 1.0, Appendix A; RG 1.206, Regulatory Position C.III.4; and ESP/DC/COL-ISG-015, the DCA contains COL information items that describe the information that is deferred to the COL applicant referencing the certified design.	Not Applicable
SRP 18.0, Rev 2: Human Factors Engineering	II.A	Review of the HFE Aspects of a New Plant	Conforms	Applicable	None.	18.1 thru 18.12
SRP 18.0, Rev 2: Human Factors Engineering	II.B	Review of the HFE Aspects of Control Room Modifications	Not Applicable	Not Applicable	This acceptance criterion is applicable to existing reactor licensees that request NRC approval of Control Room modifications.	Not Applicable
SRP 18.0, Rev 2: Human Factors Engineering	II.C	Review of the HFE Aspects of Modifications Affecting Risk Important Human Actions	Not Applicable	Not Applicable	This acceptance criterion is applicable to existing reactor licensees that request NRC approval of plant changes that affect important human actions.	Not Applicable
SRP Appendix 18-A, Rev 0: Crediting Manual Operator Actions in Diversity and Defense-in-Depth (D3) Analyses	C.1.B	Review Criteria for Phase 1 (Analysis)	Conforms	Applicable	This appendix supersedes DI&CISG-05, Section 3, "Crediting Manual Operator Actions in Diversity and Defense-in-Depth (D3) Analyses".	7.1.5 18.4 18.6

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP Appendix 18-A, Rev 0: Crediting Manual Operator Actions in Diversity and Defense-in-Depth (D3) Analyses	C.2.B	Review Criteria for Phase 2 (Preliminary Validation)	Conforms	Applicable	This appendix supersedes DI&CISG-05, Section 3, "Crediting Manual Operator Actions in Diversity and Defense-in-Depth (D3) Analyses".	7.1.5 18.4 18.6
SRP Appendix 18-A, Rev 0: Crediting Manual Operator Actions in Diversity and Defense-in-Depth (D3) Analyses	C.3.B	Review Criteria for Phase 3 (Integrated System Validation)	Conforms	Applicable	This appendix supersedes DI&CISG-05, Section 3, "Crediting Manual Operator Actions in Diversity and Defense-in-Depth (D3) Analyses".	7.1.5 18.4 18.6
SRP Appendix 18-A, Rev 0: Crediting Manual Operator Actions in Diversity and Defense-in-Depth (D3) Analyses	C.4.B	Review Criteria for Phase 3 (Maintaining Long-Term Integrity of Credited Manual Actions in the D3 Analysis)	Conforms	Applicable	This appendix supersedes DI&CISG-05, Section 3, "Crediting Manual Operator Actions in Diversity and Defense-in-Depth (D3) Analyses".	7.1.5 18.4 18.6
SRP 19.0, Rev 3: Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors	All	Various	Partially Conforms	Applicable	Evaluation of site-specific hazards and PRA update are COL applicant responsibility.	19.0 19.1 19.2
SRP 19.1, Rev 3: Determining The Technical Adequacy of Probabilistic Risk Assessment For Risk-Informed License Amendment Requests After Initial Fuel Load	All	Various	Not Applicable	Applicable	Applicable to PRAs used by a licensee to support license amendments for an operating reactor.	Not Applicable
SRP 19.2, Initial Issuance: Review of Risk Information Used to Support Permanent Plant Specific Changes to the Licensing Basis: General Guidance	All	Various	Not Applicable	Applicable	Applicable to licensees plant-specific proposals for changes to the licensing basis.	Not Applicable

**Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)**

<b>SRP or DSRS Section, Rev: Title</b>	<b>AC</b>	<b>AC Title/Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
SRP 19.3, Rev 0: Regulatory Treatment of Non-Safety Systems for Passive Advanced Light Water Reactors	All	Various	Conforms	Applicable	None.	19.3
SRP 19.4, Rev 0: Strategies and Guidance to Address Loss-of-Large Areas of the Plant Due to Explosions and Fires	All	Various	Partially Conforms	Applicable	Applicable with the exception of acceptance criterion II.17 Boiling Water Reactor: Containment Venting and Vessel Flooding (Item B.2.e) which is a BWR specific criterion and acceptance criterion II.20 SFP Mitigative Measures. The SFP mitigating measure is not required NEI 06-12 includes statement this mitigation strategy is not required if the SFP is below grade and cannot be drained. The NuScale SFP is below grade and cannot be drained.	19.4 20
SRP 19.5, Rev 0: Adequacy of Design features and functional capabilities identified and described for withstanding Aircraft Impacts	All	Various	Conforms	Applicable	None.	19.5

**Table 1.9-4: Conformance with Interim Staff Guidance (ISG)**

ISG Section/ Title	AC	AC Title / Description	Conformance Status	COL Applicability	Comments	Section
DC/COL-ISG-1: Seismic Issues of High Frequency Ground Motion	1	Seismic Issues addressed in this Interim Staff Guidance	-	-	This section roadmaps out to the guidance provided in Sections 2, 3, 4, and 5. There are no specific requirements.	3.7
DC/COL-ISG-1	2	Ground Motion Definitions	Conforms	Applicable	The definitions provided in Section 3.7 are consistent.	3.7
DC/COL-ISG-1	3	Staff Guidance/Position on the Definitions of Safe-Shutdown and Operating-Basis Earthquakes, Use of Various Ground Motions, Seismic Instrumentation and Operating-Basis Earthquake Exceedance	Conforms	Applicable	The CSDRS (and CSDRS-HF) is effectively the SSE for the DCA. The OBE is specified as 1/3 of the CSDRS thus does not require any analysis in the DCA. There are COL items for the applicant to ensure the GMRS is enveloped and to have a seismic monitoring program with responses following an OBE exceedance.	3.7
DC/COL-ISG-1	4	Staff Guidance/Position on Addressing HF Ground Motion Evaluations	Conforms	Applicable	The NuScale Certified Design includes a High Frequency CSDRS.	3.7
DC/COL-ISG-1	5	Staff Comments on the Industry Draft White Paper on Testing of Dynamic Soil Properties for Nuclear Power Plant Combined License Applications and Guidance on Information for Review	Partially Conforms	Applicable	This discusses laboratory analysis of the site-specific soil column. The FSAR includes COL items for the Applicant to develop site-specific information.	2.5
DC/COL-ISG-2: Financial Qualifications of Applicants For Combined License Applications	All	Various	Not Applicable	Not Applicable	This ISG is applicable to COL applicants.	Not Applicable
DC/COL-ISG-3: Probabilistic Risk Assessment Information to Support Design Certification and Combined License Applications	All	Various	Not Applicable	Not Applicable	Guidance concerning the review of PRA information and severe accident assessments submitted to support DC and COL applications has been incorporated into SRP 19.0, Rev 3.	Not Applicable



**Table 1.9-4: Conformance with Interim Staff Guidance (ISG) (Continued)**

<b>ISG Section/ Title</b>	<b>AC</b>	<b>AC Title / Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
DC/COL-ISG-4: Definition of Construction and on Limited Work Authorizations	All	Various	Not Applicable	Not Applicable	This ISG is applicable to all ESP and COL applicants requesting authorization to perform limited work activities or considering preconstruction activities.	Not Applicable
DC/COL-ISG-5: GALE86 Code for Calculation of Routine Radioactive Releases in Gaseous and Liquid Effluents to Support Design Certification and Combined License Applications	All	Five paragraphs under heading Final Interim Staff Guidance on Page 3 - Acceptability of GALE86	Not Applicable	Not Applicable	The NuScale design is similar to large PWRs in the existing fleet with regards to effluent release calculations. However, the development of an alternate methodology is necessary because the existing PWRGALE code was developed in the 1980s for evaluation of the large PWR reactors of that time and does not appropriately address the NuScale plant design.	Not Applicable
DC/COL-ISG-6: Evaluation and Acceptance Criteria for 10 CFR 20.1406 to Support Design Certification and Combined License Applications	Bullets 1 thru 6 (p 3 & 4)	Acceptance Criteria - Compliance with RG 4.21	Partially Conforms	Applicable	This guidance refers to Attachment C. The correct reference is Attachment B. This guidance is applicable, except for the portions that relate to site-specific, operational aspects that are the responsibility of the COL applicant referencing the NuScale design. Consistent with SRP Section 1.0, Appendix A; RG 1.206, Position C.III.4; and ESP/DC/ COL-ISG-015, the DCA contains COL information items that describe the site-specific, operational information deferred to the COL applicant referencing the certified design. The aspects of this guidance that pertain to design features, facilities, functions, and equipment that are technically relevant to the NuScale standard plant design are applicable to the DCA.	12.3.6
DC/COL-ISG-7: Assessment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures	1	Normal and Extreme Winter Precipitation Events	Conforms	Applicable	Section 3.4 identifies parameter specified for the Extreme and Normal winter precipitation events. These values are used in the structural analysis in 3.8. The COL applicant needs to determine site-specific information to compare to the design parameters. That determination is performed in Section 2.4.	2.4 3.4 3.8
DC/COL-ISG-7	2	Resulting Normal and Extreme Winter Precipitation Live Roof Loads	Conforms	Applicable	The design parameters are used in the analysis in section 3.8.	3.8

**Table 1.9-4: Conformance with Interim Staff Guidance (ISG) (Continued)**

ISG Section/ Title	AC	AC Title / Description	Conformance Status	COL Applicability	Comments	Section
DC/COL-ISG-8: Necessary Content of Plant-Specific Technical Specifications	para 1 (p4)	First paragraph under heading Final Interim Staff Guidance, specifying identification and timing of resolution of generic technical specification COL action items	Conforms	Applicable	None.	Ch 16
DC/COL-ISG-8	para 2-4 (p4 & 5)	Second, third, and fourth paragraphs under heading Final Interim Staff Guidance, specifying compliance options for COL applicants	Not Applicable	Applicable	This portion of the ISG is applicable only to COL applicants.	Not Applicable
DC/COL-ISG-10: Review of Evaluation to Address Adverse Flow Effects in Equipment Other Than Reactor Internals	All	Final paragraph on Page 1 - Review of Adverse Flow Effects	Partially Conforms	Applicable	This guidance is applicable except for aspects that are BWR specific.	
DC/COL-ISG-11: Finalizing Licensing-basis Information	All	Licensing-Basis Information Freeze Point; Changes That Should Not be Considered for Deferral	Partially Conforms	Applicable	This guidance is applicable except for aspects that are applicable only to COL applicants or early site permit.	
DC/COL-ISG-13: NUREG-0800 Standard Review Plan Section 11.2 and Branch Technical Position 11-6 Assessing the Consequences of an Accidental Release of Radioactive Materials from Liquid Waste Tanks for Combined License Applications Submitted under 10 CFR Part 52	1	Failure Mechanism and Radioactivity Releases	Not Applicable	Applicable	This guidance is applicable except for site-specific aspects that are the responsibility of the COL applicant.	11.2.3
DC/COL-ISG-13	2	Mitigating Design Features	Not Applicable	Applicable	This guidance is applicable except for site-specific aspects that are the responsibility of the COL applicant.	11.2.3

**Table 1.9-4: Conformance with Interim Staff Guidance (ISG) (Continued)**

ISG Section/ Title	AC	AC Title / Description	Conformance Status	COL Applicability	Comments	Section
DC/COL-ISG-13	3	Radioactive Source Term (Including Attachment A)	Not Applicable	Applicable	This guidance is applicable except for site-specific aspects that are the responsibility of the COL applicant.	11.2.3
DC/COL-ISG-13	4	Calculations of Transport Capabilities in Ground Water or Surface Water	Not Applicable	Applicable	This acceptance criterion governs site-specific calculations that are the responsibility of the COL applicant referencing the certified design.	Not Applicable
DC/COL-ISG-13	5	Exposure Scenarios and Acceptance Criteria	Not Applicable	Applicable	This acceptance criterion governs analysis modeling using site-specific hydrogeological data, site characteristics, and radiological analysis; as such, this guidance is the responsibility of the COL applicant referencing the certified design.	Not Applicable
DC/COL-ISG-13	6	SRP Dose Acceptance Criteria	Not Applicable	Applicable	This guidance is applicable except for site-specific aspects that are the responsibility of the COL applicant.	11.2.3
DC/COL-ISG-13	7	Specifications on Tank Waste Radioactivity Concentration Levels	Not Applicable	Applicable	This acceptance criterion is applicable except for site-specific aspects (e.g., development and implementation of the ODCM) that are the responsibility of the COL applicant.	11.2.2
DC/COL-ISG-13	8	Evaluation Findings for Combined License Reviews	Not Applicable	Applicable	This acceptance criterion is explicitly directed towards the review of combined license applications.	Not Applicable
DC/COL-ISG-14: Assessing Ground Water Flow and Transport of Accidental Radionuclide Releases	All	Area of Review; Review Interfaces; Regulatory Requirements; Onsite Hydrogeological Characterization; Contamination Source and Receptor Location; Groundwater Modeling and Pathway Prediction; and Radioactive Consequence Analysis	Not Applicable	Applicable	As a supplement to SRP Sections 2.4.12 and 2.4.13, this guidance governs site-specific hydrogeological data, site characteristics, and radiological analysis aspects that are the responsibility of the COL applicant referencing the certified design.	Not Applicable
ESP/DC/COL-ISG-15: Post-Combined License Commitments	No Num (p4-11)	New Section C.III.4.3 to Replace Section C.III.4.3 of RG 1.206	Not Applicable	Applicable	This guidance is directed explicitly towards COL applicants.	Not Applicable

Table 1.9-4: Conformance with Interim Staff Guidance (ISG) (Continued)

ISG Section/ Title	AC	AC Title / Description	Conformance Status	COL Applicability	Comments	Section
ESP/DC/COL-ISG-15	No Num (p11-23)	Anticipated NRC Revisions of NUREG0800, SRP Chapter 1.0	Partially Conforms	Applicable	The portions of this guidance that apply to the DCA include discussion concerning COL action items and COL information items and not using the term "COL holder item." COL action items are identified, as necessary, throughout the FSAR.	Ch 1
DC/COL-ISG-16: Compliance with 10 CFR 50.54(hh)(2) and 10 CFR 52.80(d)	All	-	Partially Conforms	Applicable	Portions of LOLA and the ISG have been down-graded to non-SRI. The ISG endorses NEI 06-12 Rev 3, B.5.b Phase 2 & 3 Submittal Guideline. LOLA is addressed in Chapter 20.	20
DC/COL-ISG-17: Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Interaction Analyses	All	-	Not Applicable	Applicable	As stated in Section 1.0, (t)his ISG is applicable to the review of seismic design information submitted to support combined license (COL) applications. A review of this ISG confirms that its content governs site-specific information that is applicable only to COL applicants.	Not Applicable
DC/COL-ISG-19: Gas Accumulation Issues in Safety Related Systems	All	Various	Not Applicable	Not Applicable	This guidance is applicable only to reactor plant designs for which operation of emergency core cooling, residual heat removal, and containment spray systems relies on pumps (i.e. forced circulation). For such plant designs, the gas accumulation concerns raised in GL 2008-01 could adversely impact the operability of the subject systems. The NuScale emergency core cooling and decay heat removal systems (the NuScale design does not include a containment spray system) operate via natural circulation, and do not require or include pumps.	Not Applicable
DC/COL-ISG-20: Implementation of a Probabilistic Risk Assessment-Based Seismic Margin Analysis for New Reactors	All	Various	Not Applicable	Not Applicable	Guidance concerning the performance of a SMA submitted to support DC and COL applications has been incorporated into SRP 19.0, Rev 3.	Not Applicable

Table 1.9-4: Conformance with Interim Staff Guidance (ISG) (Continued)

ISG Section/ Title	AC	AC Title / Description	Conformance Status	COL Applicability	Comments	Section
DC/COL-ISG-21: Review of Nuclear Power Plant Designs using a Gas Turbine Driven Standby Emergency Alternating Current Power System	All	Guidance for Emergency Gas Turbine Generators (Including Attachment 1)	Not Applicable	Not Applicable	This guidance is applicable only to nuclear power plants that use a gas turbine-driven standby emergency AC power system - in lieu of emergency diesel generators - to supply power to safety-related or risk-significant equipment for operational events and during postulated accident conditions. The NuScale design uses onsite backup diesel generators as opposed to gas turbine generators. However, regardless of the type of standby AC generation used in the NuScale design, the onsite standby AC generation source and the onsite AC distribution system it serves are not safety-related, nor are they relied upon to fulfill safety functions during the first 72 hours following a design basis accident.	Not Applicable
DC/COL-ISG-22: Impact of Construction (Under a Combined License) of New Nuclear Power Plant Units on Operating Units at Multi-Unit Sites	All	Various	Not Applicable	Applicable	This ISG is applicable to COL applicants.	Not Applicable
DC/COL-ISG-24: Implementation of RG 1.221 on Design-Basis Hurricane and Hurricane Missiles	All	Various	Conforms	Applicable	Section 2.0 establishes requirements for hurricane wind speed and missile spectra "consistent with guidance in Regulatory 1.221, R0." Specific design requirements are established in Sections 3.3.2 and 3.5.1.4.	2.0 3.3 3.5
DC/COL-ISG-25: Changes during Construction Under Title 10 of the Code of Federal Regulations Part 52	All	Various	Not Applicable	Applicable	This ISG is applicable to 10 CFR Part 52, COL licensees with Changes during Construction license condition.	Not Applicable
DC/COL-ISG-26: Environmental Issues Associated with New Reactors	All	Various	Not Applicable	Applicable	This ISG is applicable to the review of ESP and COL applications, including those applicants requesting a limited work authorization.	Not Applicable

Table 1.9-4: Conformance with Interim Staff Guidance (ISG) (Continued)

ISG Section/ Title	AC	AC Title / Description	Conformance Status	COL Applicability	Comments	Section
DC/COL-ISG-27: Specific Environmental Guidance for Light Water Small Modular Reactor	All	Various	Not Applicable	Applicable	This ISG is applicable to the review of ESP, LWA, OL, CP, and COL applications for light water SMR reactor technologies.	Not Applicable
DC/COL-ISG-28: Assessing the Technical Adequacy of the Advanced Light-Water Reactor Probabilistic Risk Assessment for the Design Certification Application and Combined License Application	All	Various	Conforms	Applicable	Provides guidance for DC and COL applicants to conform to PRA Standard.	19.1
Digital I&C-ISG-01: Cyber Security	5.	Staff Position	Partially Conforms	Applicable	To the extent that NuScale addresses certain cyber security provisions of 10 CFR 73.54 through the use of specific design features in its standard plant design, the guidance of RG 1.152, Rev. 3, and RG 5.71 are considered applicable to the DCA. The portions of RG 5.71 that govern site-specific operational and programmatic activities (e.g., development and implementation of operational cyber security plans) are applicable to the COL applicant.	Tech Report refd in 7.0.3 7.1.2 7.2.1 7.2.3 7.2.5 7.2.9 7.2.15
Digital I&C-ISG-02: Diversity and Defense-in-Depth (D3)	1 and 2	Adequate Diversity and Manual Operator Actions - Staff Position (Pages 2 and 3)	Partially Conforms	Applicable	These positions are applicable except for the application of certain subtier guidance (see BTP 7-19).	7.1.5
Digital I&C-ISG-02	3	BTP 7-19 Position 4 Challenges - Staff Position (Page 6)	Partially Conforms	Applicable	This position is applicable except for reference to BTP 7-19, Revision 5. NuScale intends to apply BTP 7-19, Revision 6. This position provides proposed wording for BTP 7-19, Position 4, which has been incorporated with minor changes into Revision 6 of BTP 7-19.	7.1.5
Digital I&C-ISG-02	4	Effects of Common Cause Failure (CCF) - Staff Position (Pages 8 and 9)	Conforms	Applicable	None.	7.0.4 19.1
Digital I&C-ISG-02	6	Echelons of Defense - Staff Position (Page 12)	Partially Conforms	Applicable	This position is applicable except for reference to BTP 7-19, Revision 5. NuScale intends to apply BTP 7-19, Revision 6.	7.1.5

Table 1.9-4: Conformance with Interim Staff Guidance (ISG) (Continued)

ISG Section/ Title	AC	AC Title / Description	Conformance Status	COL Applicability	Comments	Section
Digital I&C-ISG-02	7	Single Failure - Staff Position (Page 14)	Conforms	Applicable	None.	7.1 7.2
Digital I&C-ISG-03: Risk-Informed Digital Instrumentation and Controls	4	Staff Position	Conforms	Applicable	None.	7
Digital I&C-ISG-04: Highly Integrated Control Rooms & Digital Communication Systems	1	Interdivisional Communications - Staff Position (Pages 4 through 8)	Conforms	Applicable	None.	7.0.4 7.1.2 7.2.13
Digital I&C-ISG-04	2	Command Prioritization - Staff Position (Pages 8 through 10)	Conforms	Applicable	None.	7.0.4 7.1.2 7.1.6 7.2.10
Digital I&C-ISG-04	3	Multidivisional Control and Display Stations - Staff Position (Pages 11 through 16)	Conforms	Applicable	None.	7.0.4 7.1.6
Digital I&C-ISG-04	3.1	Independence and Isolation (Pages 11 through 13)	Conforms	Applicable	None.	7.0.4 7.1.2
Digital I&C-ISG-04	3.2	Human Factors Considerations (Pages 13 through 15)	Conforms	Applicable	None.	7.0.4 7.2.1
Digital I&C-ISG-04	3.3	Diversity and Defense-in-Depth (D3) Considerations (Page 15)	Conforms	Applicable	None.	7.1.5
Digital I&C-ISG-05: Highly Integrated Control Rooms - Human Factors	1	Computer-Based Procedures - Staff Position (Pages 3 through 7)	Partially Conforms	Applicable	This position is applicable except for site-specific operational elements of subtier NUREG0899 that are the responsibility of the COL applicant.	18.7 18.8
Digital I&C-ISG-05	2	Minimum Inventory - Staff Position (Pages 9 through 11)	Partially Conforms	Applicable	This acceptance criterion is applicable except for the application of certain subtier guidance.	18.7
Digital I&C-ISG-05	3	Crediting Manual Operator Actions in Diversity and Defense-In-Depth (D3) Analyses (Pages 13 through 21)	Not Applicable	Applicable	This acceptance criterion is superseded by Chapter 18 Appendix 18-A.	7.1.5 18.4 18.6

**Table 1.9-4: Conformance with Interim Staff Guidance (ISG) (Continued)**

<b>ISG Section/ Title</b>	<b>AC</b>	<b>AC Title / Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
Digital I&C-ISG-05	3.1.B	Phase 1: Analysis - Review Criteria (Pages 15 through 16)	Not Applicable	Applicable	This acceptance criterion is superseded by Chapter 18 Appendix 18-A Criterion 1.B.	7.1.5 18.4
Digital I&C-ISG-05	3.2.B	Phase 2: Preliminary Validation - Review Criteria (Page 18)	Not Applicable	Applicable	This acceptance criterion is superseded by Chapter 18 Appendix 18-A Criterion 2.B.	7.1.5 18.4
Digital I&C-ISG-05	3.3.B	Phase 3: Integrated System Validation - Review Criteria (Pages 19 through 20)	Not Applicable	Applicable	This acceptance criterion is superseded by Chapter 18 Appendix 18-A.Criterion 3.B.	7.1.5 18.4 18.10
Digital I&C-ISG-05	3.4.B	Phase 4: Maintaining Long-Term Integrity of Credited Manual Actions in the D3 Analysis - Review Criteria (Page 21)	Not Applicable	Applicable	This acceptance criterion is superseded by Chapter 18 Appendix 18-A.Criterion 4.B.	7.1.5 18.4 18.6
Digital I&C-ISG-06: Licensing Process	All	Various	Not Applicable	Applicable	This guidance is directed towards review of requests for licensing basis changes from existing licensees to implement digital I&C upgrades.	Not Applicable
Digital I&C-ISG-07: Fuel Cycle Facilities	All	Various	Not Applicable	Not Applicable	This guidance is directed towards review of proposed measures for protecting digital I&C equipment used as items relied on for safety (IROFS) at fuel cycle facilities from unintentional digital events.	Not Applicable
NSIR/DPR-ISG-01: Emergency Planning for Nuclear Power Plants	All	Various	Not Applicable	Applicable	This guidance governs site-specific programmatic and design aspects of emergency planning that are the responsibility of the COL applicant referencing the NuScale design. Consistent with SRP Section 1.0, Appendix A; RG 1.206, Position C.III.4; and ESP/DC/ COL-ISG-015, the DCA contains COL information items that describe the site-specific, programmatic information that is deferred to the COL applicant referencing the certified design.	Not Applicable
NSIR/DPR-ISG-02: Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants	All	Various	Not Applicable	Applicable	Applicable to license holder during decommissioning activities.	Not Applicable



**Table 1.9-4: Conformance with Interim Staff Guidance (ISG) (Continued)**

<b>ISG Section/ Title</b>	<b>AC</b>	<b>AC Title / Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
NSIR/DPR-ISG-03: Review of Security Exemptions/ License Amendment Requests for Decommissioning Nuclear Power Plants	All	Various	Not Applicable	Applicable	Applicable to license holder during decommissioning activities.	Not Applicable
JLD-ISG-12-01, Rev 1: Compliance with Order EA-12-049 Concerning Mitigation Strategies	All	Various	Not Applicable	Applicable	This ISG is applicable to holders of, and applicants for, operating licenses, construction permits, and combined licenses.	Not Applicable
JLD-ISG-12-03, Rev 1: Compliance with Order EA-12-051 Concerning Spent Fuel Pool Instrumentation	All	Various	Not Applicable	Applicable	This ISG is applicable to holders of, and applicants for operating licenses, construction permits, and combined licenses. Pool monitoring instrumentation that is capable of monitoring and providing indication of beyond design basis events (i.e., instrumentation that can monitor a wide range of spent fuel pool levels) is part of the NuScale design.	Not Applicable
JLD-ISG-12-04, Draft: Performing a Seismic Margin Assessment in Response to the March 2012 Request for Information Letter	All	Various	Not Applicable	Not Applicable	This ISG is only intended for the purpose of response to the March 2012 50.54(f) request for information letter. DC/COL-ISG-020 remains the NRCs current guidance for application of an SMA to new reactors licensing.	Not Applicable
JLD-ISG-12-05, Draft: Performance of an Integrated Assessment for Flooding	All	Various	Not Applicable	Not Applicable	This ISG is only intended for the purpose of response to the March 2012 50.54(f) request for information letter.	Not Applicable
JLD-ISG-12-06, Draft: Performing a Tsunami, Surge, or Seiche Hazard Assessment	All	Various	Not Applicable	Not Applicable	This ISG is only intended for the purpose of response to the March 2012 50.54(f) request for information letter.	Not Applicable
JLD-ISG-13-01, Draft: Estimating Flooding Hazards due to Dam Failure	All	Various	Not Applicable	Not Applicable	The information provided by this guidance is site-specific and is the responsibility of the COL applicant.	Not Applicable

**Table 1.9-4: Conformance with Interim Staff Guidance (ISG) (Continued)**

<b>ISG Section/ Title</b>	<b>AC</b>	<b>AC Title / Description</b>	<b>Conformance Status</b>	<b>COL Applicability</b>	<b>Comments</b>	<b>Section</b>
JLD-ISG-2015-01, Revision 0: Compliance with Phase 2 of Order EA- 13-109, Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions	All	Various	Not Applicable	Not Applicable	This ISG is applicable to BWR licensees with Mark I and Mark II containments.	Not Applicable
DSS-ISG-2010-01	1	Fuel Assembly Selection	Conforms	Applicable	One fuel assembly design is used in the criticality analysis.	9.1.1
DSS-ISG-2010-01	2	Depletion Analysis	Conforms	Applicable	The analysis does not take credit for burnup.	9.1.1
DSS-ISG-2010-01	3a	Axial Burnup Profile	Conforms	Applicable	The analysis does not take credit for burnup.	9.1.1
DSS-ISG-2010-01	3b	Rack Model	Conforms	Applicable	The rack model analysis is appropriate for conditions.	9.1.1
DSS-ISG-2010-01	3c	Interfaces	Conforms	Applicable	The analysis does not take credit for zoning or a loading pattern.	9.1.1
DSS-ISG-2010-01	3d	Normal Conditions	Conforms	Applicable	The analysis considers the presence of an additional assembly alongside the fuel storage racks. Due to the spacing and the large number of assemblies in the base analysis model, there is no statistically significant increase in reactivity.	9.1.1
DSS-ISG-2010-01	3e	Accident Conditions	Conforms	Applicable	The analysis considers fuel handling accidents, rack damage consistent with postulated accidents and full boron dilution. All analyses are within the limits established for normal conditions.	9.1.1
DSS-ISG-2010-01	4a	Area of Applicability	Conforms	Applicable	The analysis considers area of applicability in the code validation.	9.1.1
DSS-ISG-2010-01	4b	Trend Analysis	Conforms	Applicable	The analysis includes a trend analysis in the code validation.	9.1.1
DSS-ISG-2010-01	4c	Statistical Treatment	Conforms	Applicable	The analysis includes both a bias term and an uncertainty derived from the code validation.	9.1.1
DSS-ISG-2010-01	4d	Lumped Fission Products	Conforms	Applicable	The analysis does not take credit for burnup.	9.1.1
DSS-ISG-2010-01	4e	Code-to-Code Comparisons	Conforms	Applicable	The analysis is validated against critical experiments. Therefore, this requirement does not apply.	9.1.1

Table 1.9-4: Conformance with Interim Staff Guidance (ISG) (Continued)

ISG Section/ Title	AC	AC Title / Description	Conformance Status	COL Applicability	Comments	Section
DSS-ISG-2010-01	5a	Precedents	Conforms	Applicable	The analysis does not rely upon any cited precedents.	9.1.1
DSS-ISG-2010-01	5b	References	Conforms	Applicable	Cited references are publicly available and are commonly referenced in SFP criticality analyses.	9.1.1
DSS-ISG-2010-01	5c	Assumptions	Conforms	Applicable	Assumptions utilized in the analysis are either observably conservative or are justified in the presentation of the assumption.	9.1.1

**Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933)**

Item	Regulation Description / Title	Conformance Status	COL Applicability	Comments	Section
50.34(f)(1)(i)	Perform a plant/site-specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant (II.B.8)	Partially Conforms	Applicable	Design certification will address reliability of core and containment heat removal systems, with an update required by COL applicant to reflect site-specific conditions.	19.0 19.1 19.2
50.34(f)(1)(ii)	Perform an evaluation of the proposed auxiliary feedwater system (II.E.1.1)	Not Applicable	Not Applicable	This rule requires an evaluation of proposed PWR auxiliary feedwater (AFW) systems. The NuScale plant design does not involve an AFW system as would be found at a typical large LWR. Neither the literal language nor the intent of this rule applies to the NuScale design.	Not Applicable
50.34(f)(1)(iii)	Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA (II.K.2.16 and II.K.3.25)	Not Applicable	Not Applicable	The NuScale reactor design differs from that of large PWRs in that the NuScale design does not require or include reactor coolant pumps. Rather, the NuScale design uses passive natural circulation of the primary coolant, eliminating the need for reactor coolant pumps.	Not Applicable
50.34(f)(1)(iv)	Perform an analysis of the probability of a small-break LOCA caused by a stuck-open power-operated relief valve (PORV) (II.K.3.2)	Not Applicable	Not Applicable	This guidance is applicable only to PWRs that are designed with power-operated pressurizer relief valves. The NuScale design does not use power-operated relief valves.	Not Applicable
50.34(f)(1)(v)	Perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection and reactor core isolation cooling system initiation levels (II.K.3.13)	Not Applicable	Not Applicable	This requirement explicitly states its applicability only to BWRs. As a PWR, the NuScale design is fundamentally different than that of a BWR, such that separation of system initiation levels as contemplated by this requirement is not pertinent.	Not Applicable
50.34(f)(1)(vi)	Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves (II.K.3.16)	Not Applicable	Not Applicable	This requirement explicitly states its applicability only to BWRs. Regardless, the issue contemplated by this requirement was related to power-operated relief valves. The NuScale design does not use power-operated relief valves.	Not Applicable

**Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933) (Continued)**

Item	Regulation Description / Title	Conformance Status	COL Applicability	Comments	Section
50.34(f)(1)(vii)	Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system design modifications that would eliminate the need for manual activation (II.K.3.18)	Not Applicable	Not Applicable	This requirement explicitly states its applicability only to BWRs. As a PWR, the NuScale design is fundamentally different than that of a BWR, such that design modifications contemplated by this requirement are not pertinent.	Not Applicable
50.34(f)(1)(viii)	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 (II.K.3.21)	Not Applicable	Not Applicable	This requirement explicitly states its applicability only to BWRs. As a PWR, the NuScale design is fundamentally different than that of a BWR, such that the system auto-restart and initiation signal verification contemplated by this requirement is not pertinent.	Not Applicable
50.34(f)(1)(ix)	Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the high pressure coolant injection and reactor core isolation cooling systems (II.K.3.24)	Not Applicable	Not Applicable	This requirement explicitly states its applicability only to BWRs. As a PWR, the NuScale design is fundamentally different than that of a BWR, such that the study contemplated by this requirement is not pertinent.	Not Applicable
50.34(f)(1)(x)	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 (II.K.3.28)	Not Applicable	Not Applicable	This requirement explicitly states its applicability only to BWRs. As a PWR, the NuScale design is fundamentally different than that of a BWR, such that the study contemplated by this requirement is not pertinent.	Not Applicable
50.34(f)(1)(xi)	Provide an evaluation of depressurization methods (II.K.3.45)	Not Applicable	Not Applicable	This requirement explicitly states its applicability only to BWRs. As a PWR, the NuScale design is fundamentally different than that of a BWR, such that the evaluation contemplated by this requirement is not pertinent.	Not Applicable
50.34(f)(1)(xii)	Perform an evaluation of alternative hydrogen control systems	Not Applicable	Not Applicable	Pursuant to 10 CFR 52.47(a)(8) and 10 CFR 50.34(f), Paragraph (f)(1)(xii) is excluded from the information required to be included in an application for a design certification.	Not Applicable

**Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933) (Continued)**

Item	Regulation Description / Title	Conformance Status	COL Applicability	Comments	Section
50.34(f)(2)(i)	Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCAs(I.A.4.2)	Not Applicable	Applicable	Simulator capability is not included within the scope of the NuScale standard plant or design certification application. Thus, this requirement is not technically relevant to the NuScale standard plant design, and consistent with 10 CFR 52.47(a)(8) and 10 CFR 50.34(f), NuScale compliance with this provision is neither appropriate nor required. Provisions for simulator capability are the responsibility of the COL applicant referencing the certified design.	Not Applicable
50.34(f)(2)(ii)	Establish a program to improve plant procedures, with the program scope to include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts (I.C.9)	Not Applicable	Applicable	The plant procedure improvement program specified by this requirement (and development of plant procedures) is not included within the scope of the DCA, but rather is the responsibility of the COL applicant referencing the certified design. Thus, this requirement is not technically relevant to the NuScale standard plant design, and consistent with 10 CFR 52.47(a)(8) and 10 CFR 50.34(f), NuScale compliance with this provision is neither appropriate nor required. However, the NuScale application contains information related to reliability assurance and human factors engineering that the COL applicant will use in establishing the procedure improvement program specified in this requirement.	Not Applicable
50.34(f)(2)(iii)	Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts (I.D.1)	Conforms	Applicable	None.	18.7

**Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933) (Continued)**

Item	Regulation Description / Title	Conformance Status	COL Applicability	Comments	Section
50.34(f)(2)(iv)	Provide a plant safety parameter display console (I.D.2)	Conforms	Applicable	Similar to previous design certification holders for which exemptions have been granted, the NuScale safety display and indication system is integrated into the control room human-system interface design rather than having a separate console. However, based on recent precedent, NuScale has concluded that integration of the safety display and indication system into the control room human-system interface design will not require an exemption from 10 CFR 50.34(f)(2)(iv).	7.2.13 18.7.2
50.34(f)(2)(v)	Provide for automatic indication of the bypassed and operable status of safety systems (I.D.3)	Conforms	Applicable	None.	7.2.4 7.2.13 8.3.1 8.3.2
50.34(f)(2)(vi)	Provide the capability of high point venting of noncondensable gases from the reactor coolant system, 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 loss-of-coolant accident or an unacceptable challenge to containment integrity. (II.B.1)	Departure	Not Applicable	The venting of noncondensable gases is unnecessary to ensure long term core cooling capability. Therefore, the NuScale design supports an exemption from the requirements of 50.34(f)(2)(vi), as well as the substantively equivalent requirements of 10 CFR 50.46a.	5.4.12
50.34(f)(2)(vii)	Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, and design as necessary to permit adequate access (II.B.2)	Conforms	Applicable	None.	12.2 12.3.1 12.4
50.34(f)(2)(viii)	Provide capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials (II.B.3)	Partially Conforms	Applicable	As described by SRP 9.3.2, I.6, and RG 1.206, C.I.9.3.2, a post-accident sampling system is not required provided that the guidance provided in SRP 9.3.2 for utilizing the normal process sampling system (post-accident) has been satisfied.	9.3.2 11.5 12.4

**Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933) (Continued)**

Item	Regulation Description / Title	Conformance Status	COL Applicability	Comments	Section
50.34(f)(2)(ix)	Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction (II.B.8)	Not Applicable	Not Applicable	Hydrogen control system is unnecessary based on analysis of 100% fuel-clad reaction.	3.8 19.2
50.34(f)(2)(x)	Provide a test program and associated model development, and conduct tests to qualify reactor coolant system relief and safety valves and, for PWRs, PORV block valves (II.D.1)	Partially Conforms	Applicable	This requirement is applicable to the DCA except for aspects specifying PORV block valve testing. The NuScale design does not use power-operated relief valves.	5.2.2
50.34(f)(2)(xi)	Provide direct indication of relief and safety valve position (open or closed) in the control room (II.D.3)	Conforms	Applicable	None.	6.3.1 7.2.13



**Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933) (Continued)**

Item	Regulation Description / Title	Conformance Status	COL Applicability	Comments	Section
50.34(f)(2)(xii)	Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide AFW system flow indication in the control room (II.E.1.2)	Not Applicable	Not Applicable	The NuScale design does not involve an AFW system as would be found at a typical large LWR. Also, while the DHR system performs some of the functions of an AFW system at a large PWR, the NuScale DHR system is designed for NuScale-specific transients and system characteristics, and its actuation and indication is designed accordingly. Specifically with regard to the portion of this requirement specifying control room flow indication, the DHR system operation involves passive natural circulation flow, with flow characteristics that inherently vary with system conditions, which makes DHR system flow a less useful measurement. for the NuScale design. Control room indication for system parameters other than DHR system flow are more appropriate to ensure operators have the information necessary to adequately monitor DHR system operation and reactor core cooling. These parameters include DHR system pressure, DHR passive condenser level, DHR system valve position indication, and reactor coolant system pressure and temperature. As discussed with respect to 10 CFR 50.34(f)(1)(ii), above, 10 CFR 50.34(f)(2)(xii) is not considered applicable to the NuScale DHR system. Because the literal language and intent of 10 CFR 50.34(f)(1)(ii) do not apply, the requirement is not technically relevant to the NuScale design. Therefore, an exemption would be unnecessary because 10 CFR 50.34(f)(1)(ii) only applies to the technically relevant portions of the TMI requirements.	7.2.13

**Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933) (Continued)**

Item	Regulation Description / Title	Conformance Status	COL Applicability	Comments	Section
50.34(f)(2)(xiii)	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 (II.E.3.1)	Departure	Not Applicable	The NuScale design equivalent to hot standby condition as stated in 10 CFR 50.34(f)(2)(xiii) is hot shutdown condition. The NuScale design does not rely on pressurizer heaters to establish and maintain natural circulation in hot shutdown conditions. The NuScale design supports an exemption from 50.34(f)(2)(xiii).	5.4.5 8.3.1 8.3.2
50.34(f)(2)(xiv)	Provide containment isolation systems that (A) ensure all non-essential systems are isolated automatically ; (B) ensure 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) use a containment set point pressure for initiating containment isolation as low as is compatible with normal operation; and (E) include automatic closing on a high radiation signal for all systems that provide a path to the environs (II.E.4.2)	Departure	Applicable	The NuScale design does not include systems that provide a direct path from the containment to the environs. The NuScale design supports an exemption from 10 CFR 50.34(f)(2)(xiv)(E) as applied to CES.	5.2.5 6.2.4 7.1.5 7.2.13 9.3.6
50.34(f)(2)(xv)	Capability for containment purging/venting designed to minimize the purging time consistent with as low as reasonably achievable (ALARA) (II.E.4.4)	Not Applicable	Not Applicable	The NuScale containment vessel is significantly smaller than a typical containment building, does not contain sub-compartments and does not require or incorporate a purge or venting system function as contemplated by this requirement. Personnel access during reactor operation is not needed. In addition, the NuScale ECCS design does not include pumps, and does not involve a typical PWR ECCS recirculation mode where ECCS pump performance relies on containment pressure. Thus purge or vent capability as prescribed by 10 CFR 50.34(f)(2)(xv) is neither required nor included in the NuScale design. This requirement is not technically relevant to the NuScale design and no exemption is necessary.	Not Applicable

**Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933) (Continued)**

Item	Regulation Description / Title	Conformance Status	COL Applicability	Comments	Section
50.34(f)(2)(xvi)	Establish design criterion for the allowable number of actuation cycles of the ECCS and reactor protection system with the expected occurrence rates of severe overcooling events (II.E.5.1)	Not Applicable	Not Applicable	This requirement explicitly states its applicability only to Babcock and Wilcox (B&W) designs. Based on NUREG-0933, this applicability was the result of unique sensitivity that B&W reactor designs exhibited to secondary system transients (both undercooling and overcooling events). The NuScale design does not exhibit such sensitivity.	Not Applicable
50.34(f)(2)(xvii)	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 (II.F.1)	Conforms	Applicable	None.	6.2.1 6.3.1 7.2.13 11.5 12.3.4
50.34(f)(2)(xviii)	Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling (II.F.2)	Conforms	Applicable	None.	6.3 7.2.13 18.7
50.34(f)(2)(xix)	Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage (II.F.3)	Conforms	Applicable	None.	6.2.1 7.2.13
50.34(f)(2)(xx)	Provide power supplies for pressurizer relief valves, block valves, and level indicators (II.G.1)	Departure	Not Applicable	The requirements of 10 CFR 50.34(f)(2)(xx) for power supplies for pressurizer relief valves and block valves are not technically relevant to the NuScale design. The NuScale design supports an exemption for the pressurizer level indicator vital power supply portion of 50.34(f)(2)(xx).	5.4.5 7.2.13 8.1.4 8.3.1

**Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933) (Continued)**

Item	Regulation Description / Title	Conformance Status	COL Applicability	Comments	Section
50.34(f)(2)(xxi)	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 (II.K.1.22)	Not Applicable	Not Applicable	This requirement explicitly states its applicability only to BWR designs. This applicability reflects aspects of a loss of feedwater transient that are unique to BWRs. However, the NuScale DHR system is designed such that necessary automatic and manual actions are taken to ensure proper functioning when the main feedwater system is not available.	Not Applicable
50.34(f)(2)(xxii)	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 (II.K.2.9)	Not Applicable	Not Applicable	This requirement explicitly states its applicability only to B&W plant designs. This applicability reflects aspects of the B&W ICS design that were identified following the TMI incident as design/reliability deficiencies, and are not pertinent to the NuScale design.	Not Applicable
50.34(f)(2)(xxiii)	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 (II.K.2.10)	Not Applicable	Not Applicable	This requirement explicitly states its applicability only to B&W plant designs. This applicability reflects design aspects unique to B&W BWRs.	Not Applicable
50.34(f)(2)(xxiv)	Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements (II.K.3.23)	Not Applicable	Not Applicable	This requirement explicitly states its applicability only to BWR designs. As a PWR, the NuScale design is fundamentally different than that of a BWR, such that the post-accident reactor vessel water level recording contemplated by this requirement is not pertinent.	Not Applicable
50.34(f)(2)(xxv)	Provide an onsite Technical Support Center and onsite Operational Support Center (III.A.1.2)	Partially Conforms	Applicable	None.	13.3
50.34(f)(2)(xxvi)	Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term radioactive materials (III.D.1.1)	Partially Conforms	Applicable	This requirement is applicable to the DCA to the extent it is relevant to the standard plant design. Aspects of this requirement that are pertinent to testing and operational programs are the responsibility of the COL applicant. These aspects are not technically relevant to the NuScale certified design, and consistent with 10 CFR 52.47(a)(8) and 10 CFR 50.34(f), NuScale compliance is neither appropriate nor required.	5.4 6.3.1 9.3.2 9.3.4

**Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933) (Continued)**

Item	Regulation Description / Title	Conformance Status	COL Applicability	Comments	Section
50.34(f)(2)(xxvii)	Provide for monitoring of in-plant radiation and airborne radioactivity (III.D.3.3)	Conforms	Applicable	None.	11.5 11.6 12.3.4
50.34(f)(2)(xxviii)	Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in an accident source term release (III.D.3.4)	Conforms	Applicable	None.	6.4.1 6.4.4 15.0
50.34(f)(3)(i)	Provide administrative procedures for evaluating operating, design, and construction experience (I.C.5)	Partially Conforms	Applicable	This requirement is applicable to the DCA to the extent it is relevant to the standard plant design. Aspects of this requirement that are pertinent to site-specific design, operational programs, and construction details are the responsibility of the COL applicant. These aspects are not technically relevant to the NuScale certified design, and consistent with 10 CFR 52.47(a)(8) and 10 CFR 50.34(f), NuScale compliance is neither appropriate nor required.	13.2 18.8
50.34(f)(3)(ii)	Ensure that the QA list required by Criterion II in Appendix B to 10 CFR 50 includes all SSCs important to safety (I.F.1)	Conforms	Applicable	None.	3.2 17.4
50.34(f)(3)(iii)	Establish a QA Program based on the specified considerations (I.F.2)	Partially Conforms	Applicable	This requirement is applicable to the DCA to the extent it is relevant to design activities in support of the DCA. Aspects of this rule specifying QA program requirements for site-specific design and analysis, operational programs, as-built documentation, and construction and installation are the responsibility of the COL applicant. These aspects are not technically relevant to the NuScale certified design, and consistent with 10 CFR 52.47(a)(8) and 10 CFR 50.34(f), NuScale compliance is neither appropriate nor required.	17.5

**Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933) (Continued)**

Item	Regulation Description / Title	Conformance Status	COL Applicability	Comments	Section
50.34(f)(3)(iv)	Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot-diameter opening (II.B.8)	Partially Conforms	Applicable	This TMI requirement is based on traditional large LWR containment designs and the potential, as of the time of the requirement, need for future containment venting systems to accommodate severe accidents. The NuScale containment vessel design is significantly different than that of a typical LWR containment structure, the NuScale containment design is substantially different from traditional designs because of its high-pressure capability. Moreover, the NuScale design accounts for severe accidents and does not require containment venting to safely mitigate them. A 3-foot opening relative to the NuScale containment would be infeasible to accommodate and is unnecessary. Furthermore, should any future development identify a need for a new penetration, adding such a penetration to the NuScale vessel is a substantially different process versus the typical containment.	6.2
50.34(f)(3)(v)	Preliminary Design Information - Containment Integrity (II.B.8)	Not Applicable	Not Applicable	Pursuant to 10 CFR 52.47(a)(8) and 10 CFR 50.34(f), Paragraph (f)(3)(v) is excluded from the information required to be included in an application for a design certification.	Not Applicable
50.34(f)(3)(vi)	For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations (II.E.4.1)	Not Applicable	Not Applicable	The NuScale design does not have external hydrogen recombiners.	Not Applicable
50.34(f)(3)(vii)	Provide a description of the management plan for design and construction activities (II.J.3.1)	Not Applicable	Not Applicable	This requirement is applicable only to applicants and holders of reactor facility licenses. As discussed in NUREG-0933, the underlying purpose of this requirement was to require COL applicants and licensees to improve the oversight of design, construction, and modification activities so that they would gain the critical expertise necessary for the safe operation of the plant.	Not Applicable

**Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933) (Continued)**

Item	Regulation Description / Title	Conformance Status	COL Applicability	Comments	Section
Issue 191	Assessment of Debris Accumulation on PWR Sump Performance	Conforms	Applicable	Post-LOCA debris generation and its affects on ECCS and LTCC are fully evaluated and addressed.	6.3
Issue 193	Boiling Water Reactor Emergency Cooling Water System (ECCS) Suction Concerns	Not Applicable	Not Applicable	This Issue is specific to boiling water reactors.	Not Applicable
Issue 199	Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern U.S. on Existing Plants	Not Applicable	Not Applicable	This is applicable to currently operating plants.	Not Applicable
Issue 204	Flooding of Nuclear Power Plant Sites Following Upstream Dam Failures	Not Applicable	Applicable	The information governed by this guidance is site-specific.	Not Applicable

**Table 1.9-6: Evaluation of Operating Experience (Generic Letters and Bulletins)**

Doc ID	Title	Conformance Status	COL Appl.	Comments	Section
Generic Letter 88-14	Instrument Air Supply System Problems Affecting Safety-Related Equipment	Conforms	Applicable	The IAS furnishes both instrument and service air. IAS moisture separators and dryer packages ensure that the instrument air supplied is dry in accordance with the quality standards of ANS/ISA S7.3-R1981.	9.3.1
Generic Letter 88-15	Electric Power Systems - Inadequate Control Over Design Processes	Partially Conforms	Applicable	Portions relevant to the NuScale passive plant design are considered in the design of electrical systems.	8.1.4 8.3.1 8.3.2
Generic Letter 91-06	Resolution of Generic Issue A30, Adequacy Of Safety-Related DC Power Supplies Pursuant to 10 CFR 50.54(f)	Partially Conforms	Applicable	No safety-related DC systems; however, relevant portions are considered in the design of the non-Class 1E EDSS.	8.1.4 8.3.2
Generic Letter 96-01	Testing of Safety-Related Logic Circuits	Conforms	Applicable	None.	7.2.2 7.2.15 8.1.4
Generic Letter 2006-02	Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power	Not Applicable	Applicable	The NuScale Power Plant does not rely on offsite power for safety-related or risk-significant functions. Grid stability studies are the responsibility of a COL applicant that references the NuScale design certification.	8.2.1 8.2.3
Generic Letter 2007-01	Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients.	Partially Conforms	Applicable	Portions relevant to the NuScale passive plant design are considered in the design of electrical systems.	8.1.4 8.2.1 8.3.1
Generic Letter 2008-01	Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems	Partially Conforms	Applicable	NuScale has determined that gas accumulation buildup will not impact ECCS under accident conditions. DHRS does not interface with the RCS. It is connected to the secondary system; at present this has not been addressed by NuScale.	5.4 Ch 6
Bulletin 2007-01	Security Officer Attentiveness	Not Applicable	Applicable	Applicable to holders of operating licenses for nuclear power reactors.	Not Applicable



**Table 1.9-6: Evaluation of Operating Experience (Generic Letters and Bulletins) (Continued)**

Doc ID	Title	Conformance Status	COL Appl.	Comments	Section
Bulletin 2011-01	Mitigating Strategies	Not Applicable	Not Applicable	Bulletin 2011-01 was addressed to existing Licensees. It required the Licensee to “confirm continue compliance with 10 CFR 50.54(hh)(2)...”. The 10 CFR 50.54(hh)(2) is addressed in Section 20.2.	19.4 Ch 20
Bulletin 2012-01	Design Vulnerability in Electric Power System	Partially Conforms	Applicable	Consideration of this bulletin is demonstrated by the conformance with SRP BTP 8-9, which is described in Section 8.2.3.	8.2.3

**Table 1.9-7: Conformance with Advanced and Evolutionary Light Water Reactor Design Issues (SECYs and Associated SRMs)**

Doc ID	Title	Conformance Status	COL Applicability	Comments	Section
SECY-89-013	Design Requirements Related to the Evolutionary Advanced Light Water Reactors	Conforms	Applicable	Addressed through SECY-90-016 and SECY-93-087. See Table 1.9-8 for further information.	-
SECY-90-016	Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements	Partially Conforms	Applicable	This SECY was directed towards evolutionary ALWR designs. The applicability of certain SECY-90-016 issues to passive plants was later established in SECY-93-087 and/or SECY-94-084. As a passive ALWR design, the NuScale design conforms to the passive plant guidance of SECY-93-087 and SECY-94-084, rather than that of SECY-90-016. See Table 1.9-8 for further information.	19.1 19.2
SECY-90-241	Level of Detail Required for Design Certification under Part 52	Conforms	Applicable	Incorporated into 10 CFR 52 and implementing NRC guidance documents.	-
SECY-90-377	Requirements for Design Certification under 10 CFR Part 52	Conforms	Applicable	Incorporated into 10 CFR 52 and implementing NRC guidance documents.	-
SECY-91-074	Prototype Decisions for Advanced Reactor Designs	Conforms	Applicable	Incorporated into 10 CFR 52 and implementing NRC guidance documents.	-
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	Not Applicable	Not Applicable	SECY-91-078 pertains to evolutionary ALWR designs and is not directly applicable to passive plant designs.	8.1.4
SECY-91-178	ITAAC for Design Certifications and Combined Licenses	Conforms	Applicable	Incorporated into 10 CFR 52 and implementing NRC guidance documents.	14.3.2
SECY-91-210	ITAAC Requirements for Design Review and Issuance of FDA	Conforms	Applicable	Incorporated into 10 CFR 52 and implementing NRC guidance documents.	-
SECY-91-229	Severe Accident Mitigation Design Alternatives for Certified Standard Designs	Conforms	Not Applicable	Incorporated into NRC Orders, regulatory guidance, and pending rulemaking.	19.2.6
SECY-91-262	Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light-Water Reactor (LWR) Designs	Conforms	Applicable	Incorporated into NRC Orders, regulatory guidance, and pending rulemaking.	-
SECY-92-053	Use of Design Acceptance Criteria During the 10 CFR Part 52 Design Certification Reviews	Conforms	Applicable	Incorporated into NRC Orders, regulatory guidance, and pending rulemaking.	14.3.6

**Table 1.9-7: Conformance with Advanced and Evolutionary Light Water Reactor Design Issues (SECYs and Associated SRMs) (Continued)**

Doc ID	Title	Conformance Status	COL Applicability	Comments	Section
SECY-92-092	The Containment Performance Goal, External Events Sequences, and the Definition of Containment Failure for Advanced LWRs	Conforms	Applicable	Incorporated into NRC Orders, regulatory guidance, and pending rulemaking.	-
SECY-93-087	Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs	See Table 1.9-8.	See Table 1.9-8.	None.	-
SECY-94-084	Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Design (RTNSS)	Partially Conforms	Applicable	Incorporated into 10 CFR 52 and implementing NRC guidance documents. The NuScale Fire Protection System does not contain any RTNSS equipment. However, Section C, Safe Shutdown Requirements, of the SECY discusses the stable shutdown condition for passive ALWR which is applicable to the NuScale Power Plant.	5.4 8.1.4 8.3.1 8.3.2 8.4 9.2.5 Appendix 9A 15.0.4 19.3
SECY-94-302	Source-Term-Related Technical and Licensing Issues Relating to Evolutionary and Passive Light-Water-Reactor Designs	Conforms	Applicable	Incorporated into 10 CFR 52 and implementing NRC guidance documents.	-
SECY-95-132	Policy and Technical Issues Associated with Regulatory Treatment of Non-Safety Systems in Passive Plant Designs	Conforms	Applicable	Incorporated into 10 CFR 52 and implementing NRC guidance documents.	8.1.1 8.1.4 8.3.1 8.3.2 8.4 19.3
SECY-14-038	Performance-Based Framework for Nuclear Power Plant Emergency Preparedness Oversight	Conforms	Applicable	None.	13.3
SECY-14-088	Proposed Options to Address Lessons-Learned Review of the U.S. Nuclear Regulatory Commissions Force-On-Force Inspection Program in Response to Staff Requirements Memorandum - COMGEA/COMWCO-14-0001	Not Applicable	Applicable	Site-specific requirements.	Not Applicable

**Table 1.9-8: Conformance with SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs"**

Issue	Description	Conformance Status	COL Applicability	Comments	Section
I.A	Use of a Physically-Based Source Term: Incorporation of engineering judgment and a more realistic source term in design that deviates from the siting requirements in 10 CFR 100.	Conforms	Applicable	None.	15.0.3
I.B	Anticipated Transient without SCRAM (ATWS): Position on the current practices and design features to achieve a high degree of protection against an ATWS.	Partially Conforms	Applicable	NuScale submitted a white paper that describes the underlying purpose of the rule which is to reduce the risk from ATWS events (NuScale Power Plant Design for ATWS and 10 CFR 50.62 Regulatory Compliance, NP-ER-0000-2196, September 18, 2013). The proposed treatment of the rule was described in a presentation to the NRC, Design for ATWS and 10 CFR 50.62 Regulatory Compliance, PM-0114-5922-P, February 26, 2014. The NuScale design relies on diversity within the module protection system (MPS) to reduce the risk associated with ATWS events.	15.8
I.C	Mid-Loop Operation: Position on design features necessary to ensure a high degree of reliability of RHR systems in PWR.	Not Applicable	Not Applicable	Design does not use external loops and no drain down condition for refueling.	19.2
I.D	Station Blackout (SBO): Position on methods to mitigate the effects of a loss of all AC power.	Not Applicable	Not Applicable	The relevance of the SECY-90-016 SBO issue to passive ALWR designs was deferred to and addressed in Section F of SECY-94-084 and SECY-95-132. The NuScale design conforms to the passive plant guidance these documents.	8.4
I.E	Fire Protection: Positions on design configuration and features the fire protection system and other management schemes to ensure safe shutdown of the reactor.	Conforms	Applicable	None.	Appendix 9A

**Table 1.9-8: Conformance with SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs" (Continued)**

Issue	Description	Conformance Status	COL Applicability	Comments	Section
I.F	Intersystem LOCA: Position on acceptable design practices and preventative measures to minimize the probability of an ISLOCA.	Conforms	Applicable	None.	19.2
I.G	Hydrogen Control: Position on acceptable requirements to measure and mitigate the effects of hydrogen produced due to a water reaction with zirconium fuel cladding.	Partially Conforms	Applicable	Licensing Awareness.	6.2.5
I.H	Core Debris Coolability: Acceptability criteria for cooling area and quenching ability regarding corium interaction with concrete.	Conforms	Not Applicable	None.	19.2
I.I	High-Pressure Core Melt Ejection: Position on acceptable design features to prevent the event of a high-pressure core melt ejection.	Conforms	Not Applicable	None.	19.2
I.J	Containment Performance: Position on acceptable conditional containment failure probabilities or other analyses to ensure a high degree of protection from the containment.	Conforms	Not Applicable	None.	19.1 19.2
I.K	Dedicated Containment Vent Penetration: Position for a dedicated vent penetration to preclude containment failure resulting from a containment over-pressurization event.	Conforms	Not Applicable	None.	19.2
I.L	Equipment Survivability: Position on the applicability of environmental qualification and quality assurance requirements related to plant features provided only for severe-accident protection.	Conforms	Not Applicable	None.	19.2
I.M	Elimination of Operating-Basis Earthquake: Position on the applicability of the OBE in design and the possibility of decoupling the OBE and SSE in the design of safety systems.	Conforms	Applicable	By setting the OBE to 1/3 of the SSE it is decoupled from the design evaluation process.	3.7
I.N	In-Service Testing of Pumps and Valves: Position on periodic testing to confirm operability of safety-related pumps and valves.	Conforms	Applicable	None.	3.9.6
II.A	Industry Codes and Standards: Position on use of recently developed or modified design codes and industry standards in ALWR designs that have not been reviewed for acceptability by the NRC.	Conforms	Applicable	NuScale use the latest endorsed codes and standards or others on case by case basis.	all
II.B	Electrical Distribution: Positions originally addressed by SECY-91-078 that specified that an evolutionary ALWR provide: (1) an alternate power source to nonsafety-related loads, and (2) at least one offsite circuit connected directly to each redundant safety division with no intervening nonsafety-related buses.	Not Applicable	Not Applicable	The NuScale electrical system design conforms to the passive plant guidance of SECY-94-084, Section G.	8.1.4
II.C	Seismic Hazard Curves and Design Parameters: Position on use of proposed generic bounding seismic hazard curves and performance of seismic PRA.	Conforms	Applicable	None.	19.1
II.D	Leak-Before-Break: Position on use of leak-before-break concept.	Conforms	Applicable	LBB is applied to the MS and FW lines inside containment.	3.6.3

**Table 1.9-8: Conformance with SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs" (Continued)**

Issue	Description	Conformance Status	COL Applicability	Comments	Section
II.E	Classification of Main Steam Lines in BWRs: Position on the staffs defined approach for seismic classification of the main steam line in both evolutionary and passive BWRs.	Not Applicable	Not Applicable	Applicable to BWRs.	Not Applicable
II.F	Tornado Design Basis: Position on the maximum tornado wind speed to be used for a design basis tornado.	Conforms	Applicable	The FSAR uses the maximum tornado wind speed.	3.3
II.G	Containment Bypass: Position on ALWR design against containment bypass. Specifically, failure of the containment system to channel fission product releases through the suppression pool, or the failure of passive containment cooling heat exchanger tubes in large pools of water outside containment.	Conforms	Applicable	None.	15.0.3 19.1 19.2
II.H	Containment Leak Rate Testing: Position on testing duration for Type C leak rate testing (prior to rule change).	Partially Conforms	Applicable	None.	6.2.6
II.I	Post-Accident Sampling System: Position on the required capability to analyze dissolved hydrogen, oxygen, and chloride in accordance with applicable regulations.	Conforms	Applicable	As described by SRP 9.3.2, I.6, and RG 1.206, C.I.9.3.2, a post-accident sampling system is not required provided that the guidance provided in SRP 9.3.2 for utilizing the normal process sampling system (post-accident) has been satisfied.	9.3.2
II.J	Level of Detail: Position on a design certification submittal with depth of detail similar to that in an FSAR.	Conforms	Applicable	None.	All FSAR Sections
II.K	Prototyping: No guidance provided; information only	Conforms	Applicable	None.	1.5
II.L	ITAAC: Position on providing ITAAC to demonstrate that a nuclear power plant referencing a certified design is built and operates consistent with the design certification.	Conforms	Applicable	None.	Tier 1 14.3
II.M	Reliability Assurance Program: Position on providing a description of purpose, scope, objectives, and implementation of a design reliability assurance program.	Conforms	Applicable	None.	17.4
II.N	Site-Specific PRAs and Analyses of External Events: Position on the inclusion of external event analysis beyond the design basis that needs to be addressed as part of the plant PRA during the design certification review.	Conforms	Applicable	None.	19.1
II.O	Severe Accident Mitigation Design Alternatives (SAMDA): Position on the consideration of SAMDA as part of the final design approval/design certification of an advanced reactor.	Conforms	Applicable	None.	19.2.6

**Table 1.9-8: Conformance with SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs" (Continued)**

Issue	Description	Conformance Status	COL Applicability	Comments	Section
II.P	Generic Rulemaking Related to Design Certification: No guidance provided; information only.	Not Applicable	Not Applicable	Information Only.	Not Applicable
II.Q	Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems: Position on the use of defense-in-depth and diversity of instrumentation and control systems as part of the final design approval/design certification of an advanced reactor.	Conforms	Applicable	None.	7.1.5
II.R	Multiple SG Tube Ruptures: Position on requiring that analysis of multiple SGTRs of 2 to 5 SG tubes be included in the application for design certification of passive ALWRs.	Conforms	Applicable	None.	15.6 19.1
II.S	PRA Beyond Design Certification: Position on requiring conversion of the design certification PRA into a plant-specific PRA	Conforms	Applicable	None.	19.1
II.T	Control Room Annunciator (Alarm) Reliability: Position on recommending that additional requirements for ALWR alarm systems are necessary to minimize the problems experienced by operating nuclear power plants	Conforms	Applicable	None.	7.2.13
III.A	Regulatory Treatment of Active Nonsafety Systems in Passive Designs: Position on the proposed staff approach for resolving the regulatory treatment of the active non-safety systems in passive ALWRs.	Conforms	Applicable	None.	19.3
III.B	Definition of Passive Failure: Position on the staff redefining some passive failures of components as active failures (i.e., check valves) to cause valves to be evaluated in a much more stringent manner than in previous licensing review	Conforms	Applicable	None.	15.0.0
III.C	Thermal-Hydraulic Stability of the SBWR	Not Applicable	Not Applicable	BWR requirement.	Not Applicable

**Table 1.9-8: Conformance with SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs" (Continued)**

Issue	Description	Conformance Status	COL Applicability	Comments	Section
III.D	Safe Shutdown Requirements: Position on using non-safety grade active cooling systems to bring a reactor to cold shutdown since non-safety RHR systems do not comply with the guidance of 1.139 or branch technical position 5-1	Conforms	Applicable	The provisions of this SECY are met by using the non-safety related containment flood and drain system to flood the containment to allow cooldown to cold conditions for disconnection and transfer of NPMs. During shutdown and NPM movement, residual and decay heat removal is provided by heat convection and conduction from the reactor to the reactor pool via the RCS, flooded containment, and the RPV and containment vessel walls.	3.1.4 5.4.3 7.1
III.E	Control Room Habitability: Position on appropriate analytical methods (i.e., dose limits and accident duration) to be used in determining the acceptability criteria for control room habitability in accordance with regulatory standards.	Conforms	Applicable	None.	15.0.3
III.F	Radionuclide Attenuation: Position on fission product removal processes inside containment by natural effects and holdup by the secondary building and piping systems in addition to commission position on containment spray systems for passive ALWRs.	Conforms	Applicable	None.	6.5.3 15.0.3
III.G	Simplification of Offsite Emergency Planning: Position on simplifying off-site emergency planning of passive designs due to the estimated low probability of core damage of such designs.	Conforms	Applicable	None.	13.3
III.H	Role of the Passive Plant Control Room Operator: Commission position on sufficient man-in-the-loop testing and evaluation to be performed and that a fully functional integrated control room prototype is necessary for passive plant control room designs to demonstrate that functions and tasks are integrated properly into the man/machine interface decisions.	Conforms	Applicable	None.	18.7 18.10



**1.10 Nuclear Power Plants to be Operated on Multi-Unit Sites**

COL Item 1.10-1: A COL applicant that references the NuScale Power Plant design certification will evaluate the potential hazards resulting from construction activities of the new NuScale facility to the safety-related and risk significant structures, systems, and components of existing operating unit(s) and newly constructed operating unit(s) at the co-located site per 10 CFR 52.79(a)(31). The evaluation will include identification of any management and administrative controls necessary to eliminate or mitigate the consequences of potential hazards and demonstration that the limiting conditions for operation of an operating unit would not be exceeded. This COL item is not applicable for construction activities (build-out of the facility) at an individual NuScale Power Plant with operating NuScale Power Modules.