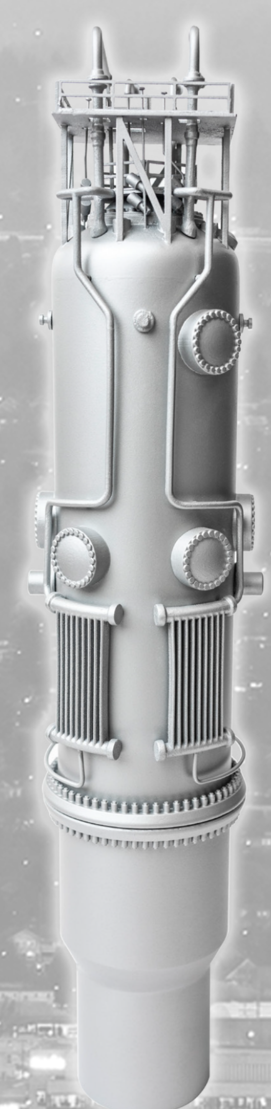
NuScale Standard Plant
Design Certification Application

Chapter One
Introduction and General Description of the Plant



PART 2 - TIER 2

Revision 5
July 2020

TABLE OF CONTENTS

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

LIST OF FIGURES

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 Power, LLC 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 instrumentation and controls (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".

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 containment vessel (CNV) is an American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Class MC (steel) containment that is designed, analyzed, fabricated, inspected, tested and stamped as an ASME BPVC 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 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 to12 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 percent 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 Regulatory Guide (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 convention 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 does not contain 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. 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

Acronym or Abbreviation	Description
AAC	alternate AC power
AAPS	auxiliary AC power source
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
AQ	augmented quality
ARM	area 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

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
BDG	backup diesel generator
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
BPVC	Boiler and Pressure Vessel Code
BRL	Ballistic Research Laboratory
BRVS	battery room ventilation system
BTP	Branch Technical Position
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
CCDF	conditional core damage frequency
CCDP	conditional core damage probability
CCF	common cause failure
CCFL	counter current flow limitation
CCFP	conditional containment failure probability
CDE	core damage event
CDF	core damage frequency
CDI	conceptual design information
CDM	certified design material
CDST	core damage source term
CEA	control element assembly
CES	containment evacuation system
CET	containment event tree
CEUS	central and eastern United States
CFD	computational fluid dynamics
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
CLRF	conditional large release frequency
CLRT	containment leakage rate testing
CMAA	Crane Manufacturers Association of America
CMS	code management software
CMTR	certified material test report

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
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
COMS	communication system
CPRS	condensate polisher resin regeneration 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
CUB	Central Utility Building
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

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
DMA	dimethylamine
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOE	Department of Energy
DOT	Department of Transportation
D-RAP	Design Reliability Assurance Program
DSRS	Design Specific Review Standard
DSS	digital safety system
DSW	dry solid waste
DTC	Doppler temperature coefficient, fuel temperature coefficient, Doppler coefficient
DWS	demineralized water system
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
EFPD	effective full-power days
EFPY	effective full-power years
EHVS	13.8 kV and switchyard system
EIM	equipment interface module
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	electrical penetration assembly
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

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
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
FCI	fuel-coolant interaction
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 NRC's 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
FSI	fluid-structure interaction
FSSA	fire safe shutdown analysis
FSSD	fire safe shutdown
FV	Fussell-Vesely
FW	feedwater
FWB	Fire Water Building
FWH	feedwater heater
FWIV	feedwater isolation valve
FWLB	feedwater line break
FWPB	feedwater pipe break

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
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
HELB	high-energy line break
HEP	human error probability
HEPA	high-efficiency particulate air
HFE	human factors engineering, human failure events
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, horsepower
HP-FWH	high pressure feedwater heater
HPM	human performance monitoring
HPME	high pressure melt ejection
HPN	health physics network
HRA	human reliability analysis
HRS	hardware requirement specification
HSI	human-system interface
HVAC	heating ventilation and air conditioning
HVDS	feedwater heater vents and drains system
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

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
IE	infrequent event, initiating 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, Instrument Society of America
ISG	interim staff guidance
ISI	inservice inspection
ISLH	inservice leak and hydro
ISLOCA	interfacing systems loss-of-coolant accident
ISM	independent support motion
ISO	International Organization for Standardization
ISRS	in-structure response spectra
IST	inservice 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
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 shutdown
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, liquid radwaste system
LSH	level switch, high
LSL	level switch, low

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
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
MCR	main control room
MCS	module control system
MCYR	module critical year
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
MLA	module lifting adapter
MLD	master logic diagram
MM	multiple, multi-module
MMAF	multi-module adjustment factor
MMI	multi-module issue
MMPSF	multi-module performance shaping factor
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
MSPB	main steam pipe break
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
N/A	Not Applicable
NDE	non-destructive examination

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
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
NOG	nuclear overhead and gantry
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
OER	operating experience review
OHLHS	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	piping 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

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
PDC	principal design criteria
PDIL	power dependent insertion limit
PDIT	differential pressure indicating transmitter
PFT	process feed tank
PGA	peak ground acceleration
pH_T	concentration of H+ ion on a logarithmic scale (temperature dependent)
PID	proportional integral derivative
PING	particulate, iodine, and noble gas
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
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

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
QPF	quadrant power fractions
RAI	request for additional information
RAP	Reliability Assurance Program
RAW	risk achievement worth
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
RHX	regenerative heat exchanger
RIS	regulatory issue summary
RLE	review level earthquake
RM	radiation monitoring
RMS	fixed area radiation monitoring system
RMTS	risk-managed technical specifications
RO	reverse osmosis
ROP	Reactor Oversight Process
RPCS	reactor pool cooling system
RPI	rod position indication
RPS	reactor protection system
RPV	reactor pressure vessel
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_{NDT}	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

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
RWB	Radioactive Waste Building
RWBCR	Radioactive Waste Building control room
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
SBLB	subscale boundary layer boiling
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
SCR	SCRAM load
SCS	secondary sampling system
SCWS	site cooling water system
SDB	safety data bus
SDIS	safety display and indication system
SDM	shutdown margin
SDOE	secure development and operational environment
SDOF	single-degree-of-freedom
SDP	software development process
SDS	site drainage system
SEB	Security Building
SECS	plant security system
SECY	Secretary of the Commission, Office of the NRC
SEI	Structural Engineering Institute
SEL	seismic equipment list
SER	Safety Evaluation Report
SFA	spent fuel assembly
SFM	safety function module
SFP	spent fuel pool
SFPCS	spent fuel pool cooling system

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
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 assessment
SMACNA	Sheet Metal and Air Conditioning Contractors' National Association
SME	subject matter expert
SMR	small modular reactor
SMS	seismic monitoring system
SNL	Sandia National Laboratories
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
SR	surveillance requirement
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 or secondary system isolation
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

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
SWYD	switchyard system
TA	task analysis
TAF	top of active fuel
TBC	Turbine Building crane
TBS	turbine bypass system
TBVS	Turbine Building HVAC system
T/C	thermocouple
TCU	temperature control unit
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
TLOSS	turbine lube oil storage system
TMI	Three Mile Island
TMR	triple module redundancy
T_{NDT}	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
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

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
WSW	wet solid waste
WTB	Waste Treatment Building
ZOI	Zone of Influence
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. Security-related information is delineated using double braces {{ }}. This information is withheld in accordance with 10 CFR 2.390(d)(1).

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.

6) Site Cooling Water System (SCWS): provides cooling water to plant auxiliary systems. Details associated with location and orientation of the cooling towers as well as equipment design and operation are site specific. Additional details of the SCWS are provided in Section 1.2.1.6.

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 percent 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.

The NPM is designed to perform normal power maneuvers. Electric power can be adjusted with turbine bypass to the condenser. In addition, core power maneuvering can be accomplished with control rods, soluble boron concentration changes, or a combination of control rods and soluble boron as described in Sections 4.2 and 4.3.

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 is used during normal plant operation to control reactivity. The shutdown bank is used during normal shutdown. All 16 CRAs are inserted for scram events.

The fuel assembly design is similar to a standard 17x17 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, UO_2, with gadolinium oxide, Gd_2O_3, as a burnable absorber homogeneously mixed within the fuel in select rod locations. The U-235 enrichment is less than 4.95 percent. 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 transfered 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 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 Inservice Inspection of Class 2 and 3 Components

The inservice 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 inservice 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 nonsafety 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 nonsafety 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. As described in Section 8.3, 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 nonsafety 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 1.2-2 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 1.2-2, 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 with a stainless steel surface and a vertical assembly comprised of a square stainless steel tube framing system and series of radiation panel assemblies that extends into the pool. The horizontal slab is bolted to the top of the bay. Nine radiation panels are attached on both sides of the vertical bioshield framing system to provide a radiation barrier and ventilation. 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.

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. The tunnel has two levels;

an upper tunnel for personnel access to the RXB and a lower tunnel that is a utilities tunnel between the CRB and the RXB.

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

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.

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 failure, 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 nonsafety 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 modules	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_2 (<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	Minimal amount of noncondensible 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.
	No insulation on RPV	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

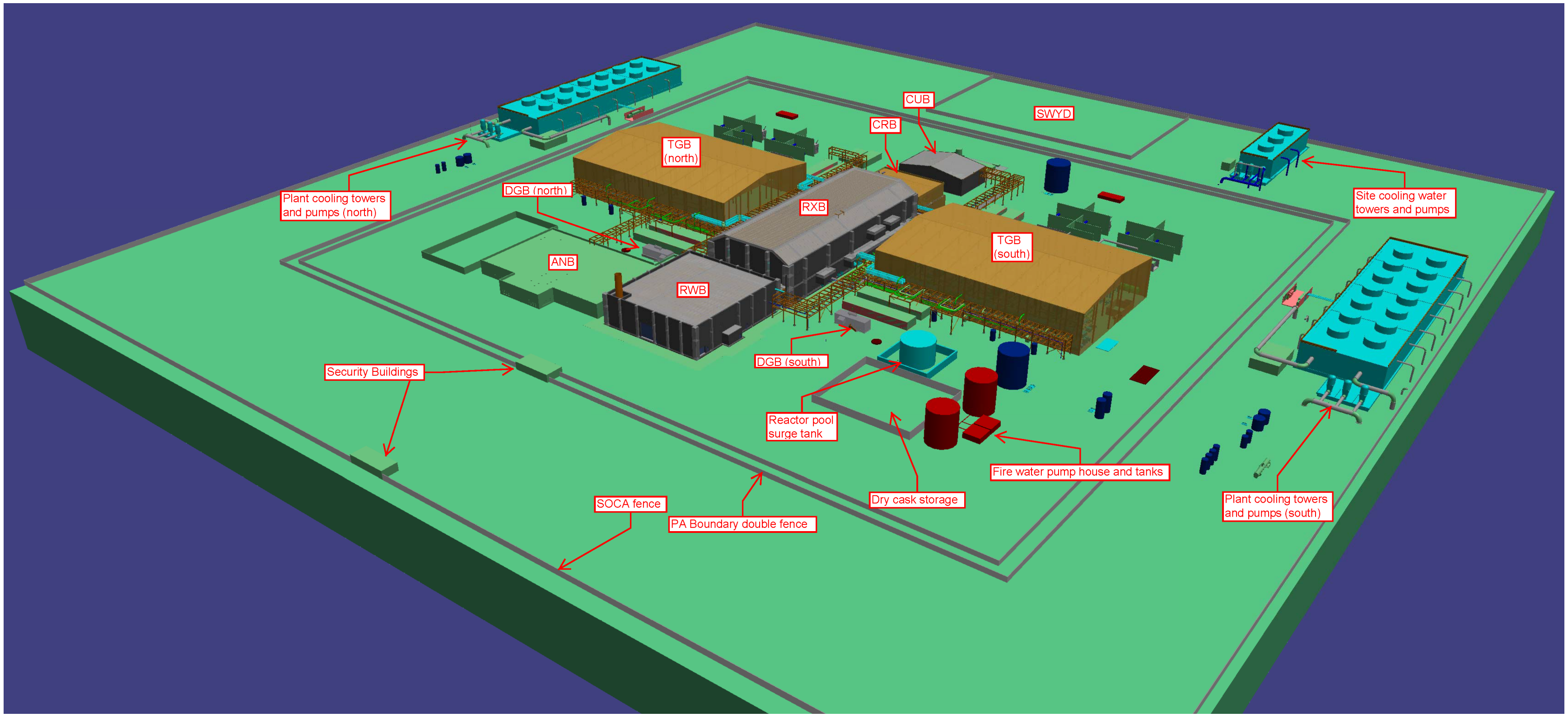


Figure 1.2-2: NuScale Functional Boundaries

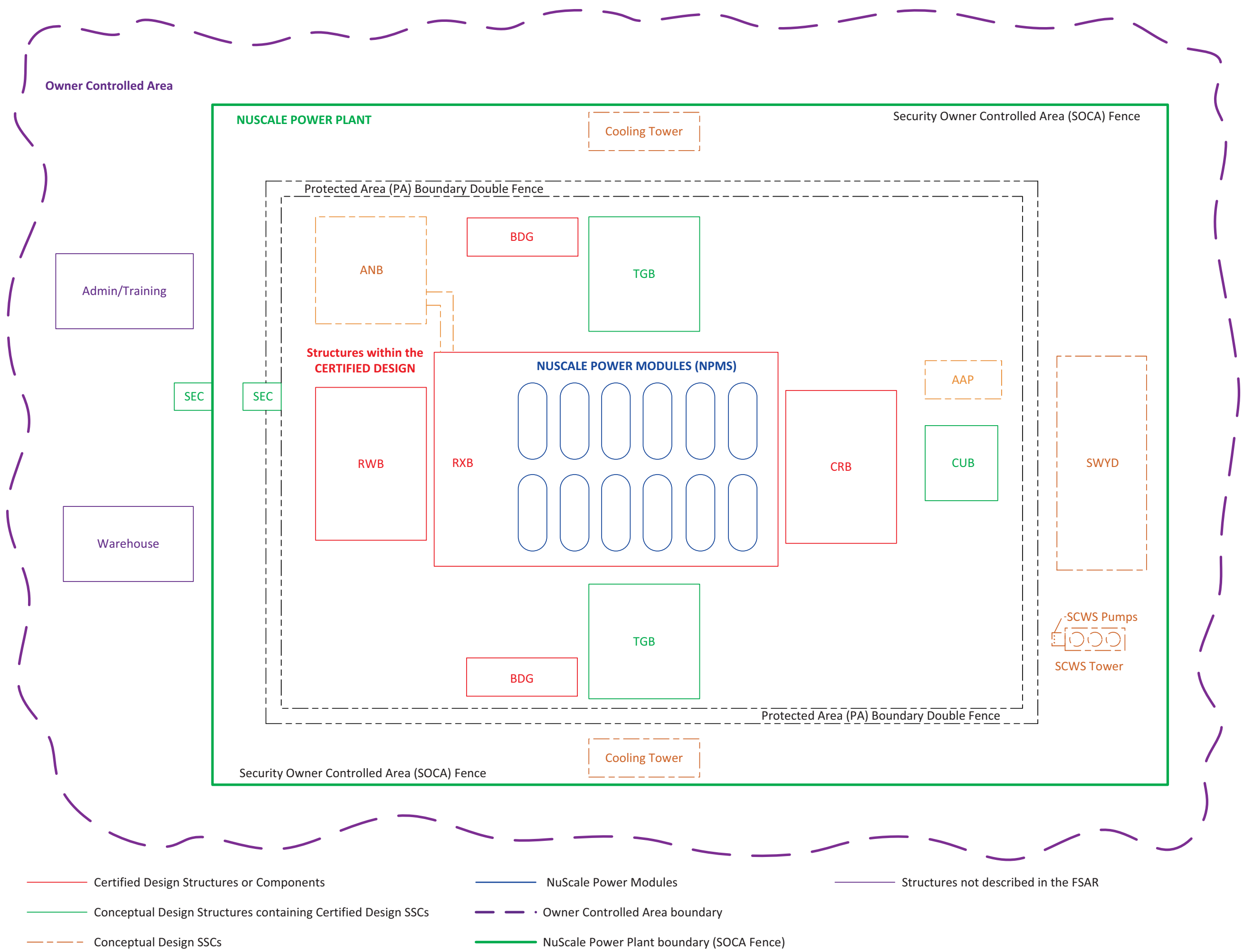


Figure 1.2-3: Schematic of a Single NuScale Power Module and Associated Secondary Equipment

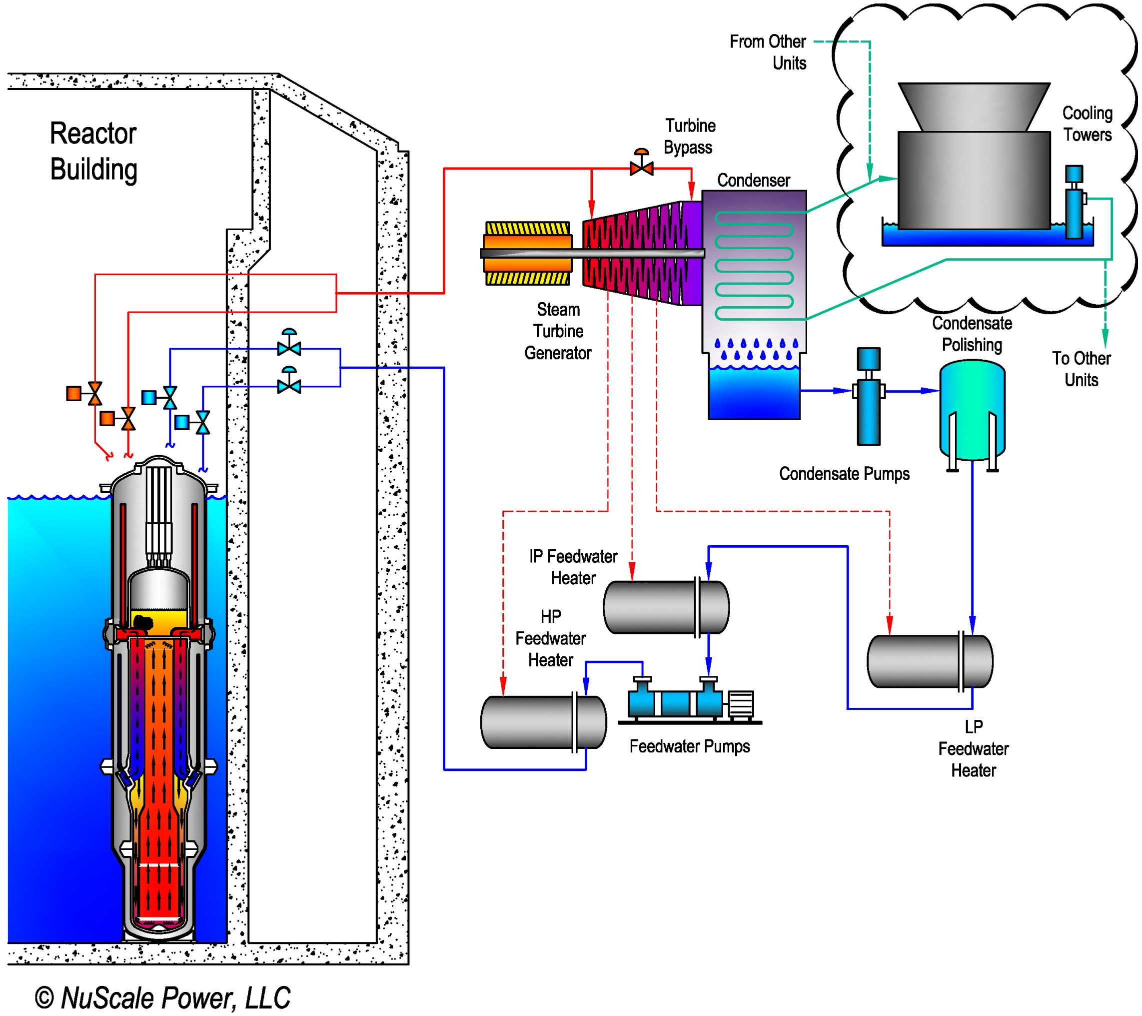


Figure 1.2-4: Layout of a Multi-Module NuScale Power Plant

{{ Withheld - See Part 9 }}

Figure 1.2-5: Cutaway Illustration of 12 Module Configuration

{{ Withheld - See Part 9 }}

Figure 1.2-6: Cutaway View of NuScale Power Module

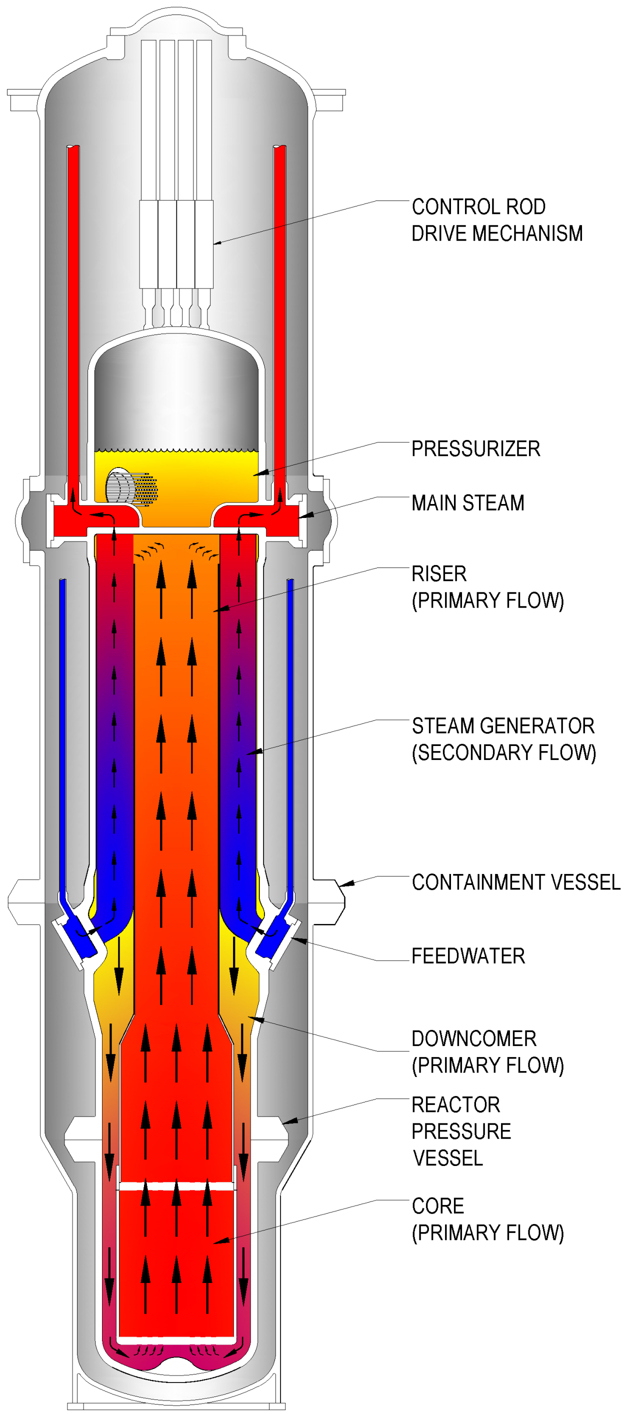


Figure 1.2-7: Steam Generator and Reactor Flow

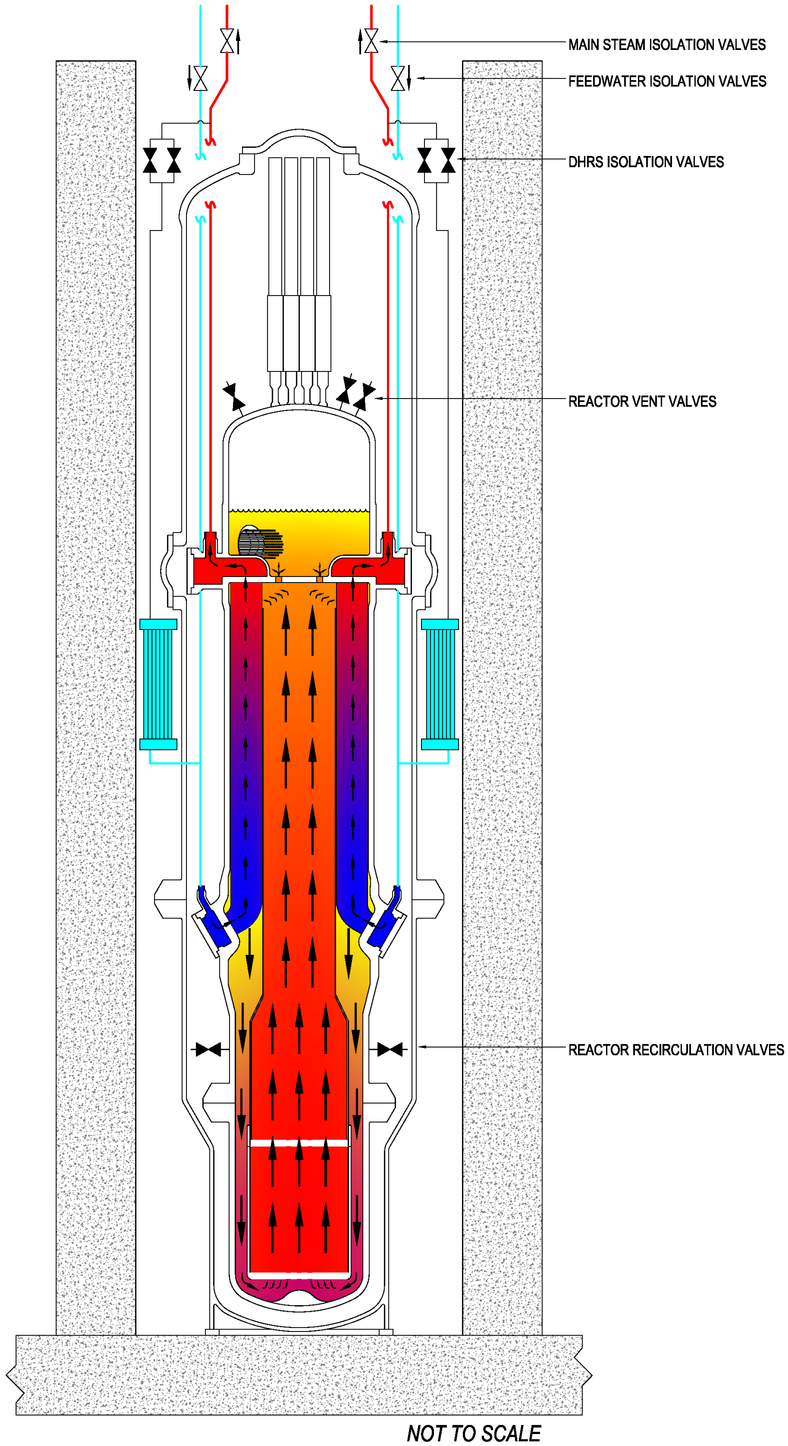


Figure 1.2-8: Decay Heat Removal System

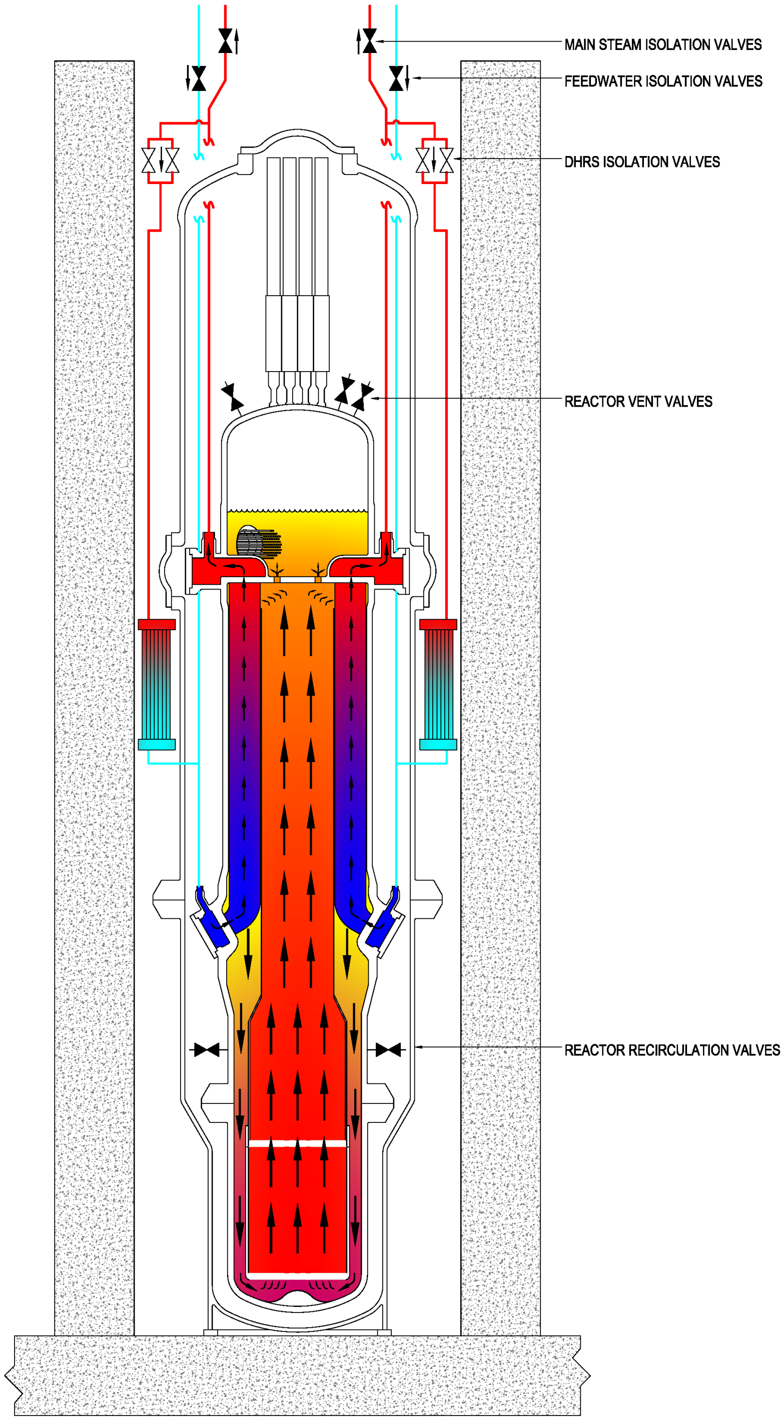


Figure 1.2-9: Emergency Core Cooling System

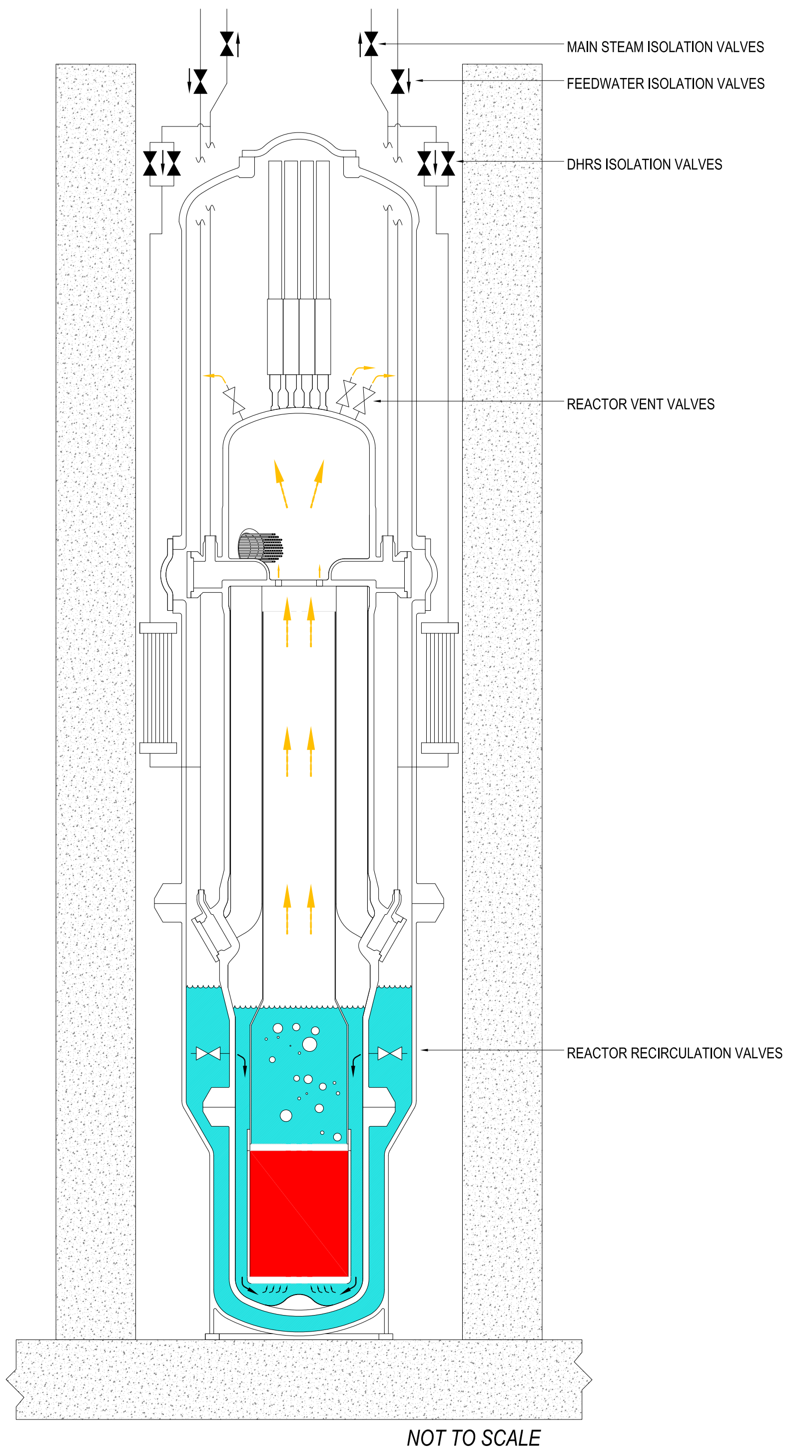


Figure 1.2-10: Reactor Building 24'-0" Elevation

{{ Withheld - See Part 9 }}

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

{{ Withheld - See Part 9 }}

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

{{ Withheld - See Part 9 }}

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

{{ Withheld - See Part 9 }}

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

{{ Withheld - See Part 9 }}

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

{{ Withheld - See Part 9 }}

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

{{ Withheld - See Part 9 }}

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

{{ Withheld - See Part 9 }}

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

{{ Withheld - See Part 9 }}

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

{{ Withheld - See Part 9 }}

Figure 1.2-20: Reactor Building South Section View

{{ Withheld - See Part 9 }}

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

{{ Withheld - See Part 9 }}

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

{{ Withheld - See Part 9 }}

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

{{ Withheld - See Part 9 }}

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

{{ Withheld - See Part 9 }}

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

{{ Withheld - See Part 9 }}

Figure 1.2-26: Control Building North Section View

{{ Withheld - See Part 9 }}

Figure 1.2-27: Control Building West Section View

{{ Withheld - See Part 9 }}

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

{{ Withheld - See Part 9 }}

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	-17x17	17x17
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- and 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 (ft2)	55,000	Approximately 18,000
Number of tubes	5,626	1,380
Reactor Coolant Pumps	4	0
Pressurizer		
Internal volume (ft3)	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

Safety System or Component	Typical PWR	NuScale
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 Not Used

1.4.2 Division of Responsibility

NuScale Power, LLC (NuScale) has the overall design responsibility for the NuScale certified design.

1.4.3 Principal Consultants and Other Participants

Fluor Corporation (Fluor) provided the balance of plant design described in the Design Certification Application.

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 that support the safety analysis for the NuScale Power Plant. The testing program described in this section was developed to provide data to support the final safety analyses.

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) for which applicable data or operational experience did not previously exist. Tests specific to the NuScale fuel design are summarized in Section 1.5.1.1 and Section 1.5.1.2; tests specific to the steam generator (SG) are summarized in Section 1.5.1.3 and Section 1.5.1.4; tests specific to the control rod assemblies are summarized in Section 1.5.1.6 and Section 1.5.1.7; and tests involving integrated system phenomena are summarized in Section 1.5.1.5.

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 NSP2 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, instrumentation was used to measure key test parameters, including: resistance temperature detectors (RTD), thermocouples, 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.

Testing for the preliminary fuel design CHF test series was performed at 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 different 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 with different axial flux shapes and fuel assembly subchannels as described in NuScale Power Critical Heat Flux Correlations topical report (Reference 1.5-1).

Tests were performed to envelope 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 for 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 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, differing axial power profiles, with and without a simulated guide tube. The testing 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 NSP2 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 develop NuScale NSP4 CHF correlation and to validate NuScale's NSP2 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. The fuel assembly simulators with different 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 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 NSP2 CHF correlation and to develop the NSP4 correlation for the NuFuel HTP2™ fuel design.

1.5.1.3 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 testing 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 Societa 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. Dynamic pressure measurements were recorded during 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, to provide benchmarking for NuScale thermal-hydraulic design and systems computer codes, and to 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.

1.5.1.4 Steam Generator Thermal-Hydraulic Performance Testing - Fluid-Heated Facility

Subsequent to the SG tests described in Section 1.5.1.3 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.

1.5.1.5 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 informed 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). NIST-1 volumes, lengths, and areas 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 and containment 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 and containment, 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.

1.5.1.6 Control Rod Drive Mechanism Proof Test

The control rod drive mechanism for the NPM contains features that are not common in conventional control rod drive mechanisms: a remote disconnect mechanism and a long control rod drive shaft. 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. Additional testing to determine misalignment limits is described in Section 1.5.1.7.

Testing was completed at the Curtiss Wright facilities in Cheswick, PA, for both remote connect 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.

1.5.1.7 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. The control rod drive shafts are aligned using the following multiple-support features:

- control rod drive mechanism nozzles in the reactor vessel head
- integrated steam plenum
- 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 control rod drive shaft creates a unique configuration of these components with no operational 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 was completed at the AREVA Technical Center in Erlangen, Germany, and was configured as an ambient pressure and temperature test. The ambient test configuration used a full-length control rod drive shaft coupled with a 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 were immersed in the water under ambient conditions with no coolant flow. During the test, the coupled CRA and control rod drive shaft assembly was dropped using multiple configurations having 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.

Test results confirmed the operability of the control rod drive shafts for a range of potential component conditions and distortions. Test results also confirmed CRA drop time and CRA impact force at end of drop.

1.5.1.8 Emergency Core Cooling System Valve Design Certification Application Demonstration Test

The NPM design utilizes emergency core cooling system (ECCS) valves. The reactor vent valves (RVVs) are located on the reactor vessel pressurizer, and the reactor recirculation valves (RRVs) are located on the reactor vessel downcomer. The RVVs and RRVs are functionally similar in design, however the RVVs are larger than the RRVs. Each of the ECCS valves are pilot operated by remote-mounted trip and reset solenoid valves located at the reactor containment boundary. An inadvertent actuation block (IAB) feature is included in the system, located inside containment on each main valve.

A test program was developed to demonstrate functional performance of the ECCS valve system design including the unique aspects of the design, such as the configuration of the ECCS valve components and the IAB feature. The purpose of this test program was to

- demonstrate the ECCS valves function at reactor operating pressures and temperatures.
- demonstrate the ECCS valves function in borated reactor coolant.

Testing activities were performed at the Curtiss-Wright Valve Group Target Rock facility in Farmingdale, New York, using high pressure and temperature valve test cells. A test article was designed to represent an RRV consisting of the main valve, trip valve, reset valve, and IAB valve components. To represent the NPM design, a pilot-operated main valve was used with an inlet connected to a source able to provide water at reactor operating pressures and temperatures, and an outlet that exhausted to atmosphere. Trip line tubing and a trip valve representative of the NPM design was connected to the main valve to simulate NPM valve actuation performance and potential for boric acid buildup in internal passages. A reset valve was used to pressurize and close the main valve prior to each test run. A representative IAB valve was used in the same functional configuration as the NPM design.

Types of tests carried out included main valve actuation, IAB functionality, and boric acid effects. Boric acid effects testing was performed using a vessel simulating the main valve control chamber in place of the main valve. Boric acid concentrations were selected to bound refueling and operating boron concentrations for the NPM design. Tests were performed through the range of reactor operating pressures and temperatures.

This test program was inspected by the NRC in accordance with Inspection Procedures IP35034.

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.

1.5.3 References

1.5-1 NuScale Power, LLC, "NuScale Power Critical Heat Flux Correlations," TR-0116-21012-P-A, Revision 1.

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

Topical Report Number	Topical Report Title	FSAR Section
NP-TR-1010-859-NP-A, Revision 5	NuScale Topical Report: Quality Assurance Program Description for the NuScale Power Plant	17
TR-0515-13952-NP-A, Revision 0	Risk Significance Determination	17, 19
TR-0815-16497-P-A, Revision 1	Safety Classification of Passive Nuclear Power Plant Electrical Systems	8, 15
TR-1015-18653-P-A, Revision 2	Design of the Highly Integrated Protection System Platform	7, 15
TR-0915-17565-P-A, Revision 4	Accident Source Term Methodology	15
TR-0116-20825-P-A, Revision 1	Applicability of AREVA Fuel Methodology for the NuScale Design	4
TR-0616-48793-P-A, Revision 1	Nuclear Analysis Codes and Methods Qualification	4
TR-0516-49417-P-A, Revision 1	Evaluation Methodology for Stability Analysis of the NuScale Power Module	4
TR-0516-49422-P-A, Revision 2	Loss-Of-Coolant Accident Evaluation Model	15
TR-0915-17564-P-A, Revision 2	Subchannel Analysis Methodology	4
TR-0516-49416-P-A, Revision 3	Non-Loss-of-Coolant Accident Analysis Methodology	15
TR-0116-21012-P-A, Revision 1	NuScale Power Critical Heat Flux Correlations	4
TR-0716-50350-P-A, Revision 1	Rod Ejection Accident Methodology	15
TR-0716-50351-P-A, Revision 1	NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces	4

Table 1.6-2: NuScale Referenced Technical Reports

Report Number	Title	FSAR Section
TR-0116-20781-P, Revision 1	Fluence Calculation Methodology and Results	4.3, 5.3
TR-0316-22048-P, Revision 3	Nuclear Steam Supply System Advanced Sensor Technical Report	7.1, 7.2
TR-0416-48929-P, Revision 1	NuScale Design of Physical Security Systems	9.5, 13.6, 14.2, 14.3
TR-0516-49084-P, Revision 3	Containment Response Analysis Methodology Technical Report	6.2
TR-0616-49121-P, Revision 3	NuScale Instrument Setpoint Methodology Technical Report	7.0, 7.2
TR-0716-50424-P, Revision 1	Combustible Gas Control	3.8, 6.2
TR-0716-50439-P, Revision 2	NuScale Comprehensive Vibration Assessment Program Analysis Technical Report	3.9, 14.2
TR-0816-49833-P, Revision 1	Fuel Storage Rack Analysis	3.7, 3.8, 9.1
TR-0816-50796-P, Revision 1	Loss of Large Areas Due to Explosions and Fires Assessment (Security Related Information)	20.2
TR-0816-50797-NP, Revision 3	Mitigation Strategies for Loss of All AC Power Event	20.1
TR-0816-51127-P, Revision 3	NuFuel-HTP2™ Fuel and Control Rod Assembly Designs	4.2
TR-0916-51299-P, Revision 3	Long-Term Cooling Methodology	5.4, 6.2, 6.3, 15.0
TR-0916-51502-P, Revision 2	NuScale Power Module Seismic Analysis	3.7, 3.12, 3B
TR-1015-18177-P, Revision 2	Pressure and Temperature Limits Methodology	5.3
TR-1016-51669-P, Revision 1	NuScale Power Module Short-Term Transient Analysis	3.9
TR-1116-51962-P, Revision 1	NuScale Containment Leakage Integrity Assurance Technical Report	6.2
TR-1116-52065-P, Revision 1	Effluent Release (GALE Replacement) Methodology and Results	11.1, 11.2, 11.3
RP-0215-10815-P, Revision 3	Concept of Operations	18.7

Material Referenced

Table 1.6-2: NuScale Referenced Technical Reports (Continued)

Report Number	Title	FSAR Section
RP-0316-17614-P, Revision 0	Human Factors Engineering Operating Experience Review Results Summary Report	18.2
RP-0316-17615-P, Revision 0	Human Factors Engineering Functional Requirements Analysis and Function Allocation Results Summary Report	18.3
RP-0316-17616-P, Revision 2	Human Factors Engineering Task Analysis Results Summary Report	18.4
RP-0316-17617-P, Revision 0	Human Factors Engineering Staffing and Qualifications Results Summary Report	18.5
RP-0316-17618-P, Revision 0	Human Factors Engineering Treatment of Important Human Actions Results Summary Report	18.6
RP-0316-17619-P, Revision 2	Human Factors Engineering Human-System Interface Design Results Summary Report	18.7
RP-0516-49116-P, Revision 1	Control Room Staffing Plan Validation Results	18.5
RP-0914-8534-P, Revision 5	Human Factors Engineering Program management Plan	18.1
RP-0914-8543-P, Revision 5	Human Factors Verification and Validation Implementation Plan	18.1
RP-0914-8544-P, Revision 4	Human Factors Engineering Design Implementation Implementation Plan	18.11
RP-1215-20253-P, Revision 3	Control Room Staffing Plan Validation Methodology	18.5
TR-0917-56119-P, Revision 1	CNV Ultimate Pressure Integrity	3.8
TR-0918-60894-P, Revision 1	NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report	3.9, 14.2
TR-0818-61384-P, Revision 2	Pipe Rupture Hazards Analysis	3.6
ES-0304-1381-P, Revision 4	Human-System Interface Style Guide	18.10
RP-1018-61289-P, Revision 1	HFE Verification and Validation Results Summary Report	18.1

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 piping and instrumentation diagrams.

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

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

Figure	Title
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
Figure 7.0-13	Plant Protection System Block Diagram
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
Figure 7.0-20	Plant Control System Internal Functions and External Interfaces
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 Alternating Current Electrical Distribution System
Figures 8.3-4a through 8.3-4z	Low Voltage Alternating Current Electrical Distribution System
Figures 8.3-5a and 8.3-5b	Backup Power Supply System
Figure 8.3-6	Highly Reliable Direct Current Power System (Common)
Figures 8.3-7a and 8.3-7b	Highly Reliable Direct Current Power System (Module Specific)
Figures 8.3-8a through 8.3-8f	Normal Direct Current Power System
Figure 11.5-2	Process and Effluent Radiation Monitoring System I&C Configuration

Table 1.7-2: System Drawings

Figure	Title
Figure 5.1-2	Reactor Coolant System Simplified Diagram
Figure 5.1-3	Reactor Coolant System Schematic Flow Diagram
Figure 5.4-9	Steam Generator Simplified Diagram
Figure 5.4-10	Decay Heat Removal System Simplified Diagram
Figure 6.2-4	Containment System Piping and Instrumentation Diagram
Figure 6.2-7	Containment Isolation Valve Actuator Hydraulic Schematic
Figure 6.2-8	Containment Isolation Valve Hydraulic Skid Schematic
Figure 6.3-1	Emergency Core Cooling System
Figure 6.3-3	Emergency Core Cooling System Valve and Actuator Schematic
Figure 6.4-1	Control Room Habitability System 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 Control System Diagram
Figure 9.2.2-1	Reactor Component Cooling Water System Diagram
Figure 9.2.3-1	Demineralized Water System Diagram
Figure 9.2.5-2	Ultimate Heat Sink Qualified Makeup Line and Instrumentation Diagram
Figure 9.2.6-1	Condensate Storage Facility
Figure 9.2.7-1	Site Cooling Water System Diagram
Figure 9.2.8-1	Chilled Water System Diagram
Figure 9.2.9-1	Utility Water System Diagram
Figure 9.3.1-1	Instrument Air and Service Air System Diagram
Figure 9.3.1-2	Nitrogen Distribution System Diagram
Figure 9.3.2-1	Containment Sampling System Diagram
Figure 9.3.3-1	Radioactive Waste Drain System Diagram
Figure 9.3.3-2	Balance-of-Plant Drain System Diagram
Figure 9.3.4-1	Chemical and Volume Control System Diagram
Figure 9.3.4-2	Boron Addition System Diagram
Figure 9.3.6-1	Containment Evacuation System Diagram
Figure 9.3.6-2	Containment Flooding and Drain System Diagram
Figure 9.4.1-1	Control Room Ventilation System Diagram
Figure 9.4.2-1	Reactor Building HVAC System Diagram
Figure 9.4.3-1	Radioactive Waste Building HVAC System Diagram
Figure 9.4.4-1	Turbine Building HVAC System 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 10.3-1	Main Steam System Piping and Instrumentation Diagram
Figure 10.4-1	Main 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 Wet Solid Waste
Figure 11.4-2b	Solid Radioactive Waste System Diagram

Table 1.7-2: System Drawings (Continued)

Figure	Title
Figure 11.5-1	Radioactive Effluent Flow Paths with Process and Effluent Radiation Monitors
Figure 11.5-3	Off-Line Radiation Monitor
Figure 11.5-4	Adjacent-to-Line Radiation Monitor
Figure 11.5-5	In-Line Radiation Monitor
Figure 11.5-6	RBVS Plant Exhaust Stack Effluent Radiation Monitor

Figure 1.7-1a: Electrical Symbols

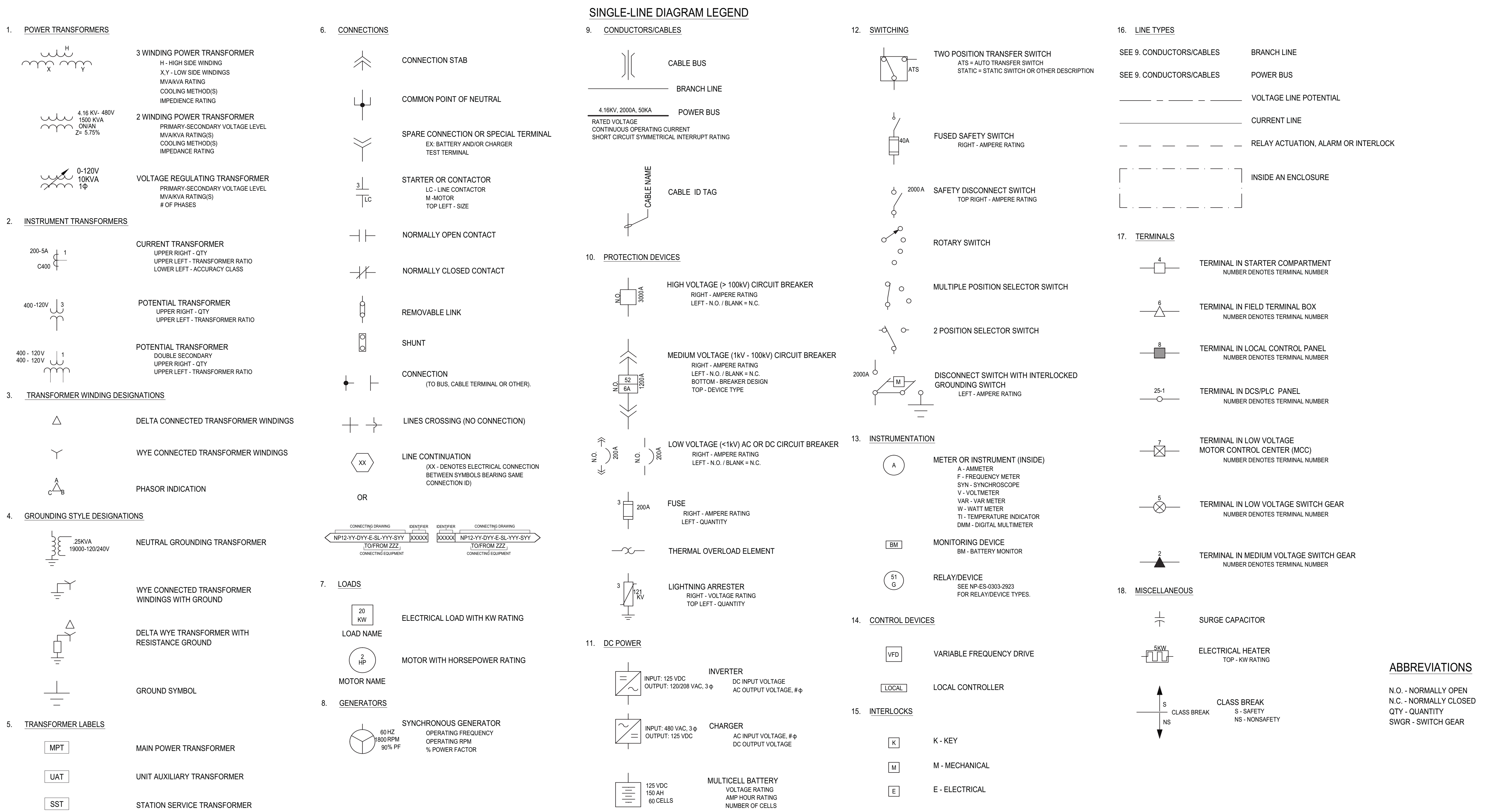


Figure 1.7-1b: Electrical Symbols

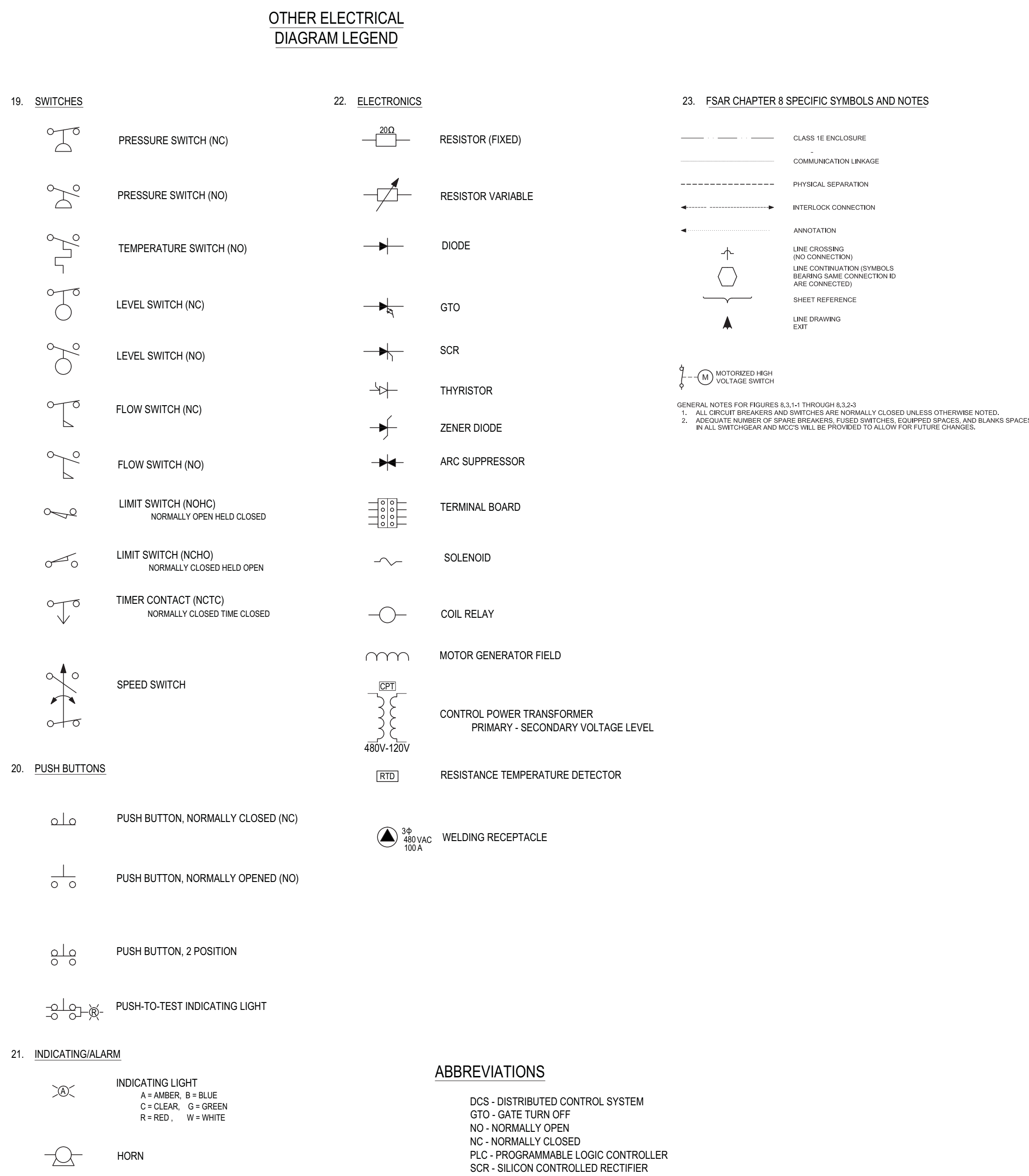


Figure 1.7-2: Instrumentation and Controls Symbol Legend

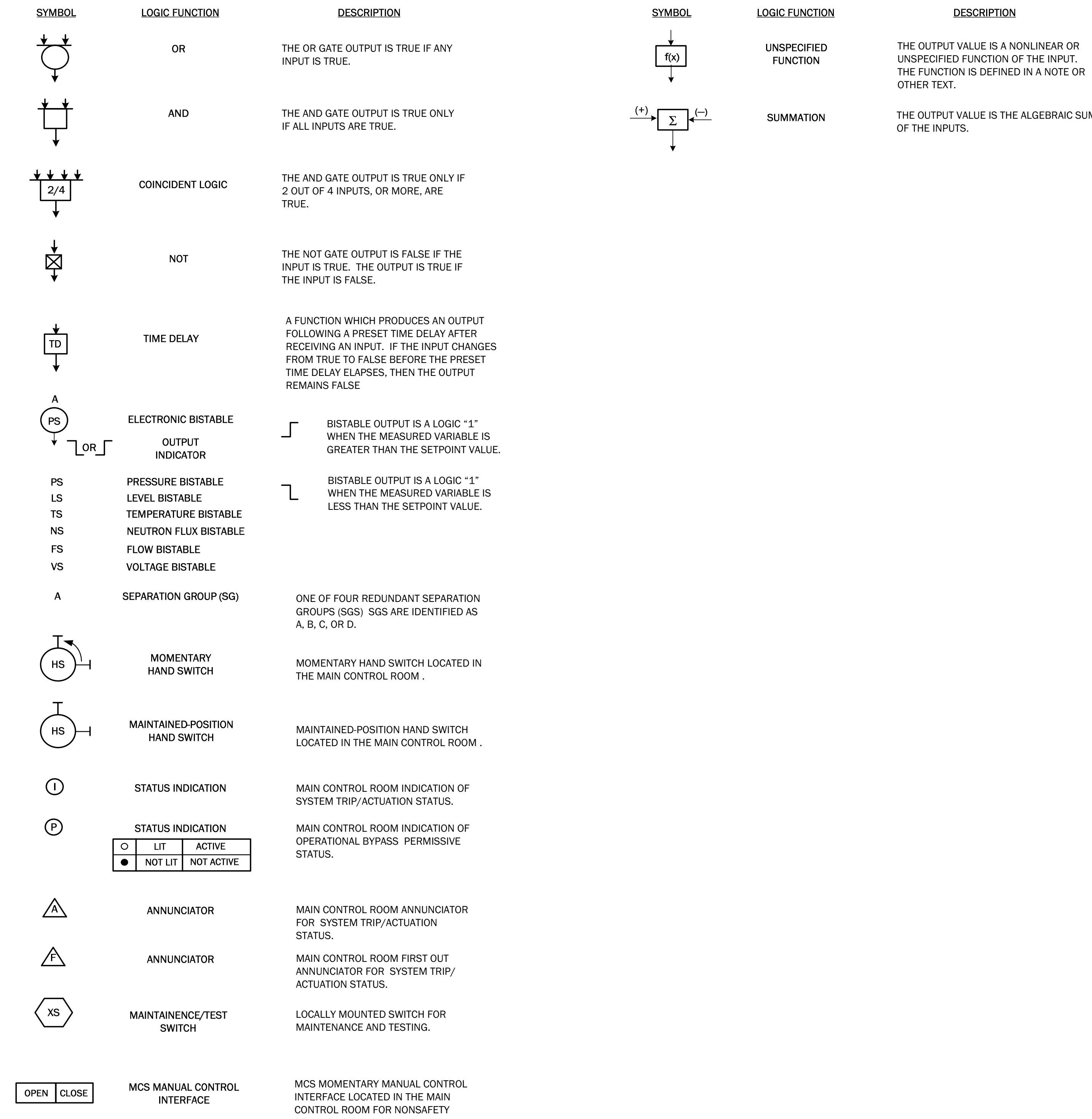


Figure 1.7-3a: Piping and Instrumentation Diagram Legends

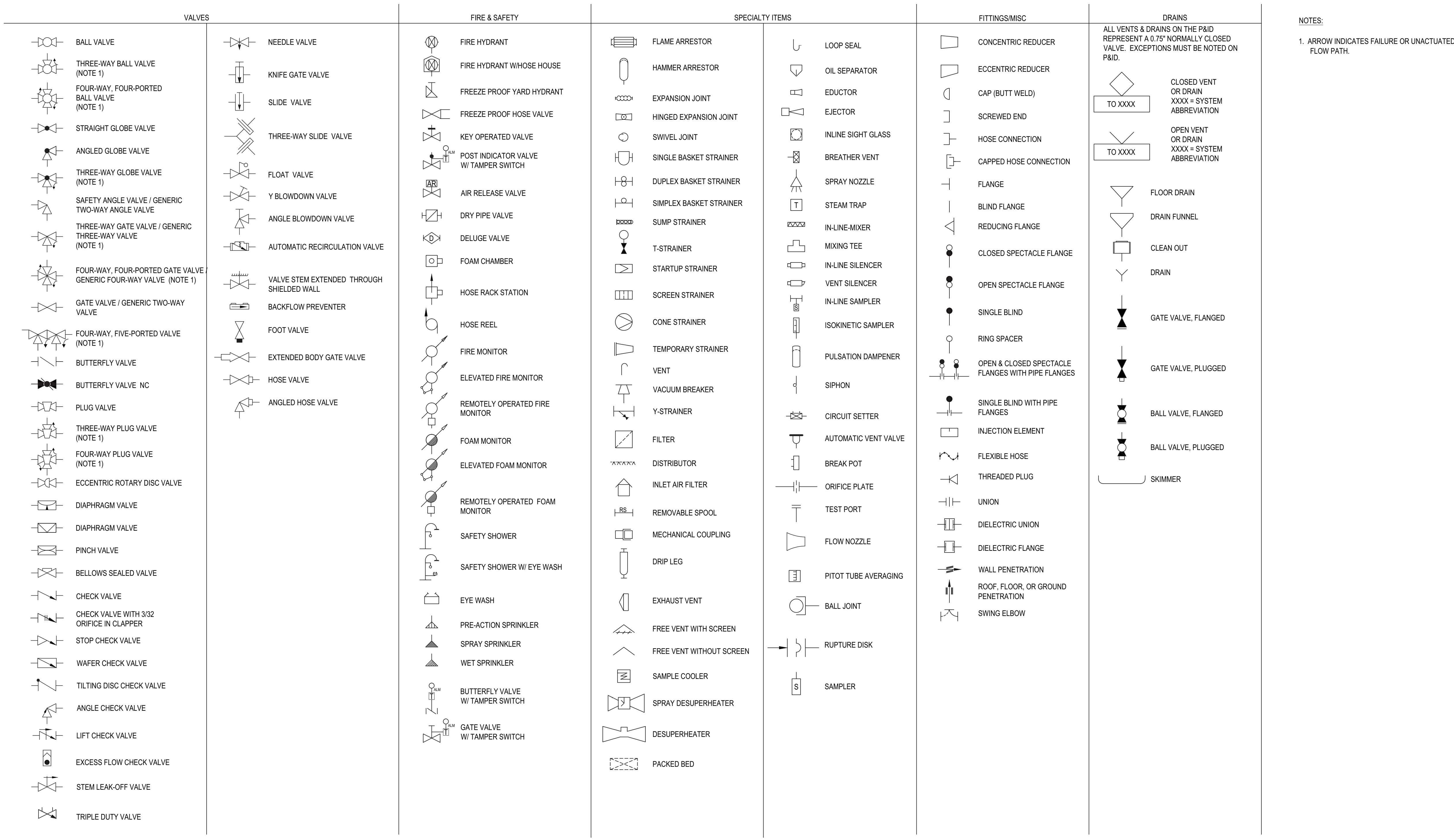


Figure 1.7-3b: Piping and Instrumentation Diagram Legends

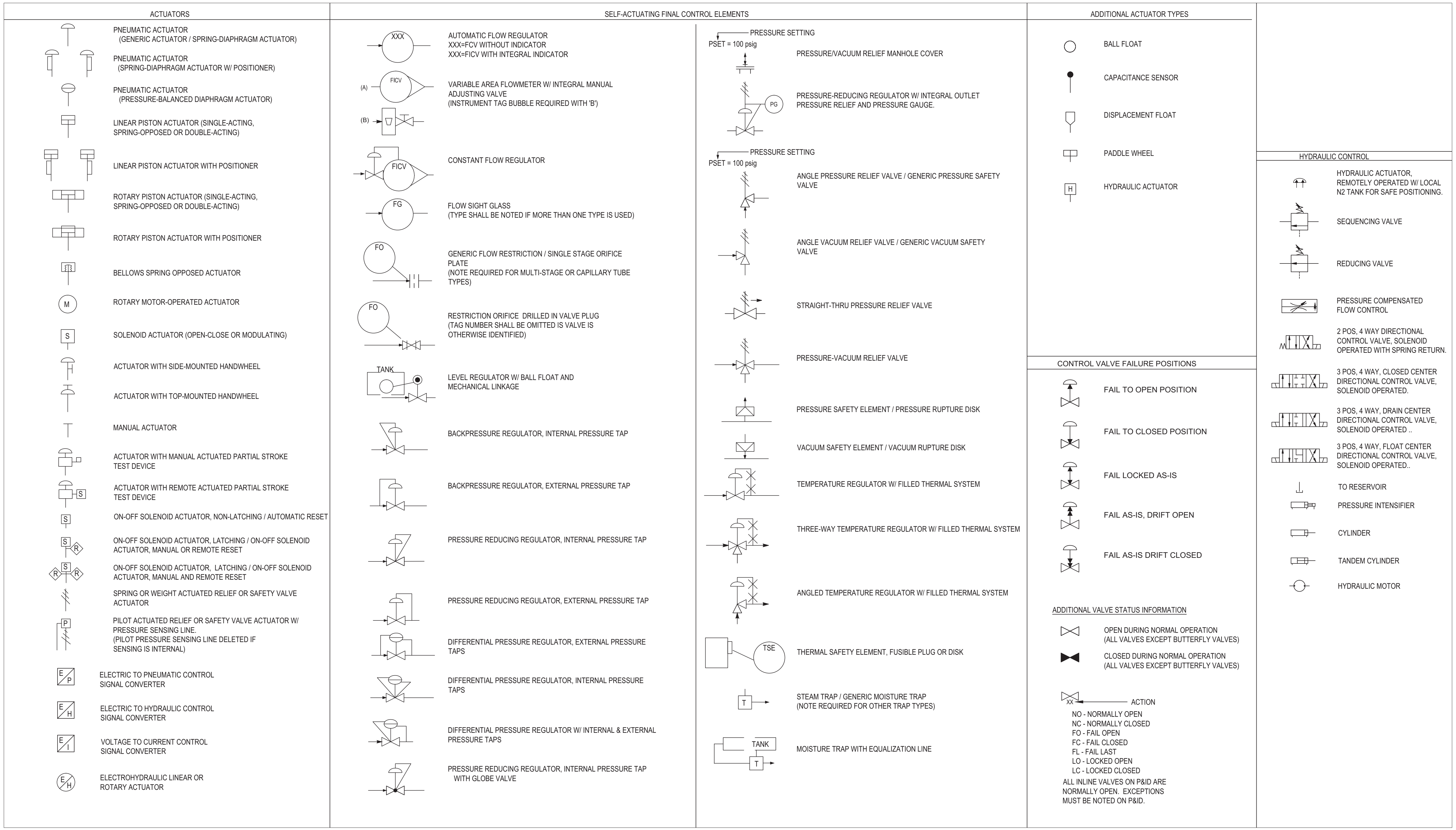


Figure 1.7-3c: Piping and Instrumentation Diagram Legends

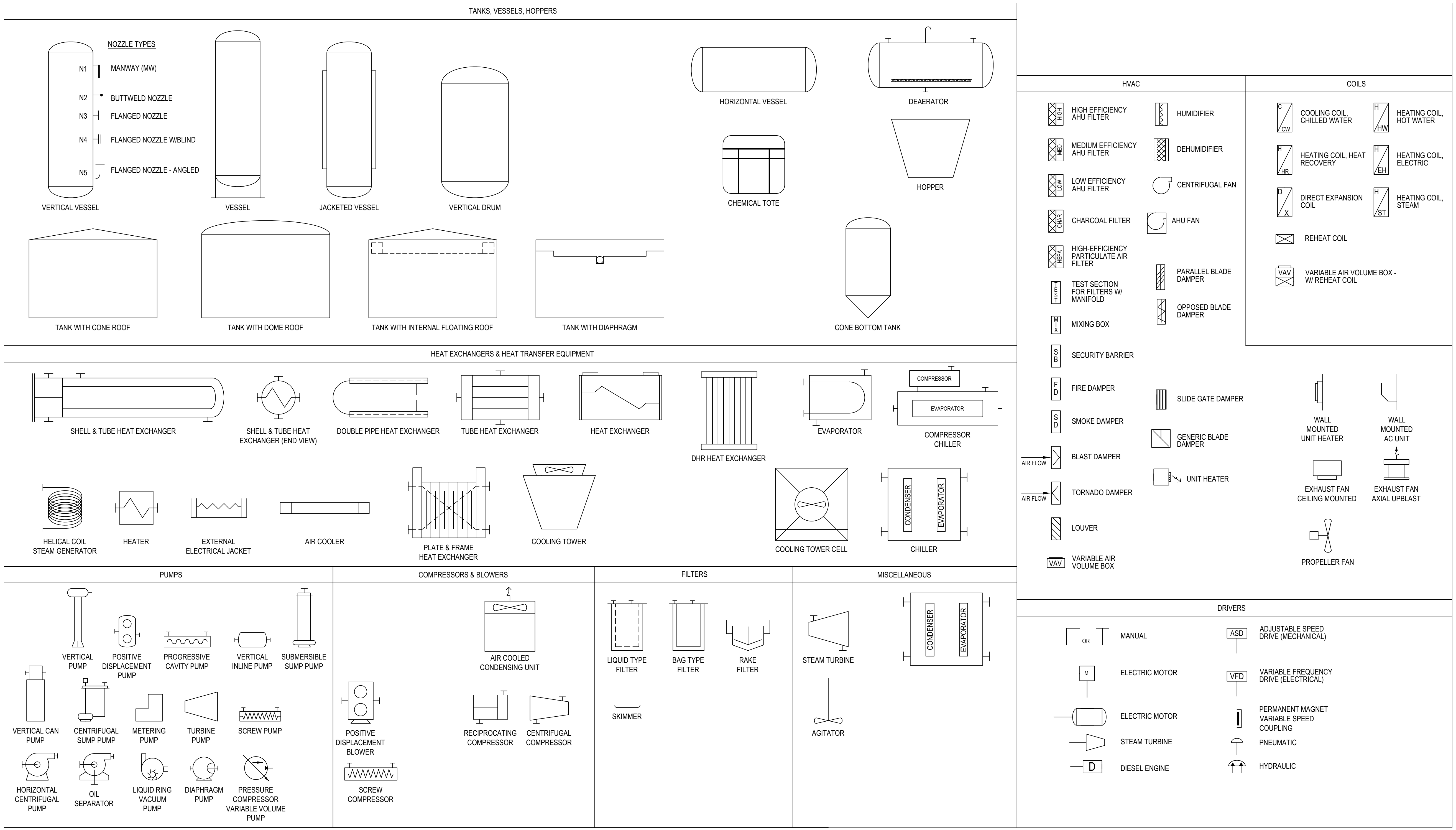
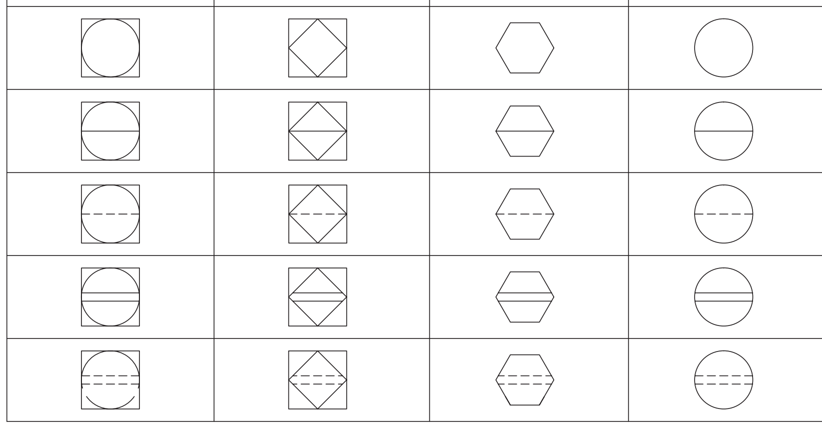
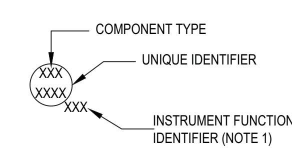


Figure 1.7-3d: Piping and Instrumentation Diagram Legends



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
				• LOCATED IN FIELD • NOT PANEL, CABINET OR CONSOLE MOUNTED • VISIBLE AT FIELD LOCATION • NORMALLY OPERATOR ACCESSIBLE
				• LOCATED IN OR ON FRONT OF CONTROL OR MAIN PANEL OR CONSOLE • VISIBLE ON FRONT OF PANEL OR ON VIDEO DISPLAY • NORMALLY OPERATOR ACCESSIBLE AT PANEL FRONT OR CONSOLE
				• LOCATED IN REAR OF CENTRAL OR MAIN PANEL • LOCATED IN CABINET BEHIND PANEL • NOT VISIBLE ON FRONT OF PANEL OR ON VIDEO DISPLAY • NOT NORMALLY OPERATOR ACCESSIBLE AT PANEL OR CONSOLE
				• LOCATED IN OR ON FRONT OF SECONDARY OR LOCAL PANEL OR CONSOLE • VISIBLE ON FRONT OF PANEL OR ON VIDEO DISPLAY • NORMALLY OPERATOR ACCESSIBLE AT PANEL FRONT OR CONSOLE
				• LOCATED IN REAR OF SECONDARY OR LOCAL PANEL • LOCATED IN FIELD CABINET • NOT VISIBLE ON FRONT OF PANEL OR ON VIDEO DISPLAY • NOT NORMALLY OPERATOR ACCESSIBLE AT PANEL OR CONSOLE

INSTRUMENT TAGGING



NOTES:

1. INSTRUMENT FUNCTION IDENTIFIER. USED ONLY WHEN THE COMPONENT TYPE REQUIRES FURTHER CLARIFICATION. SEE FUNCTION IDENTIFIERS TABLE ON FIGURE 1.7-3e.
2. FOR INSTRUMENTATION TYPE AND FUNCTION, SEE INSTRUMENTATION IDENTIFICATION LETTERS TABLE ON THIS SHEET AND FUNCTION IDENTIFIERS TABLE ON FIGURE 1.7-3e.
3. TEMPERATURE ELEMENTS ARE THERMOCOUPLES UNLESS NOTED OTHERWISE.
4. 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	PRIMARY ELEMENT	TEST POINT	WELL OR PROBE	VIEWING DEVICE, GLASS	SAFETY DEVICE	FINAL ELEMENT
FIRST LETTERS	INDICATING OR MEASURABLE VARIABLE	INDICATING	BLIND	CONTROL VALVES	INDICATING	HIGH**	LOW**	COMB**	INDICATING	BLIND							
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		FQIT	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)
KQI	(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	CALORIMETER	GC	GAS CHROMATOGRAPH	MeOH	METHYL ALCOHOL	ORP	OXIDATION REDUCTION	TDS	TOTAL DISSOLVED SOLIDS
CO	CARBON MONOXIDE	H2	HYDROGEN/HYDROGEN ANALYSIS	MOIST	MOISTURE	PHASE	PHASE	THC	TOTAL HYDROCARBON
CO2	CARBON DIOXIDE	HC	HYDROCARBON	MS	MASS SPECTROMETER	pH	HYDROGEN ION	TOC	TOTAL ORGANIC CARBON
COL	COLOR	H2O	WATER	NIR	NEAR INFRARED	REF	REFRACTOMETER	TURB	TURBIDITY
COMB	COMBUSTION	H2S	HYDROGEN SULFIDE	N2	NITROGEN	RI	REFRACTIVE INDEX	UV	ULTRAVIOLET
COND	ELECTRICAL CONDUCTIVITY	HUM	HUMIDITY	NH3	AMMONIA	SOx	OXIDES OF SULFUR	VIS	VISIBLE LIGHT
DEN	DENSITY	%RH	RELATIVE HUMIDITY	NOx	OXIDES OF NITROGEN	SP GR	SPECIFIC GRAVITY	VISC	VISCOSITY
DEWPT	DEW POINT	IR	INFRARED	O2	OXYGEN	TC	THERMAL CONDUCTIVITY		
DO	DISSOLVED OXYGEN	LC	LIQUID CHROMATOGRAPH	OP	OPACITY	TDL	TUNABLE DIODE LASER		
FLOW									
CFR	CONSTANT FLOW REGULATOR	FLT	FLOW RATE	OP-E	ECCENTRIC	PV	PITOT VENTURI	TUR	TURBINE
CONE	CONE	LAM	LAMINAR	OP-FT	FLANGE TAPS	SNR	SONAR	US	ULTRASONIC
COR	CORIOLLIS	MAG	MAGNETIC	OP-MH	MULTI-HOLE	SON	SONIC	VENT	VENTURI TUBE
DOP	DOPPLER	OP	ORIFICE PLATE	OP-P	PIPE TAPS	TAR	TARGET	VOR	VORTEX SHEDDING
DSON	DOPPLER SONIC	OP-CT	CORNER TAPS	OP-VC	VENA CONTRACTA TAPS	THER	THERMAL	WDG	WEDGE
FLN	FLOW NOZZLE	OP-CQ	CIRCLE QUADRANT	PD	POSITIVE DISPLACEMENT	TTS	TRANSIT TIME SONIC		
				PT	PITOT TUBE				
LEVEL									
CAP	CAPACITANCE	DP	DIFFERENTIAL PRESSURE	MS	MAGNETOSTRICTIVE	SON	SONIC		
d/p	DIFFERENTIAL PRESSURE	GWR	GUIDED WAVE RADAR	NUC	NUCLEAR	US	ULTRASONIC		
DI	DIELECTRIC CONSTANT	LSR	LASER	RADAR	RADAR				
DISP	DISPLACER	MAG	MAGNETIC	RES	RESISTANCE				
PRESSURE									
ABS	ABSOLUTE	MAN	MANOMETER	VAC	VACUUM				
AVG	AVERAGE	P-V	PRESSURE-VACUUM						
DRF	DRAFT	SG	STRAIN GAUGE						
TEMPERATURE									
BM	BI-METAL	RTD	RESISTANCE TEMP. DETECTOR	TCJ	THERMOCOUPLE, TYPE J	THRM	THERMISTOR		
IR	INFRARED	TC	THERMOCOUPLE	TCK	THERMOCOUPLE, TYPE K	TMP	THERMOPILE		
RAD	RADIATION	TCE	THERMOCOUPLE, TYPE E	TCT	THERMOCOUPLE, TYPE T	TRAN	TRANSISTOR		
RP	RADIATION PYROMETER								
MISCELLANEOUS									
ANNUNCIATION		BURNER, COMBUSTION		OTHER				POSITION	
ALM	ALARM	FR	FLAME ROD	CONC	CONCENTRIC	PB	PUSHBUTTON	CAP	CAPACITANCE
ANN	ANNUNCIATOR	IGN	IGNITER	HOA	HAND-OFF-AUTO	PC	PHOTOCELL	EC	EDDY CURRENT
		IR	TELEVISION	L/R	LOCAL/REMOTE	SMOKE	SMOKE	IND	INDUCTIVE
		UV	ULTRA VIOLET	MOS	MAINTENANCE OVERRIDE SWITCH	SYNC	SYNCHRONIZATION	LAS	LASER
				MULTI	MULTIVARIABLE	TDR	TIME DELAY RELAY	MAG	MAGNETIC
				O/L	OVERLOAD	TEST	TEST	MECH	MECHANICAL
				OX	OVERRIDE SWITCH	VIBR	VIBRATION	OPT	OPTICAL
				NR	NARROW RANGE	WR	WIDE RANGE	RADAR	RADAR
QUANTITY		RADIATION		SPEED		WEIGHT, FORCE			
PE	PHOTOELECTRIC	α	ALPHA RADIATION	ACC	ACCELERATION	LC	LOAD CELL		
TOG	TOGGLE	β	BETA RADIATION	EC	EDDY CURRENT	SG	STRAIN GAUGE		
		Y	GAMMA RADIATION	PROX	PROXIMITY	WS	WEIGH SCALE		
		n	NEUTRON RADIATION	VEL	VELOCITY				
		RAD	RADIATION ADSORBED DOSE						
		REM	ROENTGEN EQUIVALENT MAN						

EQUIPMENT DESCRIPTIONS (NOTE 1)

AIR COOLER
TUBE DP/DT: PSIG/°F
TUBE OP/OT : PSIG/°F
DESIGN DUTY: BTU/HR
CONFIGURATION:
MOTOR NAME PLATE: HP

COMPRESSOR
DP/DT: PSIG/°F
OP/OT: PSIG/°F
CAPACITY: FT^3/MIN
DUTY CYCLE: %
MOTOR NAME PLATE: HP

FILTERS
DP/DT: PSIG/°F
OP/OT: PSIG/°F
SIZE: ID X T-T
PARTICLE SIZE: MICRON

FAN
AIR FLOW: CFM
DUTY CYCLE: %
MOTOR NAMEPLATE: HP

HEAT EXCHANGER
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

TANK
DP/DT: PSIG/°F
OP/OT: PSIG/°F
SIZE: ID X HEIGHT: FT
CAPACITY: GAL

AHU/ ACU/ FCU
AIR FLOW: CFM
DUTY CYCLE: %
COOLING CAPACITY: BTU/ HR
HEATING CAPACITY: BTU/ HR (kW)
MOTOR NAMEPLATE: HP

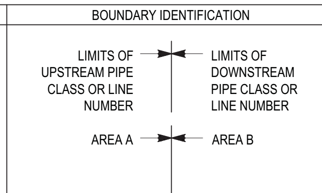
UNIT HEATER
HEATING CAPACITY: kW

PRESSURE VESSEL
DP/DT: PSIG/°F
OP/OT: PSIG/°F
SIZE: ID X T-T
CAPACITY: GAL

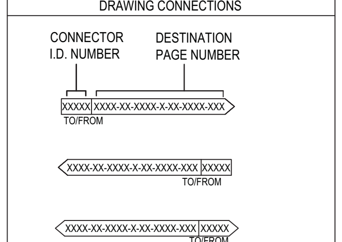
PUMP
CAPACITY: GPM
DUTY CYCLE: %
DESIGN HEAD: FT
MOTOR NAME PLATE: HP
DP/DT: PSIG/°F

CHILLER
CAPACITY: TONS
DUTY CYCLE: %
EVAP FLOW: GPM
EVAP EWT/ LWT: °F/ °F
COND FLOW: GPM
COND EWT/ LWT: °F/ °F
POWER: kW

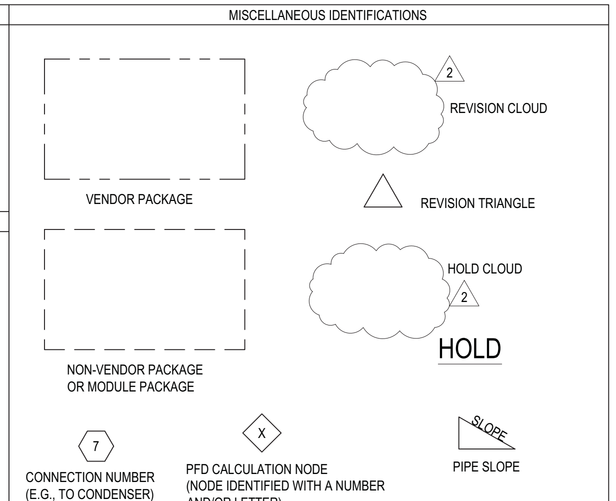
BOUNDARY IDENTIFICATION



DRAWING CONNECTIONS



MISCELLANEOUS IDENTIFICATIONS



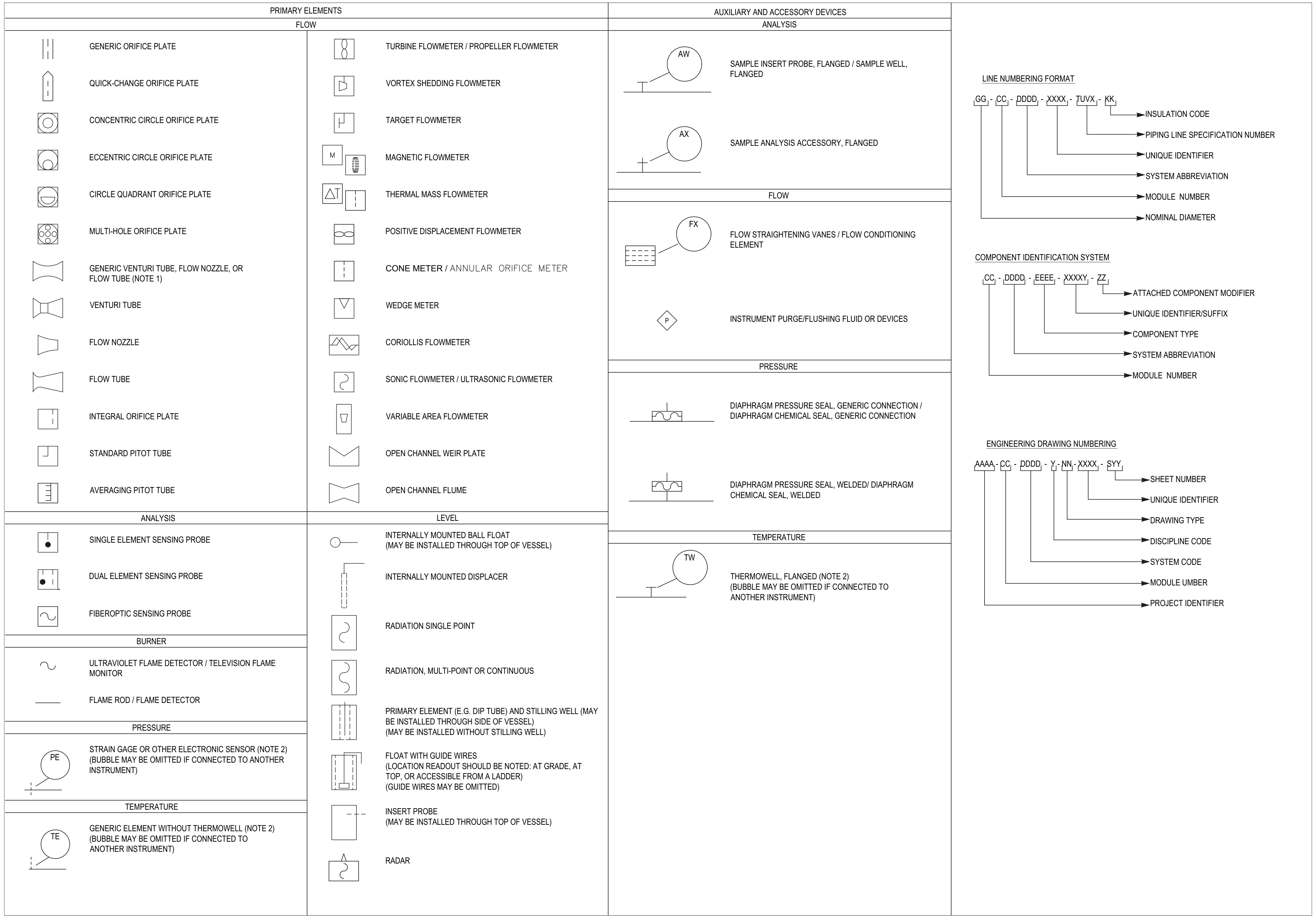
INSTRUMENTATION LINE SYMBOLS: INSTRUMENT TO PROCESS AND EQUIPMENT CONNECTIONS	
SYMBOL	DESCRIPTION
	• INSTRUMENT CONNECTION TO PROCESS AND EQUIPMENT • PROCESS IMPULSE LINES • ANALYZER SAMPLE LINES
---(ST)---	• HEAT (COOL) TRACED IMPULSE OR SAMPLE LINE FROM PROCESS • TYPE OF TRACING INDICATED BY (ET) ELECTRICAL, (ST) STEAM, (CW) CHILLED WATER, ETC
	• GENERIC INSTRUMENT CONNECTION TO PROCESS FLOW • GENERIC INSTRUMENT CONNECTION TO EQUIPMENT
	• HEAT (COOL) TRACED GENERIC INSTRUMENT IMPULSE LINE • PROCESS LINE OR EQUIPMENT MAY OR MAY NOT BE TRACED
	• HEAT(COOL) TRACED INSTRUMENT • INSTRUMENT IMPULSE LINE MAY OR MAY NOT BE TRACED

LINE SYMBOLS	
LINE TYPE/SYMBOL	DESCRIPTION
IA	• IA MAY BE REPLACED BY PA (PLANT AIR), NS (NITROGEN), OR GS (ANY GAS SUPPLY). • INDICATE SUPPLY PRESSURE AS REQUIRED, E.G. PA-70 KPA, NS-150 PSIG, ETC.
ES	• INSTRUMENT ELECTRIC POWER SUPPLY. • INDICATE VOLTAGE AND TYPE AS REQUIRED, E.G. ES-220 VAC • ES MAY BE REPLACED BY 24 VDC, 120 VAC, ETC.
HS	• INSTRUMENT HYDRAULIC POWER SUPPLY. • INDICATE PRESSURE AS REQUIRED, E.G. HS-70 PSIG.
	• ELECTRONIC OR ELECTRICAL CONTINUOUSLY VARIABLE OR BINARY SIGNAL. • FUNCTIONAL DIAGRAM BINARY SIGNAL.
	• FUNCTIONAL DIAGRAM CONTINUOUSLY VARIABLE SIGNAL. • ELECTRICAL SCHEMATIC LADDER DIAGRAM SIGNAL AND POWER RAILS.
	• FILLED THERMAL ELEMENT CAPILLARY TUBE. • FILLED SENSING LINE BETWEEN PRESSURE SEAL AND INSTRUMENT.
	• GUIDED ELECTROMAGNETIC SIGNAL. • GUIDED SONIC SIGNAL. • FIBER OPTIC SIGNAL.
	• COMMUNICATION LINK AND SYSTEM BUS, BETWEEN DEVICES AND FUNCTIONS OF A SHARED DISPLAY, SHARED CONTROL SYSTEM. • DCS, PLC, OR PC COMMUNICATION LINK AND SYSTEM BUS.
	• COMMUNICATION LINK OR BUS CONNECTING TWO OR MORE INDEPENDENT MICROPROCESSORS OR COMPUTER-BASED SYSTEMS. • DCS-TO-DCS, DCS-TO-PLC, PLC-TO-PC, DCS-TO-FIELDBUS, ETC, CONNECTIONS.
	• COMMUNICATION LINK AND SYSTEM BUS, BETWEEN DEVICES AND FUNCTIONS OF A FIELDBUS SYSTEM. • LINK FROM AND TO "INTELLIGENT" DEVICES.
	• COMMUNICATION LINK BETWEEN A DEVICE AND A REMOTE CALIBRATION ADJUSTMENT DEVICE OR SYSTEM. • LINK FROM AND TO 'SMART' DEVICES
	• MECHANICAL LINK OR CONNECTION.
	• PRIMARY LINE
	• SECONDARY LINE
---(ST)---(ST)---(ST)---(ST)---(ST)-	• STEAM TRACE LINE
---(ET)---(ET)---(ET)---(ET)---	• ELECTRICAL TRACE LINE
	• PNEUMATIC SIGNAL
	• HYDRAULIC SIGNAL.
R	• REFRIGERANT
	• OTHER SYSTEMS

NOTES:

1. SPECIFICATIONS LISTED FOR VARIOUS EQUIPMENT TYPES ARE RECOMMENDATIONS ONLY. THE REQUIRED SPECIFICATIONS ARE PER THE DISCRETION OF THE DESIGNER.

Figure 1.7-3f: Piping and Instrumentation Diagram Legends



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.

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 1.2-1 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
Plant exhaust stack	CDI	9.4.2
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	9.2.9
Site 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
Operational Support Center	COL	13.3

Table 1.8-2: Combined License Information Items

Item No.	Description of COL Information Item	Section
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 piping and instrumentation diagrams 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 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 site 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 of these potential hazards, 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, except Section 2.4.8 and Section 2.4.10.	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 structures, systems, and components.	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 Reactor Building or Seismic Category I portion of the Control Building.	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)

Item No.	Description of COL Information Item	Section
COL Item 3.4-2:	A COL applicant that references the NuScale Power plant design certification will develop the on-site program addressing the key points of flood mitigation. The key points to this program include the procedures for mitigating internal flooding events; the equipment list of structures, systems, and components subject to flood protection in each plant area; and providing assurance that the program reliably mitigates flooding to the identified structures, systems, and components.	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. Additionally, a COL applicant will provide the specified design life for waterstops, waterproofing, damp proofing, and watertight seals. If the design life is less than the operating life of the plant, the COL applicant will describe how continued protection will be ensured.	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 Reactor Building or Seismic Category I portion of the Control Building.	3.4
COL Item 3.4-7:	A COL applicant that references the NuScale Power Plant design certification will determine the extent of waterproofing and damp proofing needed to prevent groundwater and foreign material intrusion into the expansion gap between the end of the tunnel between the Reactor Building and the Control Building, and the corresponding Reactor Building connecting walls.	3.4
COL Item 3.5-1:	A COL applicant that references the NuScale Power Plant design certification will demonstrate that the site-specific turbine missile parameters are bounded by the design certification analysis, or provide a missile analysis using the site-specific turbine generator parameters to demonstrate that barriers adequately protect essential structures, systems, and components from turbine missiles. Parameters to verify are limiting turbine missile spectrum (rotor and blade material properties); turbine rotor design, geometry and number of blades; final design of the reactor building exterior wall; final design of the control building exterior wall and grade-level slab; and location of the turbines with respect to the reactor building and control building.	3.5
COL Item 3.5-2:	A COL applicant that references the NuScale Power Plant design certification will address the effect of turbine missiles from nearby or co-located facilities.	3.5
COL Item 3.5-3:	A COL applicant that references the NuScale Power Plant design certification will confirm that automobile missiles cannot be generated within a 0.5-mile radius of safety-related structures, systems, and components and risk-significant structures, systems, and components requiring missile protection that would lead to impact higher than 30 feet above plant grade. Additionally, if automobile missiles impact at higher than 30 feet above plant grade, the COL applicant will evaluate and show that the missiles will not compromise safety-related and risk-significant structures, systems, and components.	3.5
COL Item 3.5-4:	A COL applicant that references the NuScale Power Plant design certification will evaluate site-specific hazards for external events that may produce more energetic missiles than the design basis missiles defined in Tier 2, Section 3.5.1.4.	3.5
COL Item 3.6-1:	A COL applicant that references the NuScale Power Plant design certification will complete the routing of piping systems outside of the containment vessel and the area under the bioshield, identify the location of high- and moderate-energy lines, and update Table 3.6-1 as necessary. This activity includes the performance of associated final piping stress analyses, design and qualification of associated piping supports, evaluation of subcompartment pressurization effects (if applicable), and completion of the Balance of Plant Pipe Rupture Hazards Analysis, including the design and evaluation of pipe whip/jet impingement mitigation devices as required. This includes an evaluation and disposition of multi-module impacts in common pipe galleries.	3.6

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

Item No.	Description of COL Information Item	Section
COL Item 3.6-2:	A COL applicant that references the NuScale Power Plant design certification will verify that the pipe rupture hazards analysis (including dynamic and environmental effects) of the high- and moderate-energy lines outside the containment vessel (under the bioshield) is applicable. If changes are required, the COL applicant will update the pipe rupture hazards analysis, design additional protection features as necessary, and update Table 3.6-2.	3.6
COL Item 3.6-3:	A COL applicant that references the NuScale Power Plant design certification will perform the pipe rupture hazards analysis (including dynamic and environmental effects) of the high- and moderate-energy lines outside the reactor pool bay in the Reactor Building (RXB) and update Table 3.6-2 as appropriate. This includes an evaluation and disposition of multi-module impacts in common pipe galleries, and evaluations regarding subcompartment pressurization. The COL applicant will show that the analysis of RXB piping bounds the possible effects of ruptures for the routings of lines outside of the RXB or perform the pipe rupture hazards analysis of the high- and moderate-energy lines outside the buildings.	3.6
COL Item 3.6-4:	Not used.	3.6
COL Item 3.7-1:	A COL applicant that references the NuScale Power Plant design certification will describe the site-specific structures, systems, and components.	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/V2 (A, V, D, are peak ground acceleration, 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 design certification will: • develop a site-specific strain compatible soil profile. • confirm that the criterion for the minimum required response spectrum has been satisfied. • determine whether the seismic site characteristics fall within the seismic design parameters such as soil layering assumptions used in the certified design, range of soil parameters, shear wave velocity values, and minimum soil bearing capacity.	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 will not collapse and adversely affect the Reactor Building or Seismic Category I portion of the Control Building.	3.7
COL Item 3.7-5:	A COL applicant that references the NuScale Power Plant design certification will perform a soil-structure interaction analysis of the Reactor Building and the Control Building using the NuScale SASSI2010 models for those structures. The COL applicant will confirm that the site-specific seismic demands of the standard design for critical structures, systems, and components in Appendix 3B are bounded by the corresponding design certified seismic demands and, if not, the standard design for critical structures, systems, and components will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demands. Seismic demands investigated shall include forces, moments, deformations, in-structure response spectra, and seismic stability of the structures.	3.7
COL Item 3.7-6:	A COL applicant that references the NuScale Power Plant design certification will perform a structure-soil-structure interaction analysis that includes the Reactor Building, Control Building, Radioactive Waste Building and both Turbine Generator Buildings. The COL applicant will confirm that the site-specific seismic demands of the standard design structures, systems, and components are bounded by the corresponding design certified seismic demands and, if not, the standard design structures, systems, and components will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demands.	3.7
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 Regulatory Guide 1.12 "Nuclear Power Plant Instrumentation for Earthquakes," Rev. 2 (or later) and Regulatory Guide 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

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

Item No.	Description of COL Information Item	Section
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. In addition, a COL applicant that references the NuScale Power Plant design certification will prepare site-specific procedures for activities following an earthquake. These procedures and the data from the seismic instrumentation system will provide sufficient information to determine if the level of earthquake ground motion requiring shutdown has been exceeded. An activity of the procedures will be to address measurement of the post-seismic event gaps between the fuel racks and the pool walls and between the individual fuel racks and to take appropriate corrective action if needed (such as repositioning the racks or assuring that the as-found condition of the racks is acceptable based on the assumptions of the racks' design basis analysis). Acceptable guidance for procedure development is contained in Regulatory Guide 1.166 "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-earthquake Actions," Rev. 0 (or later) and 1.167, "Restart of a Nuclear Power Plant Shut Down by a Seismic Event," Rev. 0 (or later).	3.7
COL Item 3.7-9:	A COL applicant that references the NuScale Power Plant design certification will include an analysis of performance-based response spectra established at the surface and intermediate depth(s) that take into account the complexities of the subsurface layer profiles of the site and provide a technical justification for the adequacy of vertical to horizontal (V/H) spectral ratios used in establishing the site-specific foundation input response spectra and performance-based response spectra for the vertical direction.	3.7
COL Item 3.7-10:	A COL applicant that references the NuScale Power Plant design certification will perform a site-specific configuration analysis that includes the Reactor Building with applicable configuration layout of the desired NuScale Power Modules. The COL applicant will confirm the following are bounded by the corresponding design certified seismic demands: 1) The in-structure response spectra of the standard design at the foundation and roof. See FSAR Figure 3.7.2-107 and Figure 3.7.2-108 for foundation in-structure response spectra and Figure 3.7.2-113 for roof in-structure response spectra. 2) The maximum forces in the NuScale Power Module lug restraints and skirts. See Table 3B-28. 3) The site-specific in-structure response spectra for the NuScale Power Module at the skirt support will be shown to be bounded by the in-structure response spectra in Figure 3.7.2-156 and Figure 3.7.2-157. The site-specific in-structure response spectra for the NuScale Power Module at the lug restraints will be shown to be bounded by the in-structure response spectra in Figure 3.7.2-158 through Figure 3.7.2-163. 4) The maximum forces and moments in the west wing wall and pool wall. See Table 3B-23a and Table 3B-23b. 5) Not used. 6) The site-specific in-structure response spectra shown immediately below will be shown to be bounded by their corresponding certified in-structure response spectra: • Reactor Building north exterior wall at EL 75′-0″: bounded by in-structure response spectra in Figure 3.7.2-110 • Reactor Building west exterior wall at EL 126′-0″: bounded by in-structure response spectra in Figure 3.7.2-112 • Reactor Building crane wheels at EL 145′-6″: bounded by in-structure response spectra in Figure 3.7.2-114 • Control Building east wall at EL 76′-6″: bounded by in-structure response spectra in Figure 3.7.2-119a and Figure 3.7.2-119b • Control Building south wall at EL 120′-0″: bounded by in-structure response spectra in Figure 3.7.2-121a and Figure 3.7.2-121b If not, the standard design will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demands.	3.7
COL Item 3.7-11:	A COL applicant that references the NuScale Power Plant design certification will perform a site-specific analysis that assesses the effects of soil separation. The COL applicant will confirm that the in-structure response spectra in the soil separation cases are bounded by the in-structure response spectra shown in FSAR Figure 3.7.2-107 through Figure 3.7.2-122.	3.7

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

Item No.	Description of COL Information Item	Section
COL Item 3.7-12:	A COL applicant that references the NuScale Power Plant design certification will perform an analysis that uses site-specific soil and time histories to confirm the adequacy of the fluid-structure interaction correction factor.	3.7
COL Item 3.7-13:	A COL applicant that references the NuScale Power Plant design certification will perform a site-specific analysis that assesses the effects of non-vertically propagating seismic waves on the free-field ground motions and seismic responses of Seismic Category I structures, systems, and components.	3.7
COL Item 3.7-14:	A COL applicant that references the NuScale Power Plant design certification will demonstrate that the site-specific seismic demand is bounded by the FSAR capacity for an empty dry dock condition.	3.7
COL Item 3.7-15:	A COL applicant that references the NuScale Power Plant design certification will determine the appropriate site-specific number of interaction planes for soil structure interaction.	3.7
COL Item 3.7-16:	A COL applicant that references the NuScale Power Plant design certification will determine the means and methods of lifting the bioshield. A COL applicant will demonstrate that bioshield components and connections can withstand the bioshield loads and appropriate load factors.	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 Regulatory Guide 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 Reactor Building and Control Building are acceptable for use at the designated site.	3.8
COL Item 3.8-3:	A COL applicant that references the NuScale Power Plant design certification will identify local stiff and soft spots in the foundation soil and address these in the design, as necessary.	3.8
COL Item 3.8-4:	A COL applicant that references the NuScale Power Plant design certification will evaluate and document construction aid elements such as steel beams, Q-decking, formwork, lugs, and other items that are left in place after construction, but that were not part of the certified design, to verify the construction aid elements do not have an appreciable adverse effect on overall mass, stiffness, and seismic demands of the certified building structure. The COL applicant will confirm that these left-in-place construction aid elements will not have adverse effects on safety-related structures, systems, and components per Section 3.7.2.	3.8
COL Item 3.8-5:	A COL applicant that references the NuScale Power Plant design certification will verify that the reactor flange tool (RFT) and embed plates are evaluated using site-specific seismic analysis, and generate seismic loads to the reactor pressure vessel and fuel assemblies that are bounded by the certified design. The design of the structural members will be confirmed by assessing demand-to-capacity ratios for the load combinations in Table 3.8.4-23. The design of the embed plates will be confirmed by assessing demand-to-capacity ratios for the load combinations in Table 3.8.4-1 and Table 3.8.4-2, and applicable design codes in Table 3.8.4-12. In addition, the core plate in-structure response spectra for the RFT location shown in Figure B-34 through Figure B-39 of TR-0916-51502 (NuScale Power Module Seismic Analysis) shall be confirmed against the site specific spectra. If either the demands on the structural members or the embed plates exceed their capacity, or core plate motions do not maintain justifiable margin to limits for the fuel assembly, the COL applicant will address and augment the design per the criteria specified in FSAR Section 3.8.4, and the fuel assembly-imposed load limitations.	3.8
COL Item 3.8-6:	A COL applicant that references the NuScale Power Plant design certification will verify that the construction loads applied to the pool liner plate and its support structure do not exceed 600 psf per American Concrete Institute (ACI)-347, Guide to Formwork for Concrete.	3.8
COL Item 3.9-1:	A COL applicant that references the NuScale Power Plant design certification will provide the applicable test procedures before the start of testing and will submit the test and inspection results from the comprehensive vibration assessment program for the NuScale Power Module, in accordance with Regulatory Guide 1.20.	3.9

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

Item No.	Description of COL Information Item	Section
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 American Society of Mechanical Engineers (ASME) service level stresses, 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 American Society of Mechanical Engineers Operation and Maintenance 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, implementation milestones, and the applicable American Society of Mechanical Engineers (ASME) Operation and Maintenance (OM) Code (and ASME OM Code Cases) for the preservice and inservice testing programs. These programs are to be consistent with the requirements in the latest edition and addenda of the OM Code incorporated by reference in 10 CFR 50.55a in accordance with the time period specified in 10 CFR 50.55a before the scheduled initial fuel load (or the optional ASME Code Cases listed in Regulatory Guide 1.192 incorporated by reference in 10 CFR 50.55a).	3.9
COL Item 3.9-7:	Not used.	3.9
COL Item 3.9-8:	A COL applicant that references the NuScale Power Plant design certification will develop specific test procedures to allow detection and monitoring of power-operated valve assembly performance sufficient to satisfy periodic verification design basis capability requirements.	3.9
COL Item 3.9-9:	A COL applicant that references the NuScale Power Plant design certification will develop specific test procedures to allow detection and monitoring of emergency core cooling system valve assembly performance sufficient to satisfy periodic verification of design basis capability requirements.	3.9
COL Item 3.9-10:	A COL applicant that references the NuScale Power Plant design certification will verify that evaluations are performed during the detailed design of the main steam lines, using acoustic resonance screening criteria and additional calculations as necessary (e.g., Strouhal number) to determine if there is a concern. The methodology contained in "NuScale Comprehensive Vibration Assessment Program Technical Report," TR-0716-50439 is acceptable for this purpose. The COL applicant will update Section 3.9.2.1.1.3 to describe the results of this evaluation.	3.9
COL Item 3.9-11:	A COL applicant that references the NuScale Power Plant design certification will implement a control rod drive system Operability Assurance Program that meets the requirements described in Section 3.9.4.4 and provide a summary of the testing program and results.	3.9
COL Item 3.9-12:	A COL applicant that references the NuScale Power Plant design certification will perform a site-specific seismic analysis in accordance with Section 3.7.2.16. In addition to the requirements of Section 3.7, for sites where the high frequency portion of the site-specific spectrum is not bounded by the control rod drive system, the standard design of NuScale Power Module components will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demand.	3.9

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

Item No.	Description of COL Information Item	Section
COL Item 3.9-13:	A COL applicant that references the NuScale Power Plant design certification will complete an assessment of piping systems inside the reactor building to determine the portions of piping to be tested for vibration and thermal expansion. The piping systems within the scope of this testing include American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III, Class 1, 2, and 3 piping systems, other high-energy piping systems inside Seismic Category I structures or those whose failure would reduce the functioning of any Seismic Category I plant feature to an unacceptable level, and Seismic Category I portions of moderate-energy piping systems located outside of containment. The COL applicant may select the portions of piping in the NuScale design for which vibration testing is performed while considering the piping system design and analysis, including the vibration screening and analysis results and scope of testing as identified by the Comprehensive Vibration Assessment Program.	3.9
COL Item 3.9-14:	A COL applicant that references the NuScale Power Plant design certification will develop an evaluation methodology for the analysis of secondary-side instabilities in the steam generator design. This methodology will address the identification of potential density wave oscillations in the steam generator tubes, and qualification of the applicable portions of the reactor coolant system integral reactor pressure vessel and steam generator given the occurrence of density wave oscillations, including the effects of reverse fluid flows within the tubes.	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
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 3.11-3:	A COL applicant that references the NuScale Power Plant design certification will implement an equipment qualification operational program that incorporates the aspects in Section 3.11-7 specific to the environmental qualification of mechanical and electrical equipment. This program will include an update to Table 3.11-1 to include commodities that support equipment listed in Table 3.11-1.	3.11
COL Item 3.11-4:	A COL applicant that references the NuScale Power Plant design certification will ensure the environmental qualification program cited in COL Item 3.11-1 includes a description of how equipment located in harsh conditions will be monitored and managed throughout plant life. This description will include methodology to ensure equipment located in harsh environments will remain qualified if the measured dose is higher than the calculated dose.	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 the 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 American Society of Mechanical Engineers (ASME) Class 1, 2, and 3 threaded fasteners. The program will identify the applicable edition and addenda of ASME Boiler and Pressure Vessel Code Section XI and ensure compliance with 10 CFR 50.55a.	3.13

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

Item No.	Description of COL Information Item	Section
COL Item 4.2-1	A COL applicant that references the NuScale Power Plant design certification and wishes to utilize non-baseload operations will provide justification for the fuel performance codes and methods corresponding to the desired operation.	4.2
COL Item 5.2-1:	Not used.	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, including low temperature overpressure protection features.	5.2
COL Item 5.2-3:	Not used.	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 latest version of 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, monitoring of the containment atmosphere for evidence of reactor coolant system leakage, 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
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, and trending 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 trending 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 reactor pressure vessel during construction in accordance with Regulatory Guide 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 5.3-3	A COL applicant that references the NuScale Power Plant design certification will describe their reactor vessel material surveillance program consistent with NUREG 0800, Section 5.3.1.	5.3

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

Item No.	Description of COL Information Item	Section
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 the latest revision of Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," and applicable Electric Power Research Institute steam generator guidelines at the time of the COL application. The elements of the program will include: assessment of degradation, tube inspection requirements, tube integrity assessment, tube plugging, primary-to-secondary leakage monitoring, shell side integrity assessment, 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 that 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 6.2-2:	A COL applicant that references the NuScale Power Plant design certification will verify that the final design of the containment vessel meets the design basis requirement to maintain flange contact pressure at accident temperature, concurrent with peak accident pressure.	6.2
COL Item 6.2-3:	A COL applicant that references the NuScale Power Plant design certification will perform an analysis that, in consideration of the as-built containment internal free volume, demonstrates that containment design pressure and temperature bounds containment peak accident pressure and temperature. The evaluation value for containment internal free volume must include margin to address the complex shapes of internal structures and components and manufacturing processes.	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: • Foreign material exclusion controls to limit the introduction of foreign material and debris sources into containment. • Maintenance activity controls, including temporary changes, that confirm the emergency core cooling system function is not reduced by changes to analytical inputs or assumptions or other activities that could introduce debris or potential debris sources into containment. • Controls that limit the introduction of coating materials into containment. • An inspection program to confirm containment vessel cleanliness prior to closing for normal power operation.	6.3
COL Item 6.4-1:	A COL applicant that references the NuScale Power Plant design certification will comply with Regulatory Guide 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:	Not used.	6.4
COL Item 6.4-3:	Not used.	6.4
COL Item 6.4-4:	Not used.	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 control room habitability system and control room envelope integrity testing as specified in Table 6.4-4.	6.4
COL Item 6.6-1:	A COL applicant that references the NuScale Power Plant design certification will implement an inservice 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 preservice inspection and inservice inspection program plans in accordance with Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), and will establish the implementation milestones for the program. The COL applicant will identify the applicable edition of the ASME BPVC used 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 inservice inspection program for multiple NuScale Power Modules, including any alternative to the code that may be necessary to implement such an inservice inspection program.	6.6

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

Item No.	Description of COL Information Item	Section
COL Item 7.0-1:	A COL applicant that references the NuScale Power Plant design certification is responsible for demonstrating the stability of the NuScale Power Module during normal and power maneuvering operations for closed-loop module control system subsystems that use reactor power as a control input.	7.0
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 Institute of Electrical and Electronics Engineers (IEEE) Std 1074-2006 and IEEE Std 1012-2004.	7.2
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 Institute of Electrical and Electronics Engineers (IEEE) Std 1074-2006 and IEEE Std 1012-2004.	7.2
COL Item 7.2-3:	The NuScale Digital instrumentation and controls (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 Institute of Electrical and Electronics Engineers (IEEE) Std 1074-2006 and IEEE Std 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:	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 General Design Criteria (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 4. The testing description will include the details of initial testing associated with degraded offsite power conditions.	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 an 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

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

Item No.	Description of COL Information Item	Section
COL Item 9.1-5:	The COL applicant that references the NuScale Power Plant design certification will describe the process for handling and receipt of critical loads including NuScale Power Modules.	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 and handling equipment including procedures and programs for safe handling.	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 • operating and maintenance procedures • inspection and test plans • personnel qualifications and operator training • detailed description of the safe load paths for movement of heavy loads	9.1
COL Item 9.1-8:	A COL applicant that references the NuScale Power Plant design certification will provide a structural evaluation of the spent fuel storage racks, and fuel assemblies located in the racks, and confirm the thermal-hydraulic, criticality, and material analysis aspects of the design remain valid. This evaluation is dependent on the vendor-specific spent fuel storage rack design.	9.1
COL Item 9.1-9:	A COL applicant that references the NuScale Power Plant design certification will provide a neutron absorber material qualification report which demonstrates that the neutron absorber material can meet the neutron attenuation and environmental compatibility design functions described in Technical Report TR-0816-49833. The COL applicant will establish procedures to evaluate the neutron attenuation uncertainty associated with the material lot variability and will establish procedures to inspect the as-manufactured material for contamination and manufacturing defects.	9.1
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 for systems outside containment that contain (or might contain) accident source term radioactive materials following an accident (including systems and components used in post-accident hydrogen and oxygen monitoring of the containment atmosphere). The leakage control program will include an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems to as low as practical.	9.3
COL Item 9.3-2:	Not used.	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 heating ventilation and air conditioning 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 heating ventilation and air conditioning system.	9.4

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

Item No.	Description of COL Information Item	Section
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 Turbine Building heating ventilation and air conditioning system.	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 10.3-1:	A COL applicant that references the NuScale Power Plant design certification will provide a site-specific chemistry control program based on the latest revision of the Electric Power Research Institute Pressurized Water Reactor Secondary Water Chemistry Guidelines and Nuclear Energy Institute (NEI) 97-06 at the time of the COL application.	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 the steam and power conversion systems based on Generic Letter 89-08 and the latest revision of the Electric Power Research Institute NSAC-202L at the time of the COL application.	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 type of fuel supply for the auxiliary boilers.	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
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 American National Standards Institute / American Nuclear Society (ANSI/ANS)-40.37 (Reference 11.2-1), 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 of the consequences of an accidental release of radioactive liquid from the pool surge control system storage tank 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

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

Item No.	Description of COL Information Item	Section
COL Item 11.4-1:	A COL applicant that references the NuScale Power Plant design certification will describe mobile equipment used and connected to plant systems in accordance with American National Standards Institute / American Nuclear Society (ANSI/ANS) 40.37, Regulatory Guide 1.143, 10 CFR 20.1406, NRC 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 American National Standards Institute (ANSI) N13.1-2011, ANSI N42.18-2004, and Regulatory Guides 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 program to maintain exposures to ionizing radiation as far below the dose limits as practical, as low as reasonably achievable (ALARA).	12.1
COL Item 12.2-1:	A COL applicant that references the NuScale Power Plant design certification will describe additional site-specific contained radiation sources that exceed 100 millicuries (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
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 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

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

Item No.	Description of COL Information Item	Section
COL Item 12.3-8:	A COL applicant that references the NuScale Power Plant design certification will describe the radiation shielding design measures used to compensate for major shield wall penetrations in accordance with FSAR Section 12.1.2.3.2 "Minimizing Radiation Levels in Plant Access Areas and Vicinity of Equipment," Section 12.3.1.2.3 "Penetrations," and Section 12.3.2.2 "Design Considerations." Penetration compensatory measures will account for the protection of equipment, and exposures to workers and the public.	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, including a description of the qualification requirements for (1) each identified position or class of positions that provide technical support to the onsite operating organization, and (2) individuals holding management and supervisory positions in organizational units providing technical support to the onsite operating 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 and senior reactor operators.	13.2
COL Item 13.2-2:	A COL applicant that references the NuScale Power Plant design certification will provide a description and schedule of the non-licensed plant staff training programs including initial training, periodic retraining, and qualification requirements.	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 an emergency operations facility for management of overall licensee emergency response. The facility will meet the requirements of 10 CFR 50.47(b)(8) and Section IV.E, "Emergency Facilities and Equipment," of Appendix E to 10 CFR Part 50.	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)

Item No.	Description of COL Information Item	Section
COL Item 13.4-1:	A COL applicant that references the NuScale Power Plant design certification will provide site-specific information, including implementation schedule, for operational programs: • Inservice inspection programs (refer to Section 5.2, Section 5.4, and Section 6.6) • Inservice testing programs (refer to Section 3.9 and Section 5.2) • Environmental qualification program (refer to Section 3.11) • Pre-service inspection program (refer to Section 5.2 and Section 5.4) • Reactor vessel material surveillance program (refer to Section 5.3) • Pre-service testing program (refer to Section 3.9.6, Section 5.2, and Section 6.6) • Containment leakage rate testing program (refer to Section 6.2) • Fire protection program (refer to Section 9.5) • Process and effluent monitoring and sampling program (refer to Section 11.5) • Radiation protection program (refer to Section 12.5) • Non-licensed plant staff training program (refer to Section 13.2) • Reactor operator training program (refer to Section 13.2) • Reactor operator requalification program (refer to Section 13.2) • Emergency planning (refer to Section 13.3) • Process control program (PCP) (refer to Section 11.4) • Security (refer to Section 13.6) • Quality assurance program (refer to Section 17.5) • Maintenance rule (refer to Section 17.6) • Motor-operated valve testing (refer to Section 3.9) • Initial test program (refer to Section 14.2)	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 Regulatory Guide 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 operators use in the main control room and locally in the plant, including normal operating procedures, abnormal operating procedures, and emergency operating procedures. The COL applicant will describe the classification system for these procedures, and the general format and content of the different classifications.	13.5
COL Item 13.5-3:	A COL applicant that references the NuScale Power Plant design certification will describe the site-specific maintenance and other operating procedures, including how these procedures are classified, and the general format and content of the different classifications. The categories of procedures listed below should be included: • plant radiation protection procedures • emergency preparedness procedures • calibration and test procedures • chemical-radiochemical control procedures • radioactive waste management procedures • maintenance and modification procedures • material control procedures • plant security procedures	13.5
COL Item 13.5-4:	A COL applicant that references the NuScale Power Plant design certification will provide a plan for the development, implementation, and control of administrative procedures, including preliminary schedules for preparation and target dates for completion. Additionally, the COL applicant will identify the group within the operating organization responsible for maintaining these procedures.	13.5

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

Item No.	Description of COL Information Item	Section
COL Item 13.5-5:	A COL applicant that references the NuScale Power Plant design certification will provide a plan for the development, implementation, and control of operating procedures, including preliminary schedules for preparation and target dates for completion. Additionally, the COL applicant will identify the group within the operating organization responsible for maintaining these procedures.	13.5
COL Item 13.5-6:	Not used.	13.5
COL Item 13.5-7:	A COL applicant that references the NuScale Power Plant design certification will provide a plan for the development, implementation, and control of emergency operating procedures (EOPs), including preliminary schedules for preparation and target dates for completion. Included in the submittal is the Procedures Generation Package, consisting of the following: • Plant-Specific Technical Guidelines, which are guidelines based on analysis of transients and accidents that are specific to the COL applicant's plant design and operating philosophy. • A plant-specific writer's guide that details the specific methods to be used by the COL applicant in preparing EOPs based on the Plant-Specific Technical Guidelines. • A description of the program for verification and validation of the EOPs. • A description of the program for training operators on the EOPs. Additionally, the COL applicant will identify the group within the operating organization responsible for maintaining these procedures.	13.5
COL Item 13.5-8:	A COL applicant that references the NuScale Power Plant design certification will provide a plan for the development, implementation, and control of maintenance and other operating procedures, including preliminary schedules for preparation and target dates for completion. Additionally, the COL applicant will identify what group or groups within the operating organization have the responsibility for maintaining and following these procedures.	13.5
COL Item 13.6-1:	A COL applicant that references the NuScale Power Plant design certification will provide the following: • Security Plans (Physical Security, Security Training and Qualification, and Safeguards Contingency Plans) • proposed site security provisions to be implemented during construction and as modules are completed and become operational of a new plant • portions of the physical security system not located within the nuclear island and structures	13.6
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, "NuScale Design of Physical Security Systems" (Reference 13.6-1).	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 structures, systems, and components.	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

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

Item No.	Description of COL Information Item	Section
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 seismic monitoring system 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 NuScale Power Modules 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 inspections, tests, analyses, and acceptance criteria for structures, systems, and components 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 and generic Technical Specification Bases.	16.1
COL Item 16.1-2	A COL applicant that references the NuScale Power Plant design certification will prepare and maintain an owner-controlled requirements manual that includes owner-controlled limits and requirements described in the Bases of the Technical Specifications or as otherwise specified in the FSAR.	16.1
COL Item 16.1-3	A COL applicant that references the NuScale Power Plant design certification, and uses allocations for sensor response times based on records of tests, vendor test data, or vendor engineering specifications as described in the Bases for Surveillance Requirement 3.3.1.3, will do so for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.	16.1
COL Item 17.4-1:	A 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 site-specific structures, systems, and components within the scope of the Reliability Assurance Program.	17.4
COL Item 17.4-3:	A COL applicant that references the NuScale Power Plant design certification will identify the quality assurance controls for the Reliability Assurance Program structures, systems, and components during site-specific design, procurement, fabrication, construction, and preoperational testing activities.	17.4
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

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

Item No.	Description of COL Information Item	Section
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 (e.g., liquefaction, slope failure), 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 validity of the "key assumptions" and data used in the design certification application probabilistic risk assessment (PRA) and modify, as necessary, for applicability to the as-built, as-operated PRA.	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 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 site-specific regulatory treatment of nonsafety systems (RTNSS) structures, systems, and components and applicable RTNSS process controls.	19.3
COL Item 20.1-1:	Not used.	20.1
COL Item 20.1-2:	Not used.	20.1
COL Item 20.1-3:	Not used.	20.1
COL Item 20.1-4:	Not used.	20.1
COL Item 20.1-5:	Not used.	20.1
COL Item 20.1-6:	Not used.	20.1
COL Item 20.1-7:	Not used.	20.1

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

Item No.	Description of COL Information Item	Section
COL Item 20.1-8:	A COL applicant that references the NuScale Power Plant design certification will develop procedures, training, and a qualification program for operations, maintenance, testing, and calibration of ultimate heat sink level instrumentation to ensure the level instruments will be available when needed and personnel are knowledgeable in interpreting the information as addressed in Nuclear Energy Institute (NEI) 12-02, Revision 1, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation."	20.1
COL Item 20.2-1:	A COL applicant that references the NuScale Power Plant design certification will develop enhanced firefighting capabilities in accordance with 10 CFR 50.155(b)(2). 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 Plant 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

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 is 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 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, "Gap Analysis Summary Report," NP-RT-0612-023, Revision 1.

Table 1.9-1: Conformance Status Legend

Conformance Status Code	Description
Conforms	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.
Partially Conforms	The design conforms to those portions of the requirement or guidance that can be appropriately applied as written. 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: • 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. • 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. • The intent of a regulatory requirement or guidance is applicable, but the specific language refers to one of the following: - a different type of LWR design - an SSC that is not part of the design, but for which a substantively equivalent function is served by other SSC within the design
Not Applicable	The regulation or guidance is not appropriate to apply and therefore conformance is not required. The following are examples: • The regulatory requirement or guidance is applicable only to BWR designs. • The regulatory requirement or guidance is applicable only to large pressurized water reactor (PWR) designs. • The regulatory requirement or guidance is applicable to the design, but is the responsibility of the COL applicant. • The regulatory requirement or guidance is applicable to SSC that are not part of the design.
Departure	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).[1] [1]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.

Table 1.9-2: Conformance with Regulatory Guides

RG	Division Title	Rev.	Conformance Status	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	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	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	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	The onsite electrical AC power systems do not contain 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	Partially Conforms	The NuScale design complies with the intent of RG 1.7 regulatory positions that address hydrogen and oxygen monitors, atmosphere mixing, hydrogen gas production, and containment structural integrity. However, the NuScale design differs from the system designs the guidance addresses. The NuScale design combustible gas control system does not use combustible gas control systems. The NuScale design supports an exemption to 10 CFR 50.44(c)(2) as described in DCA Part 7, section 2.	6.2.5
1.8	Qualification and Training of Personnel for Nuclear Power Plants	3	Not 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	The NuScale design does not require or include safety-related emergency diesel generators.	Not Applicable
1.11	Instrument Lines Penetrating the Primary Reactor Containment	1	Not Applicable	No lines penetrate the NPM containment.	Not Applicable

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.12	Nuclear Power Plant Instrumentation for Earthquakes	2	Partially Conforms	Selection of specific equipment is the responsibility of the COL applicant or licensee. In addition, seismic detectors cannot be installed inside the containment so Section 3.7.3 indicates they are installed in the RXB.	3.7 12.3
1.13	Spent Fuel Storage Facility Design Basis	2	Partially Conforms	The design of the new and spent fuel storage facility complies with Regulatory Position C.8, Makeup Water 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, 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. In addition, 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.	3.2 9.1 9.2 3.5.2
1.14	Reactor Coolant Pump Flywheel Integrity	1	Not Applicable	This guidance is applicable only to PWR designs that rely on reactor coolant pumps. The NuScale design uses passive natural circulation of the primary coolant, eliminating the need for reactor coolant pumps.	Not Applicable
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	3	Conforms	The first operational NPM is classified as a prototype in accordance with RG 1.20. Thus, the portions of this RG which apply to prototype reactors are applicable to the first operational NPM. After the first NPM is qualified as a valid prototype, subsequent NPMs are classified as non-prototype category I and the non-prototype portions of the RG apply. The identification of departures from RG 1.20 is the responsibility of the COL applicant or licensee.	3.9 5.4 14.2

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.21	Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste	2	Partially Conforms	Site-specific, programmatic and operational aspects are the responsibility of the COL applicant or licensee.	11.5
1.22	Periodic Testing of Protection System Actuation Functions	0	Conforms	None.	7.2
1.23	Meteorological Monitoring Programs for Nuclear Power Plant	1	Not Applicable	This guidance is the responsibility of the COL applicant or licensee.	Not Applicable
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure	0	Not Applicable	Site-specific guidance is the responsibility of the COL applicant or licensee.	Not Applicable
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	This RG pertains to TID14844 source terms used by licensees of existing reactors that are not authorized to use the alternative source term under 10 CFR 50.67. 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 10 CFR 50.67).	Not Applicable
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	4	Conforms	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 9.2 9.3 10.3 10.4
1.27	Ultimate Heat Sink for Nuclear Power Plants	3	Not Applicable	RG does not apply to plants that use a passive cooling system to transfer heat to the ultimate heat sink. The NuScale design uses a passive cooling system.	Not Applicable

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.28	Quality Assurance Program Criteria (Design and Construction)	4	Conforms	None.	3.13 4.5 5.2 5.3 5.4 6.1 7.0 7.2 14.3 17.1 17.5
1.29	Seismic Design Classification for Nuclear Power Plants	5	Partially Conforms	Each SSC described in Staff Regulatory Guidance C.1.a through C.1.h is designated as Seismic Category I. SSC that meet Staff Regulatory Guidance C.1.i are designated Seismic Category II rather than Seismic Category I. The seismic classification from RG 1.29 applicable to a specific component is described throughout the FSAR.	3.2
1.30	Safety Guide 30 - Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	-	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 NQA-1-2008 and the NQA-1a-2009 addenda, as endorsed in RG 1.28, Rev. 4. NQA-1-2008 and NQA-1a-2009 (Subpart 2.4) reference IEEE Std. 336-1985 (as opposed to IEEE Std. 336-1971). The substantive content and intent of RG 1.30 is contained in Subpart 2.4 of NQA-1-2008 and NQA-1a-2009 and IEEE Std. 336-1985, which is applicable to the NuScale design per NQA-2008 and NQA-1a-2009.	Not Applicable
1.31	Control of Ferrite Content in Stainless Steel Weld Metal	4	Conforms	None.	4.5 5.2 6.1
1.32	Criteria for Power Systems for Nuclear Power Plants	3	Partially Conforms	RG 1.32 is not applicable to the offsite and onsite AC power systems. The EDSS conforms to RG 1.32 to the extent described in Section 8.3.	8.2 8.3

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.33	Quality Assurance Program Requirements (Operation)	3	Not Applicable	This guidance is the responsibility of the COL applicant or licensee.	Not Applicable
1.34	Control of Electroslag Weld Properties	1	Conforms	None.	5.3
1.35	Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments	3	Not Applicable	The NuScale design uses a steel containment vessel (i.e., does not use concrete in its design).	Not Applicable
1.35.1	Determining Prestressing Forces for Inspection of Prestressed Concrete Containments	-	Not Applicable	The NuScale design uses a steel containment vessel (i.e., does not use concrete in its design).	Not Applicable
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel	-	Not Applicable	The NuScale design does not use nonmetallic thermal insulation on RCPB or CNV components.	Not Applicable
1.40	Qualification of Continuous Duty Safety-Related Motors for Nuclear Power Plants	1	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	-	Partially Conforms	Portions of this guide are applicable to preoperational testing of divisional EDSS load groups. It does not apply to NuScale AC power systems or the EDNS.	8.3 14.2
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components	1	Conforms	None.	5.2 5.3 6.1
1.44	Control of the Processing and Use of Stainless Steel	1	Partially Conforms	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 6.1

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.45	Guidance on Monitoring and Responding to Reactor Coolant System Leakage	1	Conforms	The design satisfies RG 1.45 guidance by using two systems 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: 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 less than one hour for a leakage rate greater than one gpm. Regulatory Position C.2.4 is satisfied because 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 because 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.	3.6 5.2 6.2 9.3 11.5 14.2 14.3
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	1	Conforms	None.	7.2
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel	1	Conforms	None.	5.2 6.1

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

RG	Division Title	Rev.	Conformance Status	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	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 NuScale design does not rely on ESF filter and atmosphere cleanup systems to mitigate the consequences of a design basis accident. Nonsafety-related normal ventilation systems provide atmosphere cleanup capability, as necessary, that meets the design, testing, and maintenance guidelines 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 at a NuScale Power Plant, 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	None.	7.1 8.3 15.1 15.2 15.5
1.54	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants	2	Partially Conforms	Applicable except for portions of this RG that govern operational aspects (e.g., maintenance of safety-related coatings) that are the responsibility of the COL applicant or licensee.	11.2

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components	2	Partially Conforms	Applicable except for reference to 10 CFR 50.34(f)(3)(v), since per 10 CFR 50.34(f) and 10 CFR 52.47(a)(8), a design certification applicant does not have to show compliance with 10 CFR 50.34(f)(3)(v). Use of typical reactor pressure vessel load combinations for Class 1 vessels is more applicable to the containment vessel than using the load combinations specified in RG 1.57 because of the increased quality of the fabrication, inspection, and testing required by American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel ASME Code, Section III, Subsection NB for a Class 1 vessel. Additionally, the use of load combinations for Class 1 vessels is more applicable to the containment vessel than using the load combinations specified in RG 1.57 because of the quality of the fabrication, inspection, and testing required by ASME Code, Section III, Subsection NB for a Class 1 vessel.	3.8 6.2
1.59	Design Basis Floods for Nuclear Power Plants	2	Not Applicable	The NuScale design assumes the NPP is located above the probable maximum flood height (including wind-induced wave run-up).	Not Applicable
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants	2	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.	Not Applicable
1.61	Damping Values for Seismic Design of Nuclear Power Plants	1	Conforms	In accordance with the guidance of RG 1.61, an alternative damping value for the NPM substructure was determined.	3.7 3.8 3.12
1.62	Manual Initiation of Protective Actions	1	Conforms	None.	7.1 7.2
1.63	Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants	3	Partially Conforms	The portion of the RG 1.63 guidance that endorses IEEE-317-1983 is applicable. IEEE 741-1997 is used for external circuit protection of electrical penetration assemblies instead of IEEE 741-1986 as endorsed by RG 1.63. The 1997 version, including the additional design enhancements, is consistent with RG 1.63.	3.8.2 3.11 8.1 8.3 14.3

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.65	Materials and Inspections for Reactor Vessel Closure Studs	1	Partially Conforms	Inservice inspection is the responsibility of the COL applicant. The reactor pressure vessel (RPV) bolting material is described in Table 5.2-4. The material is not subject to the concerns addressed by RG 1.65 Positions 1(a)(i) and 2(b). Therefore, these Positions do not apply to the RPV bolting material.	3.13 5.3
1.68	Initial Test Programs for Water-Cooled Nuclear Power Plants	4	Partially Conforms	This guidance is applicable except for aspects that are BWR-specific or address specific PWR SSC design features not in the NuScale design. Site-specific program implementation activities are the responsibility of the COL applicant or licensee that references the NuScale certified design.	4.4 5.4 8.2 8.3 9.3.2 10.4 14.2 14.3
1.68.1	Initial Test Program of Condensate and Feedwater Systems for Light-Water Reactors	2	Partially Conforms	This RG is applicable except for aspects that are BWR-specific or address specific PWR design features not in the NuScale design.	14.2
1.68.2	Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants	2	Partially Conforms	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 or licensee.	14.2
1.68.3	Preoperational Testing of Instrument and Control Air Systems	1	Partially Conforms	This guidance is applicable except for site-specific aspects, including test performance and records retention, which are the responsibility of the COL applicant or licensee.	14.2
1.69	Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants	1	Partially Conforms	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 or licensee.	3.8 12.3 14.2
1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)	3	Not Applicable	RG 1.206 and NuScale Design Specific Review Standards (DSRS) are used.	Not Applicable

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.71	Welder Qualification for Areas of Limited Accessibility	1	Partially Conforms	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 or licensee.	4.5 5.2 6.1
1.72	Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin	2	Not Applicable	The design does not use fiberglass piping in spray pond applications (or for the UHS design).	Not Applicable
1.73	Qualification Tests for Safety-Related Actuators in Nuclear Power Plants	1	Partially Conforms	The guidance is applicable except for portions that apply to high temperature gas-cooled reactor designs.	3.11
1.75	Criteria for Independence of Electrical Safety Systems	3	Conforms	None.	7.1 7.2 8.3 9.5 14.3
1.76	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants	1	Conforms	Region 1 (bounding) characteristics are used as design parameters. The COL applicant or licensee is responsible for confirming the characteristics.	2.3 3.3 3.5 3.8 5.0

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors	-	Partially Conforms	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 SRP (NUREG-0800) 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.	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	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 or licensee.	3.2 6.4 9.4
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors	2	Partially Conforms	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 14.2

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.79.1	Initial Test Program of Emergency Core Cooling Systems for New Boiling-Water Reactors	-	Not Applicable	RG 1.79.1 is applicable to BWRs only.	Not Applicable
1.81	Shared Emergency and Shut-down Electric Systems for Multi-Unit Nuclear Power Plants	1	Not Applicable	RG 1.81 is not relevant to the AC power systems. As described in Section 8.3, the EDSS conforms to portions of RG 1.81.	Not Applicable
1.82	Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident	4	Partially Conforms	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. However, the NuScale design differs from the system designs the guidance addresses. 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 does not require operator action to mitigate debris accumulation. 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: • Position C1.1.1.9 (assessment of the possibility of downstream clogging). • Position C1.1.1.10 (buildup of debris and chemical reaction products downstream). • Position C.1.1.2 (minimization of debris source term, cleanness programs, monitoring/sampling for latent debris, insulation selection, restriction on coatings and cladding of carbon steel).	6.3

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
				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. The NuScale design does not comply with regulatory position C.1.2 (alternative water sources for inoperable strainers). The NuScale design 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: • Position C.1.3.1 (net positive suction head) • Portions of position C.1.3.2 that are not consistent with the NuScale design The NuScale design does not comply with regulatory positions C1.3.7 (upstream effects) or C.1.3.9 (strainer structural integrity). The NuScale design does not comply with regulatory position C1.3.12 (prototypical head loss testing). 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. The NuScale design does not comply with regulatory position C.3.	
1.84	Design, Fabrication, and Materials Code Case Acceptability, ASME Section III	36	Conforms	None.	3.12 3.13 4.5 5.2
1.86	Termination of Operating Licenses for Nuclear Reactors	-	Not Applicable	This RG governs the process for terminating nuclear reactor operating licenses.	Not Applicable

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
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	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	This RG is applicable except for: (1) aspects that are BWR-specific or related to SSC 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 are superseded by RG 1.183 for new reactors.	3.8.2 3.11 Appendix 3C
1.90	Inservice Inspection of Pre-stressed Concrete Containment Structures with Grouted Tendons	2	Not Applicable	This RG is applicable 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	This guidance governs the performance of site-specific evaluations and is the responsibility of the COL applicant or licensee.	Not Applicable
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis	3	Conforms	None.	3.7 3.8 3.9 3.10 3.12
1.93	Availability of Electric Power Sources	1	Not Applicable	This RG is not identified as an applicable RG in DSRS Section 8.1.	Not Applicable
1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants	1	Not Applicable	This RG is applicable only to BWR designs.	Not Applicable

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.97	Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants	4	Partially Conforms	The NuScale design satisfies power supply requirements in Section 6.6 of IEEE Std 497-2002 for Type B and C variables with highly reliable power rather than with Class 1E. The portions of RG 1.97 dealing with 10 CFR 50.34(f)(2)(xix) are addressed in Section 19.2.3.3.8.	3.11 Appendix 3C 5.4 7.1 7.2 8.3 11.5 12.3 14.3 19.2
1.98	Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor	0	Not Applicable	This RG is applicable only to BWR designs.	Not Applicable
1.99	Radiation Embrittlement of Reactor Vessel Materials	2	Conforms	None.	5.3
1.100	Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants	3	Partially Conforms	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. RG 1.100 endorses ASME QME-1 2007. NuScale complies with the non-mandatory Appendix QR-B with the following exceptions: QR-B5200, Identification and Specification of Qualification Requirements, (g) material activation energy. QR-B5300 Selection of Qualification Methods for determination and recording of shelf life of nonmetallics. QR-B5500 Documentation, (h) shelf life preservation requirements. Appendix 3C describes the exceptions cited.	3.9 3.10 5.2 14.3 Appendix 3C

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.101	Emergency Response Planning and Preparedness for Nuclear Power Reactors	5	Not Applicable	This RG is the responsibility of the COL applicant proposing to site a power plant that meets the definition of co-located.	Not Applicable
1.102	Flood Protection for Nuclear Power Plants	1	Applicable	The design assumes the NPP is located above the probable maximum flood height (including wind induced wave run-up).	2.4 3.4 3.8
1.105	Setpoints for Safety-Related Instrumentation	3	Partially Conforms	Chapter 15 analyses use the safety-related setpoints described in Chapter 7. This RG endorses ISA-67.04.01-1994, however, the NuScale Instrument Setpoint Methodology Technical Report (TR-0616-49121) applies the guidance contained in ISA-67.04.01-2006. A key difference is that the 1994 version of ISA-67.04.01 uses an allowable value to determine instrument channel operability during surveillance testing and calibration. The 2006 version of ISA-67.04.01 provides updated guidance for evaluating instrument channel operability based on the comparison of the as-found to the as-left value from the previous instrument calibration for the instrument setpoint.	7.2 15.1 15.2 15.4 15.5 15.6
1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves	2	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 NuScale 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	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 pre-stressed tendons in its design).	Not Applicable
1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I	1	Partially Conforms	This RG is applicable except for specification of site-specific information (e.g., meteorological data). Site-specific information is the responsibility of the COL applicant.	11.2 11.3

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors	1	Partially Conforms	This RG is applicable except for aspects related to performance of a site-specific cost-benefit analysis. Site-specific information is the responsibility of the COL applicant or licensee.	11.2 11.3
1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors	1	Partially Conforms	This RG is applicable except for specification of site-specific dispersion data. Site-specific information is the responsibility of the COL applicant or licensee.	2.3 3.3
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Nuclear Power Reactors	1	Partially Conforms	This RG is applicable except for specification of site-specific information (e.g., meteorological data). Site-specific information is the responsibility of the COL applicant or licensee.	2.3 11.2 11.3
1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I	1	Not Applicable	This RG provides guidance for analyzing the aquatic dispersion of radioactive liquid effluents from component failures, in accordance with BTP 11-6. Because the NuScale facility provides an approved design mitigative feature (metal-lined concrete dike around the PSCS storage tank), such an analysis is not required.	Not Applicable
1.114	Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit	3	Partially Conforms	Site-specific guidance is the responsibility of the COL applicant or licensee. 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). Portions of this guidance that implement operator staffing requirements of 10 CFR 50.54(m)(2)(i) and (iii) (e.g., Position C.1.5) are not applicable to COL applicants.	18.5
1.115	Protection Against Turbine Missiles	2	Conforms	None.	3.5.3
1.117	Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants	2	Conforms	Confirmation that nearby structures exposed to extreme wind loads will not adversely affect the Reactor Building or the Seismic Category I portion of the Control Building is the responsibility of the COL applicant or licensee.	3.5 9.1

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.118	Periodic Testing of Electric Power and Protection Systems	3	Partially Conforms	Site-specific guidance is the responsibility of the COL applicant or licensee.	7.2 8.3 14.2
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes	August 1976	Conforms	None.	5.4
1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components	1	Partially Conforms	Site-specific guidance is the responsibility of the COL applicant or licensee.	3.7 3.12
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Supports	3	Conforms	None.	3.9
1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants	2	Not Applicable	The NuScale 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	None.	4.2
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants	1	Not 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 or licensee.	Not Applicable
1.128	Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	2	Partially Conforms	The EDSS uses VRLA batteries; thus IEEE Std 1187-2013 is applied.	8.3 14.2
1.129	Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	3	Partially Conforms	The EDSS uses VRLA batteries. NuScale applies IEEE Std 1188-2005 with the 2014 amendment.	8.3

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.130	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Supports	3	Conforms	None.	3.9
1.132	Site Investigations for Foundations of Nuclear Power Plants	2	Not Applicable	This RG governs site investigations performed as part of site selection. Site-specific guidance is the responsibility of the COL applicant or licensee.	Not Applicable
1.133	Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors	1	Not 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. Additional justification for this information is in Section 4.4 of the FSAR.	Not Applicable
1.134	Medical Evaluation of Licensed Personnel at Nuclear Power Plants	3	Not Applicable	This RG governs site-specific operational program activities. Compliance with site-specific guidance is the responsibility of the COL applicant or licensee.	Not Applicable
1.136	Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments	3	Not Applicable	This guidance is applicable only to LWR designs that use concrete containments. The NuScale design uses a steel containment vessel.	Not Applicable
1.137	Fuel-Oil Systems for Standby Diesel Generators	2	Not Applicable	The design does not rely on or include safety-related emergency diesel generators.	Not Applicable
1.138	Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants	2	Not Applicable	This guidance is related to site-specific laboratory investigation activities. Site-specific guidance is the responsibility of the COL applicant or licensee.	Not Applicable
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	Design-related aspects of this guidance are applicable. Aspects related to construction, testing, and repairs are the responsibility of the COL applicant or licensee.	3.2 9.4 11.3 12.3 14.2
1.141	Containment Isolation Provisions for Fluid Systems	1	Conforms	The NuScale design conforms to the requirements of RG 1.141 through adherence to ANS N271-1976. Note: the provisions of ANSI/ANS 56.2-1984, Section 3.6.5 are applied to penetrations with both CIVs outside containment that serve non-ESF process systems.	6.2

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)	2	Partially Conforms	The intent of this guidance is applicable but the language endorses ACI 349-1997 with 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, and does not use 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	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 are the responsibility of the COL applicant or licensee.	3.2 3.3 3.5 3.7 9.2.6 11.2 11.3 11.4 14.3
1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants	1	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.	Not Applicable
1.147	Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1	17	Partially Conforms	Performance of inservice inspections per the American Society of Mechanical Engineers Boiler and Pressure Vessel Code is the responsibility of the COL applicant or licensee.	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	Simulation facilities and conduct of licensed operator training and qualification are the responsibility of the COL applicant or licensee.	Not Applicable
1.151	Instrument Sensing Lines	1	Partially Conforms	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 or licensee.	7.2

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.152	Criteria for Use of Computers in Safety Systems of Nuclear Power Plants	3	Partially Conforms	The NuScale I&C development lifecycle differs from the conceptual waterfall lifecycle in RG 1.152. The applicable tasks from the RG lifecycle model will be mapped to the I&C development lifecycle. Compliance with Clause 5.5 of IEEE 7-4.3.2-2003 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 14.3
1.153	Criteria for Safety Systems	1	Conforms	Applicable to EDSS.	3.11 8.3
1.155	Station Blackout	1	Partially Conforms	The design conforms to the aspects of the RG as it pertains to passive plant designs.	5.4 6.2 8.3 8.4 9.3 9.5 10.3 14.2
1.156	Qualification of Connection Assemblies for Nuclear Power Plants	1	Conforms	None.	3.11
1.157	Best-Estimate Calculations of Emergency Core Cooling System Performance	-	Not Applicable	Best estimate calculations are not used.	Not Applicable
1.158	Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants	-	Conforms	The DC system batteries are non-Class 1E.	3.11
1.159	Assuring the Availability of Funds for Decommissioning Nuclear Reactors	2	Not Applicable	Decommissioning funding activities are the responsibility of the COL applicant.	Not Applicable
1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	3	Not Applicable	Monitoring the effectiveness of maintenance activities is the responsibility of the COL applicant.	Not Applicable

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.161	Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb	-	Not 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 10 CFR 50, Appendix G and the provisions of RG 1.161 would be the responsibility of the COL applicant or licensee (see discussion of RG 1.162).	Not Applicable
1.162	Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels	-	Not Applicable	If thermal annealing becomes necessary, the requirements of 10 CFR 50.66 and the provisions of RG 1.162 would be the responsibility of the COL applicant or licensee.	Not Applicable
1.163	Performance-Based Containment Leak-Test Program	-	Not Applicable	The design of containment penetrations supports performance of local leak rate tests (Type B and Type C tests) in accordance with the guidance provided in ANSI/ANS 56.8, Regulatory Guide 1.163, and NEI 94-01. The NuScale system design, in conformance with 10 CFR 50.54(o), accommodates the 10 CFR 50, Appendix J, test method frequencies of Option A or Option B. This RG is the responsibility of a COL applicant or licensee that seeks to implement Option B.	Not Applicable
1.166	Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post Earthquake Actions	-	Not Applicable	This RG governs programmatic activities (earthquake planning and post-earthquake actions) that are the responsibility of the COL applicant or licensee.	Not Applicable
1.167	Restart of a Nuclear Power Plant Shut Down by a Seismic Event	-	Not Applicable	This RG governs post-earthquake inspections and tests that are the responsibility of the COL applicant or licensee.	Not Applicable

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.168	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	2	Partially Conforms	The NuScale design applies RG 1.152, Revision 3 and IEEE Std. 7-4.3.2-2003 that it endorses. For RG 1.168, the requirements of IEEE 1012-2004 are tailored to the NuScale I&C development lifecycle, which is different from the conceptual waterfall lifecycle in IEEE 1012-2004. The applicable tasks from IEEE 1012-2004 to the I&C development are mapped. Some administrative mandatory requirements in the standard conflict with established Engineering or QA documentation requirements. The requirements of IEEE 1028-2008 are tailored to the NuScale I&C development lifecycle.	7.2 14.3
1.169	Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	Partially Conforms	For this RG, the requirements of IEEE 828-2005 are tailored to the NuScale I&C development lifecycle, which is different from the conceptual waterfall lifecycle in RG 1.152. The applicable tasks from IEEE 828-2005 are mapped to the NuScale I&C development lifecycle.	7.2 14.3
1.170	Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	Partially Conforms	Requirements of IEEE 829-2008 are tailored to the NuScale I&C development lifecycle, which is different from the conceptual waterfall lifecycle in RG 1.152. The applicable tasks from IEEE 829-2008 are mapped to the NuScale I&C development lifecycle. NuScale takes exception to some administrative mandatory requirements in the standard that conflict with established Engineering or quality documentation requirements.	7.2 14.3
1.171	Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	Partially Conforms	NuScale takes exception to some administrative mandatory requirements in IEEE 1008-1987 that conflict with established Engineering or quality documentation requirements.	7.2 14.3
1.172	Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	Partially Conforms	NuScale takes exception to some administrative mandatory requirements in IEEE 830-1998 standard that conflict with established Engineering or quality documentation requirements.	7.2 14.3

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.173	Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	Partially Conforms	Requirements of IEEE 1074-2006 are tailored to the NuScale I&C development lifecycle, which differs from the conceptual waterfall lifecycle in RG 1.152. Applicable tasks from IEEE 1074-2006 are mapped to the NuScale I&C development lifecycle. NuScale takes exception to some administrative mandatory requirements in the standard that conflict with established Engineering or quality documentation requirements.	7.2 14.3
1.174	An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis	2	Not Applicable	This RG is applicable to licensees seeking changes in licensing basis, and is the responsibility of the licensee.	Not Applicable
1.175	An Approach for Plant-Specific, Risk Informed Decision making: Inservice Testing	-	Not Applicable	This RG is applicable to licensees seeking change to licensing basis using a risk-informed approach and is the responsibility of the COL applicant or licensee. NuScale is not using a risk-informed approach for IST.	Not Applicable
1.177	An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications	1	Not Applicable	This RG applies to existing licensees seeking NRC approval of changes to their plant-specific technical specifications. However, NuScale considered this guidance, as appropriate, in risk-informed technical specification development.	16.1
1.178	An Approach for Plant-Specific Risk-Informed Decision making for Inservice Inspection of Piping	1	Not 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 or licensee.	Not Applicable
1.179	Standard Format and Content of License Termination Plans for Nuclear Power Reactors	1	Not Applicable	This guidance governs site-specific decommissioning and license termination planning and implementation activities that are the responsibility of the COL applicant or licensee.	Not Applicable

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.180	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems	1	Partially Conforms	Aspects of this guidance related to the design of SSC to address effects of electromagnetic and radio-frequency interference (EMI/RFI) are applicable. Aspects of this guidance related to the design of site-specific SSC 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 or licensee.	3.11 7.2 9.5
1.181	Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)	-	Not Applicable	This guidance governs site-specific reporting activities that are the responsibility of the COL applicant or licensee.	Not Applicable
1.183	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors	0	Partially Conforms	For the NuScale design, the safety analysis shows that core damage does not occur during any design basis event. Thus, the RG 1.183 guidance is partially applicable to the NuScale dose consequence analysis. The basis and justification for departures from the RG 1.183 guidance for the limiting dose consequence analysis for NuScale are provided in a Topical Report. NuScale uses the alternative source term non-LOCA or transient-specific guidance of RG 1.183 for Chapter 15 events.	6.4 9.3 12.2 15.0.2 15.0.3 15.6 15.7 15.10
1.184	Decommissioning of Nuclear Power Reactors	1	Not Applicable	This RG governs site-specific decommissioning planning and implementation activities that are the responsibility of the COL applicant or licensee.	Not Applicable
1.185	Standard Format and Content for Post-Shutdown Decommissioning Activities Report	1	Not Applicable	This RG governs site-specific decommissioning planning activities that are the responsibility of the COL applicant or licensee.	Not Applicable
1.186	Guidance and Examples for Identifying 10 CFR 50.2 Design Bases	-	Not Applicable	This RG endorses NEI 97-04 Appendix B is the responsibility of the COL applicant or licensee.	Not Applicable
1.187	Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments	-	Not Applicable	This RG implements change process requirements that are the responsibility of the COL applicant or licensee.	Not Applicable
1.188	Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses	1	Not Applicable	This RG is applicable 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	Comments	Section
1.189	Fire Protection for Nuclear Power Plants	2	Partially Conforms	This RG is applicable except for portions (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 or licensee.	9.4 9.5 Appendix 9A 11.3 14.3 19.1
1.190	Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence	-	Partially Conforms	The neutron flux and fluence calculation methods are consistent with the guidance of RG 1.190 with exceptions as described in NuScale Technical Report TR-0116-20781, "Fluence Calculation Methodology and Results."	5.3
1.191	Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown	-	Not 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	Implementation of inservice testing is the responsibility of the COL applicant or licensee.	Not Applicable
1.193	ASME Code Cases Not Approved for Use	4	Conforms	ASME code cases in RG 1.193 are not used unless authorized by the NRC pursuant to 10 CFR 50.55a(z).	5.2
1.194	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants	-	Conforms	None.	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	This RG pertains to TID14844 source terms used by licensees of existing reactors that are not authorized to use the alternative source term under 10 CFR 50.67. 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 10 CFR 50.67.	Not Applicable

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.196	Control Room Habitability at Light-Water Nuclear Power Reactors	1	Partially Conforms	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 or licensee.	3.8 6.4 18.7 (via the Human-System Interface Design Results Summary Report)
1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors	-	Not Applicable	Inleakage testing activities are the responsibility of the COL applicant or licensee referencing the certified design.	Not Applicable
1.198	Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites	-	Not Applicable	The evaluation governed by this guidance is the responsibility of the COL applicant or licensee referencing the certified design.	Not Applicable
1.199	Anchoring Components and Structural Supports in Concrete	-	Partially Conforms	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	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)

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.201	Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance	1	Partially Conforms	10 CFR 50.69 provides an alternative regulatory framework for a licensee to use a risk-informed process for categorizing SSC by their safety significance, and based on this process can remove SSC of low safety significance from the scope of identified special treatment requirements. Thus, these requirements are applicable to licensees that choose this alternative framework. NuScale uses 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 combines the traditional approach using the definitions of 10 CFR 50.2 and guidance of RG 1.26 and SRP Section 3.2.2 with the alternative regulatory framework similar to that prescribed in 10 CFR 50.69 and RG 1.201 (and NEI 00-04 endorsed by RG 1.201). This methodology is consistent with SECY-03-0047 and SECY-10-0034, which recommend the use of a probabilistic, risk-informed approach for SSC safety classification. NuScale applies the guidance of RG 1.201 and NEI 00-04 to the extent appropriate given the baseline risk metrics for the NuScale advanced reactor design.	3.2
1.202	Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors	-	Not 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	None.	15.0.2
1.204	Guidelines for Lightning Protection of Nuclear Power Plants	-	Conforms	None.	3.8 7.0 7.2 8.1 8.3

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.205	Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants	1	Not Applicable	This RG applies to reactor licensees or 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 or licensees that reference the NuScale design, and that elect to implement the provisions of 10 CFR 50.48(c).	Not Applicable
1.206	Combined License Applications for Nuclear Power Plants (LWR Edition)	-	Partially Conforms	This RG is the 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	None.	3.8 3.9 3.12
1.208	A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion	-	Not Applicable	This guidance is for development of site-specific ground motion response spectra and is the responsibility of the COL applicant or licensee.	Not Applicable
1.209	Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants	-	Conforms	None.	3.11 Appendix 3C 7.2 14.3
1.210	Qualification of Safety-Related Battery Chargers and Inverters for Nuclear Power Plants	-	Not Applicable	The NuScale design does not use safety-related battery chargers or inverters; EDSS battery chargers are not located in a harsh environment.	Not Applicable
1.211	Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants	-	Conforms	None.	3.11
1.212	Sizing of Large Lead-Acid Storage Batteries	November 2008	Partially Conforms	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

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
1.213	Qualification of Safety-Related Motor Control Centers for Nuclear Power Plants	-	Not Applicable	The NuScale electrical system design does not use safety-related motor control centers.	Not Applicable
1.214	Response Strategies for Potential Aircraft Threats	1	Conforms	None.	19.5
1.215	Guidance for ITAAC Closure Under 10 CFR Part 52	2	Not Applicable	This guidance describes acceptable methods of complying with the requirements of 10 CFR 52.99, which is applicable to COL applicants and licensees.	Not Applicable
1.216	Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design-Basis Pressure	0	Conforms	None.	3.8 6.2
1.217	Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts	-	Conforms	None.	19.5
1.218	Condition-Monitoring Techniques for Electric Cables Used in Nuclear Power Plants	-	Not Applicable	The COL holder determines whether a cable is subject to condition monitoring during the development of the maintenance rule (10 CFR 50.65) program. Cables that meet the criteria for inclusion in the maintenance rule program are subject to the guidance of RG 1.218.	Not Applicable
1.219	Guidance on Making Changes to Emergency Plans for Nuclear Power Reactors	-	Not 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	NuScale uses the highest wind speed postulated in Regulatory Position 1 (which occurs in Figure 2 of RG 1.221 Rev. 0) as the wind speed for the design basis hurricane. Confirmation of site characteristics is the responsibility of the COL applicant.	3.3 3.5 3.8
4.1	Radiological Environmental Monitoring for Nuclear Power Plants	2	Not Applicable	This guidance governs site-specific, programmatic environmental monitoring activities that are the responsibility of the COL applicant or licensee.	Not Applicable
4.2	Preparation of Environmental Reports for Nuclear Power Stations	2	Not Applicable	This guidance governs site-specific environmental evaluation activities that are the responsibility of a license or construction permit applicant.	Not Applicable

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
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	This guidance is applicable 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	This guidance governs site-specific environmental activities related to modeling temperature impact of plant discharge on aquatic systems. These activities are the responsibility of the COL applicant or licensee.	Not Applicable
4.7	General Site Suitability Criteria for Nuclear Power Stations	2	Not Applicable	This guidance governs site-specific evaluation activities that are the responsibility of the COL applicant.	Not Applicable
4.9	Preparation of Environmental Reports for Commercial Uranium Enrichment Facilities	1	Not Applicable	This guidance applies only to uranium enrichment facilities.	Not Applicable
4.11	Terrestrial Environmental Studies for Nuclear Power Stations	2	Not 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	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	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 to COL applicants.	Not Applicable
4.16	Monitoring and Reporting Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Cycle Facilities	2	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	This guidance is applicable only to geological repositories.	Not Applicable

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
4.18	Standard Format and Content of Environmental Reports for Near-Surface Disposal of Radioactive Waste	-	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
4.19	Guidance for Selecting Sites for Near-Surface Disposal of Low-Level Radioactive Waste	-	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	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	This guidance is applicable, except for the portions that relate to site-specific, operational aspects that are the responsibility of the COL applicant or licensee referencing the NuScale design.	9.1 9.2 9.4 10.4 11.2 11.3 12.3 12.5 14.3
4.22	Decommissioning Planning During Operations	-	Not 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	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 (UF4) and Uranium Hexafluoride (UF6)	-	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	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	Comments	Section
5.7	Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas	1	Partially Conforms	Site-specific, programmatic aspects of this guidance are the responsibility of the COL applicant or licensee 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	Applicable to Part 70 facilities.	Not Applicable
5.9	Guidelines for Germanium Spectroscopy Systems for Measurement of Special Nuclear Material	2	Not Applicable	This guidance governs processes, procedures, equipment, and methods that are not applicable to the NuScale design.	Not Applicable
5.11	Nondestructive Assay of Special Nuclear Material Contained in Scrap and Waste	1	Not Applicable	This RG applies to facilities that process SNM. The NuScale design 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	Site-specific, programmatic aspects of this guidance are the responsibility of the COL applicant or licensee referencing the NuScale design.	13.6 (via Security Technical Report)
5.13	Conduct of Nuclear Material Physical Inventories	-	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	This RG is applicable to a special nuclear material licensee.	Not Applicable
5.20	Training, Equipping, and Qualifying of Guards and Watchmen	-	Not Applicable	Applicable to COL applicant or licensee.	Not Applicable
5.21	Nondestructive Uranium-235 Enrichment Assay by Gamma Ray Spectrometry	1	Not Applicable	Applicable to Part 70 facilities.	Not Applicable
5.22	Assessment of the Assumption of Normality (Employing Individual Observed Values)	-	Not Applicable	This RG is not applicable to the DCA because NuScale is not a special nuclear material licensee.	Not Applicable
5.23	In Situ Assay of Plutonium Residual Holdup	1	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	Applicable to Part 70 facilities.	Not Applicable
5.26	Selection of Material Balance Areas and Item Control Areas	1	Not Applicable	Applicable to Part 70 facilities.	Not Applicable

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
5.27	Special Nuclear Material Doorway Monitors	-	Not Applicable	Applicable to COL applicant.	Not Applicable
5.28	Evaluation of Shipper-Receiver Differences in the Transfer of Special Nuclear Materials	-	Not Applicable	This RG applies to fuel processing and fuel fabrication licensees.	Not Applicable
5.29	Nuclear Material Control systems for Nuclear Power Plants	2	Not Applicable	This RG is not applicable to the NuScale design but may be used by a COL applicant to meet the material control and accounting requirements in Subpart B of 10 CFR Part 74.	Not Applicable
5.31	Specially Designed Vehicle with Armed Guards for Road Shipment of Special Nuclear Material	1	Not Applicable	Applicable to COL applicant.	Not Applicable
5.33	Statistical Evaluation of Material Unaccounted For	-	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	Applicable to Part 70 processing.	Not Applicable
5.36	Recommended Practice for Dealing with Outlying Observations	1	Not Applicable	This RG is applicable to a special nuclear material licensee.	Not Applicable
5.39	General Methods for the Analysis of Uranyl Nitrate Solutions for Assay, Isotopic Distribution, and Impurity Determinations	-	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	Applicable to Part 70 facilities.	Not Applicable
5.43	Plant Security Force Duties	-	Not Applicable	Applicable to COL applicant or licensee.	Not Applicable
5.44	Perimeter Intrusion Alarm Systems	3	Partially Conforms	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 to COL applicant or licensee.	Not Applicable
5.49	Internal Transfers of Special Nuclear Material (for Comment)	-	Not Applicable	Issued for comment.	Not Applicable

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
5.51	Management Review of Nuclear Material Control and Accounting Systems	0	Not Applicable	This RG applies to fuel cycle facilities.	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 to nuclear power plants.	Not Applicable
5.53	Qualification, Calibration, and Error Estimation Methods for Nondestructive Assay	1	Not Applicable	This RG applies to fuel processing licensees.	Not Applicable
5.54	Standard Format and Content of Safeguards Contingency Plans for Nuclear Power Plants	1	Not Applicable	This guidance governs site-specific physical protection features and security program activities that are the responsibility of the COL applicant or licensee.	Not Applicable
5.55	Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities	0	Not Applicable	Applicable to fuel cycle facilities.	Not Applicable
5.56	Standard Format and Content of Safeguards Contingency Plans for Transportation	0	Not Applicable	Applicable to transportation of special nuclear material.	Not Applicable
5.57	Shipping and Receiving Control of Strategic Special Nuclear Material	1	Not Applicable	Applicable to COL applicant or licensee.	Not Applicable
5.58	Considerations for Establishing Traceability of Special Nuclear Material Accounting Measurements	1	Not Applicable	Applicable to COL applicant or licensee.	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 to COL applicant or licensee.	Not Applicable
5.60	Standard Format and Content of a Licensee Physical Protection Plan for Strategic Special Nuclear Material in Transit	-	Not Applicable	Applicable to COL applicant or licensee.	Not Applicable

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
5.61	Intent and Scope of the Physical Protection Upgrade Rule Requirements for Fixed Sites	-	Not Applicable	This guidance applies to fuel cycle facilities.	Not Applicable
5.62	Reporting of Safeguards Events	1	Not Applicable	This guidance applies to site-specific security issues concerning SNM and is the responsibility of the COL applicant or licensee.	Not Applicable
5.63	Physical Protection for Transient Shipments	-	Not Applicable	Applicable to COL applicant or licensee.	Not Applicable
5.65	Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls	-	Partially Conforms	Site-specific, programmatic aspects of this guidance are the responsibility of the COL applicant or licensee referencing the NuScale design.	13.6 (via Security Technical Report)
5.66	Access Authorization Program for Nuclear Power Plants	2	Not Applicable	This guidance governs site-specific physical security program activities that are the responsibility of the COL applicant or licensee.	Not Applicable
5.68	Protection Against Malevolent Use of Vehicles at Nuclear Power Plants	-	Partially Conforms	Although applicable to the COL applicant or licensee, 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 10 CFR 73.55 Requirements (SGI)	-	Not Applicable	This guidance is the responsibility of the COL applicant or licensee.	13.6 (via Security Technical Report)
5.70	Guidance for the Application of the Theft and Diversion Design-Basis Treat in the Design Development, and Implementation of a Physical Security Program that Meets CFR 73.45 and 73.46 (SGI)	-	Not Applicable	COL applicant or licensee responsibility.	Not Applicable
5.71	Cyber Security Programs for Nuclear Facilities	-	Partially Conforms	The portions of RG 5.71 that govern site-specific operational and programmatic activities (e.g., development and implementation of operational cyber security plans) apply to the COL applicant or licensee.	13.6 (via Security Technical Report)

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
5.73	Fatigue Management for Nuclear Power Plant Personnel	-	Not Applicable	This RG is not applicable to the NuScale design but may be used by a COL applicant or licensee to meet the fatigue management requirements of 10 CFR 26 Subpart I.	Not Applicable
5.74	Managing the Safety/Security Interface	-	Not Applicable	COL applicant or licensee responsibility.	Not Applicable
5.75	Training and Qualification of Security Personnel at Nuclear Power Reactor Facilities	-	Not Applicable	COL applicant or licensee responsibility.	Not Applicable
5.76	Physical Protection Programs at Nuclear Power Reactors	-	Not Applicable	This guidance governs site-specific physical protection program activities that are the responsibility of the COL applicant or licensee.	13.6 (via Security Technical Report)
5.77	Insider Mitigation Program (OUO-SRI)	-	Not Applicable	COL applicant or licensee responsibility.	Not Applicable
5.78	Physical Protection of Mixed Oxide Fuels in Nuclear Power Plants (SGI)	-	Not Applicable	The NuScale design does not use mixed oxide fuels.	Not Applicable
5.79	Protection of Safeguard Information	-	Conforms	NuScale protects Safeguards Information against unauthorized disclosure in accordance with RG 5.79.	Not Applicable
5.80	Pressure-Sensitive and Tamper-Indicating Device Seals for Material Control and Accounting of Special Nuclear Material	-	Not Applicable	This RG is not applicable to the NuScale design.	Not Applicable
5.81	Target Set Identification and Development for Nuclear Power Reactors (OUO-SRI)	-	Not Applicable	COL applicant or licensee responsibility.	Not Applicable
5.83	Cyber Security Event Notifications	-	Not Applicable	COL applicant or licensee responsibility.	Not Applicable
5.84	Fitness-For-Duty for New Nuclear Power Plant Construction Sites	-	Not Applicable	COL applicant or licensee responsibility.	Not Applicable
8.2	Administrative Practices in Radiation Surveys and Monitoring	1	Not Applicable	This guidance governs site-specific, programmatic activities related to radiation surveys and monitoring that are the responsibility of the COL applicant or licensee.	Not Applicable

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
8.4	Personnel Monitoring Device - Direct-Reading Pocket Dosimeters	1	Not Applicable	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 or licensee.	Not Applicable
8.7	Instructions for Recording and Reporting Occupational Radiation Dose Data	2	Not Applicable	This guidance governs site-specific programmatic activities related to recording and reporting dose data that are the responsibility of the COL applicant or licensee.	Not Applicable
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	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 10.4 11.2 11.4 11.5 12.1 12.3 12.5 14.3
8.9	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program	1	Not Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the COL applicant.	Not Applicable
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable	1-R	Not Applicable	These site-specific aspects are the responsibility of the COL applicant referencing the certified design.	Not Applicable
8.11	Applications of Bioassay for Uranium	-	Not Applicable	This guidance governs programmatic activities that apply to licensees for which uranium bioassay is required.	Not Applicable
8.13	Instruction Concerning Prenatal Radiation Exposure	3	Not Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the COL applicant.	Not Applicable
8.15	Acceptable Programs for Respiratory Protection	1	Not Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods 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	Comments	Section
8.18	Information Relevant to Ensuring that Radiation Exposures at Medical Institutions Will Be as Low as is Reasonably Achievable	2	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	Partially Conforms	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. Construction activities dose assessments are the responsibility of the COL applicant or licensee.	12.4
8.20	Applications of Bioassay for Radioiodine	2	Not Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the COL applicant or licensee.	Not Applicable
8.21	Health Physics Surveys for Byproduct Material at NRC Licensed Processing and Manufacturing Plants	1	Not Applicable	Applicable only to processing and manufacturing plants.	Not Applicable
8.22	Bioassay at Uranium Mills	1	Not Applicable	Applicable only to uranium mills.	Not Applicable
8.23	Radiation Safety Surveys at Medical Institutions	1	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	This guidance governs activities applicable only to facilities that process or fabricate fuel with uranium enriched with the U-235 isotope.	Not Applicable
8.25	Air Sampling in the Workplace	1	Not Applicable	This guidance governs site-specific, programmatic activities related to air sampling in the workplace that are the responsibility of the COL applicant or licensee.	Not Applicable
8.26	Applications of Bioassay for Fission and Activation Products	-	Not Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the COL applicant or licensee.	Not Applicable
8.27	Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants	-	Not Applicable	This guidance governs site-specific operational training programs, plans, and procedures that are the responsibility of the COL applicant or licensee.	Not Applicable

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
8.28	Audible-Alarm Dosimeters	-	Not Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the COL applicant or licensee.	Not Applicable
8.29	Instruction Concerning Risks from Occupational Radiation Exposure	1	Not Applicable	This guidance governs site-specific, programmatic training and instructional activities that are the responsibility of the COL applicant or licensee.	Not Applicable
8.30	Health Physics Surveys in Uranium Recovery Facilities	1	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	This guidance governs activities applicable only to uranium recovery facilities.	Not Applicable
8.32	Criteria for Establishing a Tritium Bioassay Program	-	Not 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
8.34	Monitoring Criteria and Methods to Calculate Occupational Radiation Doses	-	Not Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the COL applicant or licensee.	Not Applicable
8.35	Planned Special Exposure	1	Not Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the COL applicant or licensee.	Not Applicable
8.36	Radiation Dose to the Embryo/ Fetus	-	Not Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the COL applicant or licensee.	Not Applicable
8.37	ALARA Levels for Effluents from Materials Facilities	-	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	Implementation of this guidance is site-specific and is the responsibility of the COL applicant. However, NuScale considered this guidance to be applicable to the extent necessary to provide reasonable assurance that the COL applicant or licensee referencing the certified design can meet these requirements.	12.1 12.3 12.5 14.2

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

RG	Division Title	Rev.	Conformance Status	Comments	Section
8.39	Release of Patients Administered Radioactive Materials	-	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	This guidance governs dosimetry methods for determining effective dose equivalent for external radiation exposures. These methods are the responsibility of the COL applicant or licensee.	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	Comments	Section
SRP 1.0, Rev 2: Introduction and Interfaces	II.1	No Specific Acceptance Criteria	-	No Specific Acceptance Criteria.	Not Applicable
SRP 1.0, Rev 2: Introduction and Interfaces	II.2	SRP Acceptance Criteria Associated with Each Referenced SRP section	Conforms	None.	Ch 1
SRP 1.0, Rev 2: Introduction and Interfaces	II.3	Performance of New Safety Features and Design Qualification Testing Requirements	Conforms	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	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	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	II.3	COL Application Referencing a Certified Design	Not Applicable	This acceptance criterion is for COL applicants to meet the design parameters established in the DCA.	Not Applicable
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	This acceptance criterion is for COL applicants to meet the design parameters established in the DCA.	Not Applicable
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	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	NuScale provides site 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	NuScale provides site 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	Site-specific.	Not Applicable
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	Site-specific.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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	Site-specific.	Not Applicable
SRP 2.2.1-2.2.2, Rev 3: Identification of Potential Hazards in Site Vicinity	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.2.3, Rev 3: Evaluation of Potential Accidents	All	Event Probability and Design-Basis Event Analysis	Not Applicable	Site-specific.	Not Applicable
SRP 2.3.1, Rev 3: Regional Climatology	All	Various	Not Applicable	Site-specific	Not Applicable
SRP 2.3.1, Rev 3: Regional Climatology	III.4.b.1	Postulated Site Parameters	Conforms	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	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	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	None.	Table 2.0-1 2.3.1
SRP 2.3.2, Rev 3: Local Meteorology	II.1 thru II.4	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.3.2, Rev 3: Local Meteorology	III.4.b.i	Postulated Site Parameters	Conforms	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	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	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	None.	Table 2.0-1 2.3.2
SRP 2.3.3, Rev 3: Onsite Meteorological Measurements Program	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.3.4, Rev 3: Short-Term Atmospheric Dispersion Estimates for Accident Releases	All	Various	Not Applicable	Site-specific.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 2.3.4, Rev 3: Short-Term Atmospheric Dispersion Estimates for Accident Releases	III.6.b.1	Postulated Site Parameters	Conforms	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	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	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	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	None.	2.3.4
SRP 2.3.5, Rev 3: Long-Term Atmospheric Dispersion Estimates for Routine Releases	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.3.5, Rev 3: Long-Term Atmospheric Dispersion Estimates for Routine Releases	III.5.b.1	Postulated Site Parameters	Conforms	None.	2.3.5
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	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	None.	2.3.5

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 2.3.5, Rev 3: Long-Term Atmospheric Dispersion Estimates for Routine Releases	III.5.b.4	Basis for Site Parameters	Conforms	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	None.	2.3.5
SRP 2.4.1, Rev 3: Hydrologic Description	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.4.1, Rev 3: Hydrologic Description	III.7.B.i	Postulated Site Parameters	Conforms	None.	2.4.1
SRP 2.4.1, Rev 3: Hydrologic Description	III.7.B.ii	Site Parameters Included as Tier 1 Information	Conforms	None.	2.4.1
SRP 2.4.1, Rev 3: Hydrologic Description	III.7.B.iii	Site Parameters Summary Table	Conforms	None.	2.4.1
SRP 2.4.1, Rev 3: Hydrologic Description	III.7.B.iv	Basis for Site Parameters	Conforms	None.	2.4.1
SRP 2.4.2, Rev 4: Floods	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.4.2, Rev 4: Floods	III.11.B.i	Postulated Site Parameters	Conforms	None.	2.4.2
SRP 2.4.2, Rev 4: Floods	III.11.B.ii	Site Parameters Included as Tier 1 Information	Conforms	None.	2.4.2
SRP 2.4.2, Rev 4: Floods	III.11.B.iii	Site Parameters Summary Table	Conforms	None.	2.4.2
SRP 2.4.2, Rev 4: Floods	III.11.B.iv	Basis for Site Parameters	Conforms	None.	2.4.2
SRP 2.4.3, Rev 4: Probable Maximum Flood (PMF) on Streams and Rivers	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.4.3, Rev 4: Probable Maximum Flood (PMF) on Streams and Rivers	III.4.B.i	Postulated Site Parameters	Conforms	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	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	None.	2.4.3

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 2.4.3, Rev 4: Probable Maximum Flood (PMF) on Streams and Rivers	III.4.B.iv	Basis for Site Parameters	Conforms	None.	2.4.3
SRP 2.4.4, Rev 3: Potential Dam Failures	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.4.4, Rev 3: Potential Dam Failures	III.8.B.i	Postulated Site Parameters	Conforms	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	None.	2.4.4
SRP 2.4.4, Rev 3: Potential Dam Failures	III.8.B.iii	Site Parameters Summary Table	Conforms	None.	2.4.4
SRP 2.4.4, Rev 3: Potential Dam Failures	III.8.B.iv	Basis for Site Parameters	Conforms	None.	2.4.4
SRP 2.4.5, Rev 3: Probable Maximum Surge and Seiche Flooding	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.4.5, Rev 3: Probable Maximum Surge and Seiche Flooding	III.7.B.i	Postulated Site Parameters	Conforms	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	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	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	None.	2.4.5
SRP 2.4.6, Rev 3: Probable Maximum Tsunami Hazards	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.4.6, Rev 3: Probable Maximum Tsunami Hazards	III.9.B.i	Postulated Site Parameters	Conforms	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	None.	2.4.6
SRP 2.4.6, Rev 3: Probable Maximum Tsunami Hazards	III.9.B.iii	Site Parameters Summary Table	Conforms	None.	2.4.6

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 2.4.6, Rev 3: Probable Maximum Tsunami Hazards	III.9.B.iv	Basis for Site Parameters	Conforms	None.	2.4.6
SRP 2.4.7, Rev 3: Ice Effects	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.4.7, Rev 3: Ice Effects	III.6.B.i	Postulated Site Parameters	Conforms	None.	2.4.7
SRP 2.4.7, Rev 3: Ice Effects	III.6.B.ii	Site Parameters Included as Tier 1 Information	Conforms	None.	2.4.7
SRP 2.4.7, Rev 3: Ice Effects	III.6.B.iii	Site Parameters Summary Table	Conforms	None.	2.4.7
SRP 2.4.7, Rev 3: Ice Effects	III.6.B.iv	Basis for Site Parameters	Conforms	None.	2.4.7
SRP 2.4.8, Rev 3: Cooling Water Canals and Reservoirs	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.4.8, Rev 3: Cooling Water Canals and Reservoirs	III.5.B.1	Postulated Site Parameters	Conforms	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	None.	2.4.8
SRP 2.4.8, Rev 3: Cooling Water Canals and Reservoirs	III.5.B.3	Site Parameters Summary Table	Conforms	None.	2.4.8
SRP 2.4.8, Rev 3: Cooling Water Canals and Reservoirs	III.5.B.4	Basis for Site Parameters	Conforms	None.	2.4.8
SRP 2.4.9, Rev 3: Channel Diversions	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.4.9, Rev 3: Channel Diversions	III.8.B.i	Postulated Site Parameters	Conforms	None.	2.4.9
SRP 2.4.9, Rev 3: Channel Diversions	III.8.B.ii	Site Parameters Included as Tier 1 Information	Conforms	None.	2.4.9
SRP 2.4.9, Rev 3: Channel Diversions	III.8.B.iii	Site Parameters Summary Table	Conforms	None.	2.4.9
SRP 2.4.9, Rev 3: Channel Diversions	III.8.B.iv	Basis for Site Parameters	Conforms	None.	2.4.9
SRP 2.4.10, Rev 3: Flooding Protection Requirements	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.4.10, Rev 3: Flooding Protection Requirements	III.5.B.i	Postulated Site Parameters	Conforms	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	None.	2.4.10

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 2.4.10, Rev 3: Flooding Protection Requirements	III.5.B.iii	Site Parameters Summary Table	Conforms	None.	2.4.10
SRP 2.4.10, Rev 3: Flooding Protection Requirements	III.5.B.iv	Basis for Site Parameters	Conforms	None.	2.4.10
SRP 2.4.11, Rev 3: Low Water Considerations	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.4.11, Rev 3: Low Water Considerations	III.6.B.i	Postulated Site Parameters	Conforms	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	None.	2.4.11
SRP 2.4.11, Rev 3: Low Water Considerations	III.6.B.iii	Site Parameters Summary Table	Conforms	None.	2.4.11
SRP 2.4.11, Rev 3: Low Water Considerations	III.6.B.iv	Basis for Site Parameters	Conforms	None.	2.4.11
SRP 2.4.12, Rev 3: Groundwater	II.1 thru II.5	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.4.12, Rev 3: Groundwater	III.6.B.i	Postulated Site Parameters	Conforms	None.	2.4.12
SRP 2.4.12, Rev 3: Groundwater	III.6.B.ii	Site Parameters Included as Tier 1 Information	Conforms	None.	2.4.12
SRP 2.4.12, Rev 3: Groundwater	III.6.B.iii	Site Parameters Summary Table	Conforms	None.	2.4.12
SRP 2.4.12, Rev 3: Groundwater	III.6.B.iv	Basis for Site Parameters	Conforms	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	Site-specific.	Not Applicable
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	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	None.	2.4.13

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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	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	None.	2.4.13
SRP 2.4.14, Rev 3: Technical Specifications and Emergency Operation Requirements	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.4.14, Rev 3: Technical Specifications and Emergency Operation Requirements	III.5.B.i	Postulated Site Parameters	Conforms	None.	2.4.14
SRP 2.4.14, Rev 3: Technical Specifications and Emergency Operation Requirements	III.5.B.ii	Site Parameters Included as Tier 1 Information	Conforms	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	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	None.	2.4.14
SRP 2.5.1, Rev 4: Basic Geologic and Seismic Information	All	Regional and Site Geology	Not Applicable	Site-specific.	Not Applicable
SRP 2.5.2, Rev 4: Vibratory Ground Motion	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.5.2, Rev 4: Vibratory Ground Motion	III.2.a	Postulated Site Parameters	Conforms	None.	2.5.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 2.5.2, Rev 4: Vibratory Ground Motion	III.2.b	Site Parameters Included as Tier 1 Information	Conforms	None.	2.5
SRP 2.5.2, Rev 4: Vibratory Ground Motion	III.2.c	Site Parameters Summary Table	Conforms	None.	2.5.2
SRP 2.5.2, Rev 4: Vibratory Ground Motion	III.2.d	Basis for Site Parameters	Conforms	None.	2.5.2
SRP 2.5.3, Rev 4: Surface Faulting	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.5.3, Rev 4: Surface Faulting	III.2.a	Postulated Site Parameters	Conforms	None.	2.5.3
SRP 2.5.3, Rev 4: Surface Faulting	III.2.b	Site Parameters Included as Tier 1 Information	Conforms	None.	2.5.3
SRP 2.5.3, Rev 4: Surface Faulting	III.2.c	Site Parameters Summary Table	Conforms	None.	2.5
SRP 2.5.3, Rev 4: Surface Faulting	III.2.d	Basis for Site Parameters	Conforms	None.	2.5.3
SRP 2.5.4, Rev 4: Stability of Subsurface Materials and Foundations	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.5.4, Rev 4: Stability of Subsurface Materials and Foundations	III.2.A	Postulated Site Parameters	Conforms	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	None.	2.5.4
SRP 2.5.4, Rev 4: Stability of Subsurface Materials and Foundations	III.2.C	Site Parameters Summary Table	Conforms	None.	2.5
SRP 2.5.4, Rev 4: Stability of Subsurface Materials and Foundations	III.2.D	Basis for Site Parameters	Conforms	None.	2.5.4
SRP 2.5.5, Rev 4: Stability of Slopes	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.5.5, Rev 4: Stability of Slopes	III.2.A	Postulated Site Parameters	Conforms	None.	2.5.5

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 2.5.5, Rev 4: Stability of Slopes	III.2.B	Site Parameters Included as Tier 1 Information	Conforms	None.	2.5.5
SRP 2.5.5, Rev 4: Stability of Slopes	III.2.C	Site Parameters Summary Table	Conforms	None.	2.5
SRP 2.5.5, Rev 4: Stability of Slopes	III.2.D	Basis for Site Parameters	Conforms	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	Partially Conforms	This acceptance criterion is applicable except that SSC meeting Staff Regulatory Guidance C.1.i of Regulatory Guide 1.29 are designated Seismic Category II rather than Seismic Category I.	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	None.	3.2.2
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	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	The intent of Table A-1 is applicable but some of the specific language refers to SSC not part of the NuScale design. For example, the NuScale design does not include 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	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	None.	3.3.1
SRP 3.3.1, Rev. 3: Wind Loadings	II.3	Procedures for Transforming Wind Speed Into Equivalent Pressure	Conforms	None.	3.3.1
SRP 3.3.2: Rev. 3: Tornado Loads	II.1	Most Severe Tornado Wind and Associated Missiles	Partially Conforms	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	None.	3.3.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 3.3.2: Rev. 3: Tornado Loads	II.3	Procedures for Transforming Tornado Parameters Into Equivalent Loads on Structures	Conforms	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 SSC to Perform Safety Functions	Conforms	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	None.	3.4.1
SRP 3.4.1, Rev. 3: Internal Flood Protection for Onsite Equipment Failures	II.2	Compliance with GDC 4	Conforms	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	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	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	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	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	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	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	None.	3.5.1

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 3.5.1.3, Rev. 0: Turbine Missiles	II.1	Probability of Unacceptable Damage From Turbine Missiles	Not Applicable	Per DSRS 10.2.3, Section I and DSRS 3.5.1.3, Section I.1, plants that use barriers to protect essential SSCs specified in RG 1.115 do not have to rely on the turbine missile generation probabilities, including turbine rotor integrity.	Not Applicable
DSRS 3.5.1.3, Rev. 0: Turbine Missiles	II.2	Turbine Missile Generation	Not Applicable	Per DSRS 10.2.3, Section I and DSRS 3.5.1.3, Section I.1, plants that use barriers to protect essential SSCs specified in RG 1.115 do not have to rely on the turbine missile generation probabilities, including turbine rotor integrity.	Not Applicable
DSRS 3.5.1.3, Rev. 0: Turbine Missiles	II.3	Acceptably Low Missile Generation Probability	Not Applicable	Per DSRS 10.2.3, Section I and DSRS 3.5.1.3, Section I.1, plants that use barriers to protect essential SSCs specified in RG 1.115 do not have to rely on the turbine missile generation probabilities, including turbine rotor integrity.	Not Applicable
DSRS 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	Per DSRS 10.2.3, Section I and DSRS 3.5.1.3, Section I.1, plants that use barriers to protect essential SSCs specified in RG 1.115 do not have to rely on the turbine missile generation probabilities, including turbine rotor integrity.	Not Applicable
DSRS 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	Per DSRS 10.2.3, Section I and DSRS 3.5.1.3, Section I.1, plants that use barriers to protect essential SSCs specified in RG 1.115 do not have to rely on the turbine missile generation probabilities, including turbine rotor integrity.	Not Applicable
DSRS 3.5.1.3, Rev. 0: Turbine Missiles	II.6	Protective Barriers	Conforms	None.	3.5.1.3
SRP 3.5.1.4, Rev. 4: Missiles Generated by Extreme Winds	II.1	Design Basis Tornado-Generated Missile Spectrum	Conforms	The NuScale 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	None.	3.5.1.4

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 3.5.1.4, Rev. 4: Missiles Generated by Extreme Winds	II.3	Identifying Appropriate Design Basis Missiles Generated by Natural Phenomena	Conforms	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	The NuScale 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	The NuScale design assumes no proximity missiles.	Not Applicable
SRP 3.5.1.6, Rev 4: Aircraft Hazards	II.1 and II.2	Various	Not Applicable	The NuScale design assumes no aircraft hazard missiles.	Not Applicable
SRP 3.5.1.6, Rev 4: Aircraft Hazards	III.8.B.1	Postulated Site Parameters	Conforms	The NuScale 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	The NuScale 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	The NuScale 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	The NuScale design assumes no aircraft hazard missiles.	Table 2.0-1
SRP 3.5.2, Rev 3: Structures, Systems, and Components to be Protected From Externally-Generated Missiles	II (no number)	Capability of SSC to Withstand the Effects of Externally Generated Missiles	Conforms	None.	3.5.2
SRP 3.5.3, Rev. 3: Barrier Design Procedures	II.1.A	For Local Damage Prediction - Concrete	Departure	NuScale uses a finite element analysis for predicting penetration distance of turbine missiles in concrete, rather than the modified National Defense Research Council (NDRC) formula specified in Section II.1.A. In some locations perforation and scabbing is predicted. However, given the physical separation of the redundant safety-related equipment, there is no turbine missile that can prevent essential systems from performing their function.	3.5.3
SRP 3.5.3, Rev. 3: Barrier Design Procedures	II.1.B	For Local Damage Prediction - Steel	Conforms	None.	3.5.3

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 3.5.3, Rev. 3: Barrier Design Procedures	II.1.C	For Local Damage Prediction - Composite sections	Not Applicable	This acceptance criterion specifies provisions when using composite or multi-element barriers. NuScale does not use composite or multi-element barriers.	Not Applicable
SRP 3.5.3, Rev. 3: Barrier Design Procedures	II.2	For Overall Damage Prediction	Partially Conforms	This acceptance criterion is applicable except for reference to subtier ANSI/AISC N690-1994 with Supplement 2 (2004). NuScale uses the 2012 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	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	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	None.	3.6.1
SRP 3.6.1, Rev 3: Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	II.4	Design Features	Conforms	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	None.	3.6.1

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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	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	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	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	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.2	Dynamic Effects	Conforms	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	Conforms	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.4	Analyses of Pipe Break Dynamic Effects on Mechanical Components and Supports	Conforms	None.	3.6.2
SRP 3.6.3, Rev 1: Leak-Before-Break Evaluation Procedures	II.1	Compliance with GDC 4	Conforms	None.	3.6.3

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 3.6.3, Rev 1: Leak-Before-Break Evaluation Procedures	II.2	Low Probability of Pipe Rupture	Conforms	None.	3.6.3
DSRS 3.7.1, Rev. 0: Seismic Design Parameters	II.1	Design Ground Motion	Conforms	None.	3.7.1
DSRS 3.7.1, Rev. 0: Seismic Design Parameters	II.2	Percentage of Critical Damping Values	Conforms	None.	3.7.1
DSRS 3.7.1, Rev. 0: Seismic Design Parameters	II.3	Supporting Media for Seismic Category I Structures	Conforms	None.	3.7.1
DSRS 3.7.1, Rev. 0: Seismic Design Parameters	II.4	Review Considerations for DC and COL Applications	Conforms	None.	3.7.1
DSRS 3.7.2, Rev 0: Seismic System Analysis	II.1	Seismic Analysis Methods	Conforms	None.	3.7.2
DSRS 3.7.2, Rev 0: Seismic System Analysis	II.2	Natural Frequencies and Responses	Conforms	None.	3.7.2
DSRS 3.7.2, Rev 0: Seismic System Analysis	II.3	Procedures Used for Analytical Modeling	Conforms	None.	3.7.2
DSRS 3.7.2, Rev 0: Seismic System Analysis	II.4	Soil-Structure Interaction	Conforms	None.	3.7.2
DSRS 3.7.2, Rev 0: Seismic System Analysis	II.5	Development of In-Structure Response Spectra	Conforms	None.	3.7.2
DSRS 3.7.2, Rev 0: Seismic System Analysis	II.6	Three Components of Design Ground Motion	Conforms	None.	3.7.2
DSRS 3.7.2, Rev 0: Seismic System Analysis	II.7	Combination of Modal Responses	Conforms	None.	3.7.2
DSRS 3.7.2, Rev 0: Seismic System Analysis	II.8	Interaction of Non-Seismic Category I Structures with Seismic Category I SSCs	Conforms	None.	3.7.2
DSRS 3.7.2, Rev 0: Seismic System Analysis	II.9	Effects of Parameter Variations on Floor Response Spectra	Conforms	None.	3.7.2
DSRS 3.7.2, Rev 0: Seismic System Analysis	II.10	Use of Equivalent Vertical Static Factors	Conforms	None.	3.7.2
DSRS 3.7.2, Rev 0: Seismic System Analysis	II.11	Methods Used to Account for Torsional Effects	Conforms	None.	3.7.2
DSRS 3.7.2, Rev 0: Seismic System Analysis	II.12	Comparison of Responses	Not Applicable	NuScale does not perform both time history analysis and response spectrum analysis in its analysis of structures.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 3.7.2, Rev 0: Seismic System Analysis	II.13	Analysis Procedure for Damping	Conforms	None.	3.7.1 3.7.2
DSRS 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	None.	3.7.2
DSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.1	Seismic Analysis Methods	Conforms	None.	3.7.3
DSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.2	Determination of Number of Earthquake Cycles	Conforms	None.	3.7.3
DSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.3	Procedures Used for Analytical Modeling	Conforms	None.	3.7.3
DSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.4	Basis for Selection of Frequencies	Conforms	None.	3.7.3
DSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.5	Analysis Procedure for Damping	Conforms	None.	3.7.3
DSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.6	Three Components of Design Ground Motion	Conforms	None.	3.7.3
DSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.7	Combination of Modal Responses	Conforms	None.	3.7.3
DSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.8	Interaction of Non-Seismic Category I Subsystems with Seismic Category I SSCs	Conforms	None.	3.7.3
DSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.9	Multiply-Supported Equipment and Components with Distinct Inputs	Conforms	None.	3.7.3
DSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.10	Use of Equivalent Vertical Static Factors	Conforms	None.	3.7.3
DSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.11	Torsional Effects of Eccentric Masses	Conforms	None.	3.7.3
DSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.12	Seismic Category I Buried Piping, Conduits, and Tunnels	Conforms	None.	3.7.3
DSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.13	Methods for Seismic Analysis of Seismic Category I Concrete Dams	Not Applicable	The NuScale design does not use dams.	Not Applicable
DSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	II.14	Methods for Seismic Analysis of Above-Ground Tanks	Conforms	None.	3.7.3

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 3.7.4, Rev 2: Seismic Instrumentation	II.1	Comparison with RG 1.12	Partially Conforms	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	See RG 1.166 in Table 1.9-2.	Not Applicable
SRP 3.7.4, Rev 2: Seismic Instrumentation	II.3	Comparison with the requirements of 10 CFR 20.1101 (ALARA)	Not Applicable	Identified as an expectation for COL applicants.	Not Applicable
SRP 3.8.1, Rev 4: Concrete Containment	All	Various	Not Applicable	The NuScale design does not have a concrete containment.	Not Applicable
DSRS 3.8.2, Rev. 0: Steel Containment	II.1	Description of the Containment	Conforms	None.	3.8.2
DSRS 3.8.2, Rev. 0: Steel Containment	II.2	Applicable Codes, Standards, and Specifications	Conforms	None.	3.8.2
DSRS 3.8.2, Rev. 0: Steel Containment	II.3	Loads and Loading Combinations	Conforms	None.	3.8.2
DSRS 3.8.2, Rev. 0: Steel Containment	II.4	Design and Analysis Procedures	Conforms	None.	3.8.2
DSRS 3.8.2, Rev. 0: Steel Containment	II.5	Structural Acceptance Criteria	Conforms	None.	3.8.2
DSRS 3.8.2, Rev. 0: Steel Containment	II.6	Materials, Quality Control, and Special Construction Techniques	Conforms	None.	3.8.2
DSRS 3.8.2, Rev. 0: Steel Containment	II.7	Testing and Inservice Surveillance Requirements	Conforms	None.	3.8.2
SRP 3.8.3, Rev 4: Concrete and Steel Internal Structures of Steel or Concrete Containments	All	Various	Not Applicable	The NuScale containment does not have internal structures.	Not Applicable
DSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.1	Description of the Structures	Conforms	None.	3.8.4
DSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.2	Applicable Codes, Standards, and Specifications	Conforms	None.	3.8.4
DSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.3	Loads and Load Combinations	Conforms	None.	3.8.4
DSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.3.A	Concrete Structures	Conforms	None.	3.8.4

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.3.B	Steel Structures	Conforms	None.	3.8.4
DSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.4	Design and Analysis Procedures	Conforms	None.	3.8.4
DSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.5	Structural Acceptance Criteria	Conforms	None.	3.8.4
DSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.6	Materials, Quality Control, and Special Construction Techniques	Conforms	None.	3.8.4
DSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.7	Testing and Inservice Surveillance Requirements	Conforms	None.	3.8.4
DSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	II.8	Masonry Walls	Not Applicable	Masonry walls are not used in the NuScale design.	Not Applicable
DSRS 3.8.5, Rev. 0: Foundations	II.1	Description of the Foundation	Conforms	None.	3.8.5
DSRS 3.8.5, Rev. 0: Foundations	II.2	Applicable Codes, Standards, and Specifications	Conforms	None.	3.8.5
DSRS 3.8.5, Rev. 0: Foundations	II.3	Loads and Load Combinations	Conforms	None.	3.8.5
DSRS 3.8.5, Rev. 0: Foundations	II.4	Design and Analysis Procedures	Conforms	None.	3.8.5
DSRS 3.8.5, Rev. 0: Foundations	II.5	Structural Acceptance Criteria	Conforms	None.	3.8.5
DSRS 3.8.5, Rev. 0: Foundations	II.6	Materials, Quality Control, and Special Construction Techniques	Conforms	None.	3.8.5
DSRS 3.8.5, Rev. 0: Foundations	II.7	Testing and Inservice Surveillance Requirements	Conforms	None.	3.8.5
SRP 3.9.1, Rev 3: Special Topics for Mechanical Components	II.1	Specification of Transients	Conforms	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	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	Experimental Stress Analysis Method is not used.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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	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	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	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	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	None.	3.9.2
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	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	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	None.	3.9.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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	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	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	NRC Bulletin 88-11 applies 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) to the reactor pressure vessel: there is no pressurizer surge line within which thermal stratification (that is the issue of this bulletin) would occur.	Not Applicable
SRP 3.9.4, Rev 3: Control Rod Drive Systems	II.1	Adequacy of Descriptive Information	Conforms	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
SRP 3.9.4, Rev 3: Control Rod Drive Systems	II.2	Codes and Standards for Construction	Conforms	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	None.	3.9.4
SRP 3.9.4, Rev 3: Control Rod Drive Systems	II.4	Operability Assurance Program	Conforms	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	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	None.	3.9.3 3.9.5

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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	None.	3.9.2
SRP 3.9.5, Rev 3: Reactor Pressure Vessel Internals	II.4	Deformation Limits for Reactor Internals	Conforms	None.	3.9.5
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	The intent of subtier NUREG-0609 is applicable but the 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 have 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	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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	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	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.	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	Partially Conforms	Refer to Section 3.9.6.3.2 for valve testing and Section 3.9.6.6 for augmented valves testing program.	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	Not Applicable	The NuScale Power Plant does not have pumps or dynamic restraints that perform a specific function identified in the ASME OM Code Subsection ISTA-1100.	Not Applicable
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	Refer to Section 3.9.6.5 for relief requests and alternative authorizations to the code.	3.9.6
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	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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 3.9.7, Rev 0: Risk-Informed Inservice Testing	All	Various	Not Applicable	Development and implementation of a risk-informed, performance-based inservice testing program is the responsibility of COL applicants that reference the NuScale certified design 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	Development and implementation of a risk-informed, inservice inspection program for piping is the responsibility of COL applicants that reference the NuScale certified design, 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	None.	3.10
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	See RG 1.97 in Table 1.9-2	3.11
SRP 3.10, Rev 3: Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	II.3	Experience-Based Qualification	Not Applicable	Experience based seismic qualification is not used.	Not Applicable
SRP 3.10, Rev 3: Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	II.4	Records	Conforms	The NuScale design indicates that a Records program is required and includes a COL item 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	None.	3.10
SRP 3.10, Rev 3: Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	II.6	Documentation of Qualification Program	Conforms	None.	3.10

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 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	See RG 1.89 in Table 1.9-2.	3.11
DSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.2	Application of Clarification Related to IEEE Std. 323 Criteria	Conforms	None.	3.11
DSRS 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	Conforms	The portion of the guidance that endorses IEEE 317-1983 is applicable. See RG 1.63 entry in Table 1.9-2 with respect to the other aspects of RG 1.63.	3.11.2
DSRS 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	None.	3.11.2
DSRS 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	See RG 1.89 in Table 1.9-2.	3.11.2
DSRS 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	See RG 1.97 in Table 1.9-2.	3.11.2
DSRS 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	None.	3.11
DSRS 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	None.	3.11

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 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	None.	3.11
DSRS 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	None.	3.11.2
DSRS 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	None.	3.11.2
DSRS 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	See RG 1.158 in Table 1.9-2.	Not Applicable
DSRS 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	See RG 1.180 in Table 1.9-2.	3.11.2
DSRS 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	See RG 1.183 in Table 1.9-2.	3.11.2
DSRS 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	See RG 1.100 in Table 1.9-2.	3.11
DSRS 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	Lightning protection is not applicable to EQ because it is associated with an external/ natural event.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.17	Effects of Environmental Conditions for All Important to Safety Equipment	Conforms	None.	3.11
DSRS 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	None.	3.11
DSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.19	Qualification of Nonmetallic Parts	Conforms	None.	3.11
DSRS 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	None.	3.11
DSRS 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	None.	3.11
DSRS 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	The programs are described and maintained by the COL applicant.	Not Applicable
DSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.23	Operational Program Implementation	Not Applicable	This is a COL applicant item.	Not Applicable
DSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.24	Exposure of Organic Components on Engineered Safety Features Systems	Conforms	None.	3.11

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.25	Design and Procurement Specifications	Not Applicable	This is a COL applicant item.	Not Applicable
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	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	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	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	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	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	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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	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	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	None.	3.6
BTP 3-3, Rev 3: Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	B.2	Design Features	Conforms	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	None.	3.6
BTP 3-3, Rev 3: Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	B.4	Implementation	Conforms	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	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	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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 4.2, Rev 3: Fuel System Design	II.1	Design Bases	Conforms	None.	4.2.1
SRP 4.2, Rev 3: Fuel System Design	II.1.A	Fuel System Damage	Conforms	None.	4.2.1
SRP 4.2, Rev 3: Fuel System Design	II.1.B	Fuel Rod Failure	Conforms	None.	4.2.1 4.2.3
SRP 4.2, Rev 3: Fuel System Design	II.1.C	Fuel Coolability	Conforms	None.	4.2.1
SRP 4.2, Rev 3: Fuel System Design	II.2	Description and Design Drawings	Conforms	None.	4.2.2
SRP 4.2, Rev 3: Fuel System Design	II.3	Design Evaluation	Conforms	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	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	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	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	None.	4.3.1
SRP 4.3, Rev 0: Nuclear Design	II.2	Reactivity Coefficients	Conforms	None.	4.3.2
SRP 4.3, Rev 0: Nuclear Design	II.3	Control Rod Patterns and Reactivity Worth	Conforms	None.	4.3.2
SRP 4.3, Rev 0: Nuclear Design	II.4	Analytical Methods and Data	Conforms	None.	4.3.3
DSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.1	Fuel Design Limits, Core Design, and Thermal Margin	Conforms	None.	4.4.1 4.4.2
DSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.2	Subchannel Hydraulic Analysis Codes	Conforms	None.	4.4.4
DSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.3	Core Oscillations and Thermal-Hydraulic Instabilities	Conforms	None.	4.4.7

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.4	RPV Fluid Flow Calculations	Conforms	None.	4.4.4
DSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.5	Technical Specifications	Conforms	None.	4.4.3 4.4.6 16.1
DSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.6	Preoperational and Initial Test Programs	Conforms	None.	4.4.5
DSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.7	Loose Parts Monitoring System	Departure	Low flow in primary systems precludes damage from loose parts and the need for loose parts monitoring system.	4.4.6
DSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.8	Critical Heat Flux Calculations and Process Monitoring	Conforms	None.	4.4.2 4.4.4 4.4.6
DSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.9	Instrumentation and Procedures for Detection and Recovery from Inadequate Core Cooling	Conforms	None.	4.4.6
DSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.10	Core Stability Performance During Anticipated Transient without Scram Event	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.	Not Applicable
SRP 4.5.1, Rev 3: Control Rod Drive Structural Materials	II.1	Materials Specifications	Conforms	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	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	None.	4.5.1
SRP 4.5.1, Rev 3: Control Rod Drive Structural Materials	II.4	Cleaning and Cleanliness Control	Conforms	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	None.	4.5.2
SRP 4.5.2, Rev 3: Reactor Internal and Core Support Structure Materials	II.2	Controls on Welding	Conforms	None.	4.5.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 4.5.2, Rev 3: Reactor Internal and Core Support Structure Materials	II.3	Nondestructive Examination	Conforms	None.	4.5.2
SRP 4.5.2, Rev 3:Reactor Internal and Core Support Structure Materials	II.4	Austenitic Stainless Steels	Conforms	None.	4.5.2
SRP 4.5.2, Rev 3: Reactor Internal and Core Support Structure Materials	II.5	Other Materials	Conforms	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	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	None.	4.6.2
SRP 4.6, Rev 2: Functional Design of Control Rod Drive System	II.3	Single Malfunction - GDC 25	Conforms	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	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 and volume control system, which is not safety-related.	4.6.2
SRP 4.6, Rev 2: Functional Design of Control Rod Drive System	II.5	Combined Capability - GDC 27	Departure	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	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	None.	4.6.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 4.6, Rev 2: Functional Design of Control Rod Drive System	II.8	BWR Alternate Rod Injection System	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	This BTP is applicable only to PWR designs that use the Constant Axial Offset Control operating scheme. NuScale does not 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	See RG 1.26 in Table 1.9-2.	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	See RG 1.26 in Table 1.9-2.	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	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	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	None.	5.2.2
SRP 5.2.2, Rev 3: Overpressure Protection	II.2	Design Requirements for BWRs Operating at Power	Not Applicable	This guidance is applicable only to BWR plants.	Not Applicable
SRP 5.2.2, Rev 3: Overpressure Protection	II.3	Design Requirements for PWRs Operating at Power	Partially Conforms	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	None.	5.2.2
SRP 5.2.2, Rev 3: Overpressure Protection	II.5	Testing and Inspections	Conforms	None.	5.2.2
SRP 5.2.2, Rev 3: Overpressure Protection	II.6	Technical Specifications	Partially Conforms	Certain subtier guidance documents referenced in this acceptance criterion are not applicable or only partially applicable.	5.2.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 5.2.2, Rev 3: Overpressure Protection	II.7	TMI Action Plan Requirements	Conforms	None.	5.2.2
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.1	Material Specifications	Partially Conforms	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	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	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	None.	5.2.3
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.3.B	Control of Ferritic Steel Welding	Conforms	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	None.	5.2.3
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.4	Fabrication and Processing of Austenitic Stainless Steel	Partially Conforms	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	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	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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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	This acceptance criterion is applicable only to LWRs that use nonmetallic thermal insulation on reactor coolant pressure boundary components. NuScale does not use nonmetallic thermal insulation on reactor coolant pressure boundary components.	Not Applicable
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.4.D	Control of Welding of Austenitic Stainless Steels	Partially Conforms	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	None.	5.2.3
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.4.G	Operational Programs	Not Applicable	This acceptance criterion governs plant-specific programmatic information that is the responsibility of the COL applicant.	Not Applicable
DSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.1	System Boundary Subject to Inspection	Conforms	None.	5.2.4
DSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.2	Accessibility	Conforms	None.	5.2.4
DSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.3	Examination Categories and Methods	Conforms	None.	5.2.4
DSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.4	Inspection Intervals	Conforms	None.	5.2.4
DSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.5	Evaluation of Examination Results	Conforms	None.	5.2.4

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.6	System Pressure Tests	Conforms	None.	5.2.4
DSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.7	Code Exemptions	Conforms	None.	5.2.4
DSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.8	Relief Requests	Conforms	None.	5.2.4
DSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.9	Code Cases	Conforms	None.	5.2.4
DSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.10	Augmented ISI to Protect Against Postulated Piping Failures	Conforms	None.	5.2.4
DSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.11	Other Inspection Programs	Partially Conforms	Although a boric acid control program has not been established, a brief description of the program is provided in the DCA.	5.2.4
DSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.12	Operational Programs	Not Applicable	This acceptance criterion governs plant-specific programmatic information that is the responsibility of the COL applicant.	Not Applicable
DSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.13	ITAAC	Partially Conforms	A portion of this acceptance criterion is applicable only to COL applicants.	5.2.4
DSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.14	Risk Informed ISI Program	Not 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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 5.2.5, Rev 0: Reactor Coolant Pressure Boundary Leakage Detection	II.1	Criteria to Meet GDC 2	Conforms	None.	5.2.5
DSRS 5.2.5, Rev 0: Reactor Coolant Pressure Boundary Leakage Detection	II.2	Criteria to Meet GDC 14	Conforms	None.	5.2.5
DSRS 5.2.5, Rev 0: Reactor Coolant Pressure Boundary Leakage Detection	II.3	Criteria to Meet GDC 30	Conforms	None.	5.2.5
DSRS 5.3.1, Rev 0: Reactor Vessel Materials	II.1	Materials	Conforms	None.	5.3.1
DSRS 5.3.1, Rev 0: Reactor Vessel Materials	II.2	Special Processes Used for Manufacture and Fabrication of Components	Conforms	None.	5.3.1
DSRS 5.3.1, Rev 0: Reactor Vessel Materials	II.3	Special Methods for Nondestructive Examination	Conforms	None.	5.3.1
DSRS 5.3.1, Rev 0: Reactor Vessel Materials	II.4	Special Controls and Special Processes Used for Ferritic Steels and Austenitic Stainless Steels	Conforms	None.	5.3.1
DSRS 5.3.1, Rev 0: Reactor Vessel Materials	II.5	Fracture Toughness	Conforms	None.	5.3.1
DSRS 5.3.1, Rev 0: Reactor Vessel Materials	II.6	Material Surveillance	Conforms	None.	5.3.1
DSRS 5.3.1, Rev 0: Reactor Vessel Materials	II.7	Reactor Vessel Fasteners	Conforms	None.	5.3.1
DSRS 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	None.	5.3.2
DSRS 5.3.2, Rev 0: Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	II.1.B	Pressure-Temperature Requirements	Conforms	None.	5.3.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 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	None.	5.3.2
DSRS 5.3.2, Rev 0: Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	II.2.B	Upper-Shelf Energy Requirements	Conforms	None.	5.3.1 5.3.2
DSRS 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	None.	5.3.2
DSRS 5.3.2, Rev 0: Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	II.3.B	Pressurized Thermal Shock Requirements	Conforms	None.	5.3.2
DSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.1	Design	Conforms	None.	5.3.3
DSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.2	Materials of Construction	Conforms	None.	5.3.3
DSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.3	Fabrication Methods	Conforms	None.	5.3.3
DSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.4	Inspection Requirements	Conforms	None.	5.3.3
DSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.5	Shipment and Installation	Conforms	None.	5.3.3
DSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.6	Operating Conditions	Conforms	None.	5.3.3
DSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.7	Inservice Surveillance	Conforms	Inservice surveillance of the reactor vessel is described in the DCD. However, the COL applicant develops and implements the reactor vessel surveillance program.	5.3.3
DSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.8	Operational Programs	Not 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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.9	10 CFR 52.47(b)(1) compliance	Not Applicable	This requirement applies to plant-specific verification and is the responsibility of the COL applicant.	Not Applicable
SRP 5.4.1.1, Rev 3: Pump Flywheel Integrity (PWR)	All	Various	Not Applicable	This SRP section and its acceptance criteria (II.1 through II.7) apply only to PWR designs that use reactor coolant pumps. The NuScale reactor 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
DSRS 5.4.2.1, Rev 0: Steam Generator Materials	II.1	Selection, Processing, Testing, and Inspection of Materials	Conforms	None.	5.4.1
DSRS 5.4.2.1, Rev 0: Steam Generator Materials	II.2	Steam Generator Design	Conforms	None.	5.4.1
DSRS 5.4.2.1, Rev 0: Steam Generator Materials	II.3	Fabrication and Processing of Ferritic Materials	Conforms	None.	5.4.1
DSRS 5.4.2.1, Rev 0: Steam Generator Materials	II.4	Fabrication and Processing of Austenitic Stainless Steel	Conforms	None.	5.2 5.4.1
DSRS 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	None.	5.4.1
DSRS 5.4.2.1, Rev 0: Steam Generator Materials	II.6	Provisions for Accessing the Secondary Side of the Steam Generator	Conforms	None.	5.4.1
DSRS 5.4.2.2, Rev 0: Steam Generator Program	II.1	Steam Generator Tube Susceptibility to Degradation	Conforms	None.	5.4
DSRS 5.4.2.2, Rev 0: Steam Generator Program	II.2	Steam Generator Monitoring Program Elements	Partially Conforms	A portion of this acceptance criterion is applicable to COL applicants referencing a certified design.	5.4.1
DSRS 5.4.2.2, Rev 0: Steam Generator Program	II.3	Steam Generator Program Elements in Technical Specifications	Partially Conforms	Certain subtier guidance documents referenced in this acceptance criterion are only partially applicable.	5.4.1
DSRS 5.4.2.2, Rev 0: Steam Generator Program	II.4	Steam Generator Tube Repair Criteria	Conforms	None.	5.4.1

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 5.4.2.2, Rev 0: Steam Generator Program	II.5	Steam Generator Tube Repair Methods	Conforms	None.	5.4.1
DSRS 5.4.2.2, Rev 0: Steam Generator Program	II.6	Steam Generator Tube Preservice Inspection	Conforms	None.	5.4.1
DSRS 5.4.2.2, Rev 0: Steam Generator Program	II.7	Periodic Tube Inspection and Testing in Certified Design Technical Specifications	Partially Conforms		5.4.1
DSRS 5.4.2.2, Rev 0: Steam Generator Program	II.8	Operational Programs	Partially Conforms	This acceptance criterion governs plant-specific programmatic activities that are the responsibility of the COL applicant referencing a certified design.	5.4.1
DSRS 5.4.2.2, Rev 0: Steam Generator Program	II.9	ITAAC	Partially Conforms	A portion of this acceptance criterion is applicable only to COL applicants.	5.4.1
SRP 5.4.6, Rev 4: Reactor Core Isolation Cooling System (BWR)	All	Various	Not Applicable	This SRP section and its acceptance criteria (II.1 through II.10) apply only to BWRs.	Not Applicable
DSRS 5.4.7, Rev 0: Decay Heat Removal (DHR) System Responsibilities	II.1 thru II.3	Various	Conforms	None.	5.4.3
DSRS 5.4.7, Rev 0: Decay Heat Removal (DHR) System Responsibilities	II.4	GDC 5	Conforms	None.	5.4.3
DSRS 5.4.7, Rev 0: Decay Heat Removal (DHR) System Responsibilities	II.5	GDC 14	Not Applicable	The DHRS is connected to the secondary system and does not directly interface with the RCPB.	Not Applicable
DSRS 5.4.7, Rev 0: Decay Heat Removal (DHR) System Responsibilities	II.6	GDC 19	Departure	The NuScale design supports an exemption from GDC 19. As described in Section 3.1.2, the design complies with a NuScale-specific principal design criterion (PDC) in lieu of this GDC.	5.4.3
DSRS 5.4.7, Rev 0: Decay Heat Removal (DHR) System Responsibilities	II.7	GDC 34	Departure	The NuScale design supports an exemption from the power provisions of GDC 34. As described in Section 3.1.4, the design complies with a NuScale-specific principal design criterion in lieu of this GDC.	5.4.3

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 5.4.7, Rev 0: Decay Heat Removal (DHR) System Responsibilities	II.8	GDC 54	Partially Conforms	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
DSRS 5.4.7, Rev 0: Decay Heat Removal (DHR) System Responsibilities	II.9	DHRS Interface with other systems	Conforms	None.	5.4.3
SRP 5.4.8, Rev 3: Reactor Water Cleanup System (BWR)	All	Various	Not Applicable	This SRP section and its acceptance criteria (II.1 through II.4) apply only to BWRs.	Not Applicable
SRP 5.4.11, Rev 4: Pressurizer Relief Tank	All	Various	Not Applicable	This SRP section and its acceptance criteria (II.1 and II.2) apply only to PWRs that use a pressurizer relief tank. A pressurizer relief tank is not used in the NuScale design. Fluid relieved through the reactor coolant system overpressure protection system is routed directly to the containment vessel.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 5.4.12, Rev 1: Reactor Coolant System High Point Vents	All	Various	Departure	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. The NuScale design supports an exemption from the requirements of 10 CFR 50.46a related to reactor coolant system high point venting, as well as the substantively similar requirements of 10 CFR 50.34(f)(2)(vi).	Not Applicable
SRP 5.4.13, (March 2007): Isolation Condenser System (BWR)	All	Various	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	None.	5.4.1 10.3.5
BTP 5-1, Rev 3: Monitoring of Secondary Side Water Chemistry in PWR Steam Generators	B.2	Sampling Schedule for Critical Parameters	Partially Conforms	A portion of this acceptance criterion governs information that is site-specific and 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	A portion of this acceptance criterion governs information that is site-specific and 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	This acceptance criterion 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	None.	5.2.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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	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	None.	5.2.2
BTP 5-2, Rev 3: Overpressurization Protection of Pressurized-Water Reactors While Operating at Low Temperatures	B.4	System Instrumentation and Controls Design	Conforms	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	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	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	None.	5.2.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
BTP 5-2, Rev 3: Overpressurization Protection of Pressurized-Water Reactors While Operating at Low Temperatures	B.8	Backup Electrical Power Source	Partially Conforms	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 - applies 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	None.	5.2.2
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	The intent of this acceptance criterion is applicable but the criterion 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 low-pressure 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	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_{NDT} for Vessel Materials	Partially Conforms	Portions of this acceptance criterion apply only to older plants for which fracture toughness testing on vessel material did not include all tests necessary to determine RTNDT. The rest of this guidance applies to the NuScale design.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	1.2	Estimation of Charpy V-Notch Upper Shelf Energies	Partially Conforms	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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
BTP 5-3, Rev 2: Fracture Toughness Requirements	1.3	Reporting Requirements	Conforms	None.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	2	Operating Limitations for Fracture Toughness	Conforms	None.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	2.1	Pressure-Temperature Operating Limitations	Conforms	None.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	2.2	Recommended Bases for Operating Limitations	Conforms	None.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	2.3	Reporting Requirements	Conforms	None.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	3	Inservice Surveillance of Fracture Toughness	Partially Conforms	This acceptance criterion applies 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	None.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	3.2	SAR Criteria	Conforms	None.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	3.3	Surveillance Test Procedures	Conforms	None.	5.3
BTP 5-3, Rev 2: Fracture Toughness Requirements	3.4	Reporting Criteria	Not Applicable	This acceptance criterion governs plant-specific reporting criteria that are the responsibility of the COL applicant that references the NuScale certified design.	Not Applicable
BTP 5-3, Rev 2: Fracture Toughness Requirements	3.5	Technical Specification Changes	Not Applicable	This acceptance criterion governs plant-specific activities that are the responsibility of the COL applicant that references the NuScale certified design.	Not Applicable
BTP 5-3, Rev 2: Fracture Toughness Requirements	4.1	Pressurized Thermal Shock Requirements	Conforms	None.	5.3
DSRS BTP 5-4, Rev 0: Design Requirements of the Residual Heat Removal System	B.1	Functional Requirements	Conforms	None.	5.4.3
DSRS BTP 5-4, Rev 0: Design Requirements of the Residual Heat Removal System	B.2	Pressure Relief Requirements	Conforms	None.	5.4.3

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS BTP 5-4, Rev 0: Design Requirements of the Residual Heat Removal System	B.3	Test Requirements	Conforms	None.	5.4.3
DSRS BTP 5-4, Rev 0: Design Requirements of the Residual Heat Removal System	B.4	Operational Procedures	Partially Conforms	The procedures governed by this acceptance criterion are site-specific and are the responsibility of the COL applicant.	5.4.3
DSRS BTP 5-4, Rev 0: Design Requirements of the Residual Heat Removal System	B.5	Implementation	Conforms	None.	5.4.3
SRP 6.1.1, Rev 2: Engineered Safety Features Materials	II.1	Materials and Fabrication	Conforms	None.	6.1.1
SRP 6.1.1, Rev 2: Engineered Safety Features Materials	II.1.A	Austenitic Stainless Steels	Conforms	None.	6.1.1
SRP 6.1.1, Rev 2: Engineered Safety Features Materials	II.1.B	Ferritic Steel Welding	Conforms	None.	6.1.1
SRP 6.1.1, Rev 2: Engineered Safety Features Materials	II.2	Composition and Compatibility of ESF Systems Fluids	Conforms	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	RG 1.37 has been withdrawn by the NRC.	6.1.1
SRP 6.1.1, Rev 2: Engineered Safety Features Materials	II.4	Thermal Insulation	Conforms	None.	6.1.1
SRP 6.1.2, Rev 3: Protective Coating Systems (Paints) - Organic Materials	All	Various	Not Applicable	This SRP section is applicable only to the use of protective coatings on surfaces inside the containment. The NuScale Power Module design does not use protective coatings inside the containment vessel.	Not Applicable
DSRS 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.	See SRP 2.4.6, 2.4.10, 2.4.12, 3.9.3, 19.0, and DSRS 3.8.2.	2.4 3.8.2 3.9.3 6.2.1 19.2
DSRS 6.2.1.1.A, Rev 0: Containment	II.1	Design Margin for Containment Design Pressure	Conform	The peak containment pressure for the limiting event is less than the design pressure.	6.2.1

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 6.2.1.1.A, Rev 0: Containment	II.2	Reducing Containment Pressure Following Postulated Design Basis Accident	Conforms	None.	6.2.1
DSRS 6.2.1.1.A, Rev 0: Containment	II.3	Containment Heat Removal Capability and Design Margin - LOCA Assumptions	Conforms	None.	6.2.1 6.2.2
DSRS 6.2.1.1.A, Rev 0: Containment	II.4	Containment Heat Removal Capability and Design Margin - Containment Response Analysis Assumptions	Conforms	None.	6.2.1 6.2.2
DSRS 6.2.1.1.A, Rev 0: Containment	II.5	Protection of Containment from External Pressure Conditions	Conforms	None.	6.2.1
DSRS 6.2.1.1.A, Rev 0: Containment	II.6	Containment Monitoring Instrumentation	Conforms	None.	6.2.1
DSRS 6.2.1.1.A, Rev 0: Containment	II.7	Design of Containment Internal Structures and System Components	Conforms	None.	6.2.1
DSRS 6.2.1.1.A, Rev 0: Containment	II.8	Evaluation of Accident Involving Generated Hydrogen	Conforms	None.	6.2.1
DSRS 6.2.1.1.A, Rev 0: Containment	II.9	Evaluation of an Accident on other Modules	Conforms	None.	6.2.1
SRP 6.2.1.1.B, Draft Rev 3: Ice Condenser Containments	All	Various	Not Applicable	The NuScale 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	This SRP section and its acceptance criteria apply 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	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 have subcompartments housing high-energy piping as defined in this guidance (or internal compartments as referred to in GDC 50).	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 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	Departure	The energy from metal-water reactions is not included. The NuScale design supports an exemption from selected portions of 10 CFR 50, Appendix K.	6.2.1
DSRS 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	None.	6.2.1
DSRS 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	This acceptance criterion is applicable only to COL applicants.	Not Applicable
DSRS 6.2.1.4, Rev 0: Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	II.1	Sources of Energy	Conforms	None.	6.2.1
DSRS 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	None.	6.2.1
DSRS 6.2.1.4, Rev 0: Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	II.3	Single-Failure Analyses	Conforms	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	This SRP section and its acceptance criteria are applicable only to PWRs for which a postulated LOCA results in core uncovery. For the NuScale reactor design, a LOCA does not result in core uncovery.	Not Applicable
DSRS 6.2.2, Rev 0: Containment Heat Removal Systems	II.1	GDC 5, Sharing of Structures, Systems, and Components	Conforms	None.	6.2.2 9.2.5

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 6.2.2, Rev 0: Containment Heat Removal Systems	II.2	GDC 38, Containment Heat Removal	Departure	The NuScale design supports an exemption from the power provisions of GDC 38. As described in Section 3.1.4, the design complies with a NuScale-specific principal design criterion in lieu of this GDC.	6.2.2
DSRS 6.2.2, Rev 0: Containment Heat Removal Systems	II.3	GDC 39, Inspection of Containment Heat Removal System	Conforms	None.	6.2.2
DSRS 6.2.2, Rev 0: Containment Heat Removal Systems	II.4	GDC 40, Testing of Containment Heat Removal System	Departure	The NuScale design does not conform to GDC 40 and the design supports an exemption.	3.1.4 6.2.2
DSRS 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	None.	6.2.2
DSRS 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	None.	6.2.2
SRP 6.2.3, Rev 3: Secondary Containment Functional Design	All	Various	Not Applicable	This SRP section and its acceptance criteria (II.1 through II.4) apply only to LWR designs that incorporate primary and secondary containment. The NuScale containment vessel design does not include a secondary containment.	Not Applicable
DSRS 6.2.4, Rev 0: Containment Isolation System	II.1	Instrument Line Isolation	Conforms	No instrumentation process lines penetrate containment.	6.2.4
DSRS 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	None.	6.2.4
DSRS 6.2.4, Rev 0: Containment Isolation System	II.3	Isolation of and Leak Detection in Lines in Systems Needed for Safe Shutdown	Conforms	None.	6.2.4
DSRS 6.2.4, Rev 0: Containment Isolation System	II.4	Containment Isolation Valve Requirements	Conforms	None.	6.2.4

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 6.2.4, Rev 0: Containment Isolation System	II.5	Containment Isolation Valve Requirements for Engineered Safety Feature (or Related) Systems	Conforms	None.	6.2.4
DSRS 6.2.4, Rev 0: Containment Isolation System	II.6	Use of Sealed-Closed Barriers in Place of Automatic Isolation Valves	Conforms	None.	6.2.4
DSRS 6.2.4, Rev 0: Containment Isolation System	II.7	Use of Relief Valves as Isolation Valves	Not Applicable	Relief valves are not used as CIVs.	Not Applicable
DSRS 6.2.4, Rev 0: Containment Isolation System	II.8	Classification of Essential or NonEssential Systems	Conforms	None.	6.2.4
DSRS 6.2.4, Rev 0: Containment Isolation System	II.9	Location of Isolation Valves Outside Containment	Conforms	None.	6.2.4
DSRS 6.2.4, Rev 0: Containment Isolation System	II.10	Loss of Power to Automatic Isolation Valves	Conforms	None.	6.2.4
DSRS 6.2.4, Rev 0: Containment Isolation System	II.11	Isolation Reliability	Conforms	None.	6.2.4
DSRS 6.2.4, Rev 0: Containment Isolation System	II.12	Parameter Diversity for Initiation of Containment Isolation	Conforms	None.	6.2.4

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 6.2.4, Rev 0: Containment Isolation System	II.13	Radiation Monitors for Initiation of Containment Isolation on Open Paths to the Environs	Departure	The containment evacuation system has the potential for an open path from containment to the environs but is isolated upon a high containment vessel pressure signal, and a low-low pressurizer level signal. Any in-containment event resulting in core damage or degradation also results in containment isolation on low low pressurizer level and high containment pressure. Any event leading to core damage or degradation, results in containment isolation on low low pressurizer level. These features provide an alternative, reliable means to prevent radiological release from the CES to the environs, consistent with the intent of this Acceptance Criterion. The NuScale design supports an exemption from 50.34(f)(2)(xiv).	6.2.4
DSRS 6.2.4, Rev 0: Containment Isolation System	II.14	Isolation Valve Closure Times	Conforms	None.	6.2.4
DSRS 6.2.4, Rev 0: Containment Isolation System	II.15	Use of Closed System Inside Containment	Conforms	None.	6.2.4
DSRS 6.2.4, Rev 0: Containment Isolation System	II.16	Specific Design Criteria for Containment Isolation Components	Conforms	None.	6.2.4
DSRS 6.2.4, Rev 0: Containment Isolation System	II.17	Provisions to Allow Control Room Operator Actions	Conforms	None.	6.2.4
DSRS 6.2.4, Rev 0: Containment Isolation System	II.18	Operability and Leakage Rate Testing	Conforms	None.	6.2.4
DSRS 6.2.4, Rev 0: Containment Isolation System	II.19	Reopening of Containment Isolation Valves	Conforms	None.	6.2.4

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 6.2.4, Rev 0: Containment Isolation System	II.20	Station Blackout	Conforms	None.	6.2.4
DSRS 6.2.4, Rev 0: Containment Isolation System	II.21	Source Term in Radiological Calculations	Conforms	None.	6.2.4
DSRS 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	The NuScale design combustible gas control system does not use combustible gas control systems. Systems to control hydrogen concentrations within containment are not required because combustion has no impact on CNV integrity. The NuScale design supports an exemption to 10 CFR 50.44(c)(2) as described in DCA Part 7, section 2. The NuScale design includes hydrogen and oxygen monitoring equipment that is capable of continuously measuring the concentration of hydrogen and oxygen in the containment atmosphere following a significant beyond design-basis accident for accident management and emergency planning.	6.2.5
DSRS 6.2.5, Rev 0: Combustible Gas Control in Containment	II.2	Equipment Survivability and Containment Structural Integrity	Conforms	None.	6.2.5
DSRS 6.2.5, Rev 0: Combustible Gas Control in Containment	II.3	Ensuring a Mixed Atmosphere	Conforms	None.	6.2.5
DSRS 6.2.5, Rev 0: Combustible Gas Control in Containment	II.4	Design Requirements of GDC 41	Departure	The NuScale design supports an exemption from the power provisions of GDC 41. As described in Section 3.1.4, the design complies with a NuScale-specific principal design criterion in lieu of this GDC.	6.2.5

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 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	For GDC 42 and 43, 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. See acceptance criterion II.4 above for GDC 41 compliance.	Not Applicable
DSRS 6.2.5, Rev 0: Combustible Gas Control in Containment	II.6	Containment Structural Integrity Analysis	Conforms	None.	6.2.5
DSRS 6.2.6, Rev 0: Containment Leakage Testing	All	Various	Departure	The NuScale design supports an exemption from the containment leakage rate testing at design pressure requirements of GDC 52 and Type A test requirements of 10 CFR 50 Appendix J.	6.2.6
SRP 6.2.7, Rev 1: Fracture Prevention of Containment Pressure Boundary	All	Various	Conforms	None.	6.2.7
DSRS 6.3, Rev 0: Emergency Core Cooling System	II.1	ECCS Acceptance Criteria of 10 CFR 50.46	Conforms	None.	6.3.1 6.3.3
DSRS 6.3, Rev 0: Emergency Core Cooling System	II.2	Single-Failure Consideration	Conforms	None.	6.3.1
DSRS 6.3, Rev 0: Emergency Core Cooling System	II.3	Inservice Inspection and Operability Testing	Departure	None.	6.3.2
DSRS 6.3, Rev 0: Emergency Core Cooling System	II.4	Combined Reactivity Control System Capability and Actuation Provisions	Departure	The guidance in this acceptance criterion related to actuation signals is applicable to ECCS actuation. For the requirements of GDC 27, the NuScale ECCS does not perform a poison addition safety function nor does it provide a makeup function. The NuScale design supports an exemption to GDC 27. As described in Section 3.1.4, the design complies with a NuScale-specific principal design criterion in lieu of this GDC.	6.3.1

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 6.3, Rev 0: Emergency Core Cooling System	II.5	Water Hammer	Conforms	None.	6.3.1
DSRS 6.3, Rev 0: Emergency Core Cooling System	II.6	Design of Non-Safety-Related Portions of ECCS	Conforms	None.	6.3.1
DSRS 6.3, Rev 0: Emergency Core Cooling System	II.7	ECCS Interfaces and Shared Systems	Conforms	None.	6.3.1
DSRS 6.3, Rev 0: Emergency Core Cooling System	II.8	Long Term Cooling	Conforms	None.	6.3.1
DSRS 6.3, Rev 0: Emergency Core Cooling System	II.9	ECCS Outage Times and Reports on Unavailability	Conforms	None.	6.3.2
DSRS 6.3, Rev 0: Emergency Core Cooling System	II.10	Programmatic Requirements	Conforms	None.	6.3.1
SRP 6.4, Rev 3: Control Room Habitability System	II.1	Control Room Emergency Zone	Conforms	None.	6.4
SRP 6.4, Rev 3: Control Room Habitability System	II.2	Ventilation System Criteria	Conforms	None.	6.4
SRP 6.4, Rev 3: Control Room Habitability System	II.3	Pressurization Systems	Conforms	None.	6.4
SRP 6.4, Rev 3: Control Room Habitability System	II.4	Emergency Standby Atmosphere Filtration System	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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 6.4, Rev 3: Control Room Habitability System	II.5	Relative Location of Source and Control Room	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
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	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	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	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 6.5.1-6, Design, Testing, and Maintenance of ESF Atmosphere Cleanup System Air Filtration and Adsorption Units	Not Applicable	The NuScale Power Plant design does not use engineered safety feature (ESF) filter systems or ESF ventilation systems to mitigate the consequences of a design basis accident. In the NuScale Power Plant design there is a nonsafety-related Reactor Building heating ventilating and air conditioning 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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 6.5.3, Rev 3: Fission Product Control Systems and Structures	II.1	Primary Containment	Partially Conforms	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 Power 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	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	The only credited ESF fission product control system in the NuScale Power Plant design is the containment vessel in conjunction with the containment isolation valves and passive containment isolation barriers.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 6.5.4, Draft Rev 4: Ice Condenser as a Fission Product Cleanup System	All	Various	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	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
DSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.1	Components Subject to Inspection	Conforms	None.	6.6.1
DSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.2	Accessibility	Conforms	None.	6.6.2
DSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.3	Examination Categories and Methods	Conforms	None.	6.6.3
DSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.4	Inspection Intervals	Conforms	None.	6.6.4
DSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.5	Evaluation of Examination Results	Conforms	None.	6.6.5
DSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.6	System Pressure Tests	Conforms	None.	6.6.7
DSRS 6.6, Rev. 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.7	Structural Supports	Conforms	None.	6.6.1 6.6.5 Table 6.6-1
DSRS 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	None.	6.6.8

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.9	Code Exemptions	Conforms	None.	6.6
DSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.10	Relief Requests	Conforms	None.	6.6
DSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.11	Code Cases	Conforms	None.	6.6
DSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.12	Operational Programs	Not Applicable	The operational program and implementation milestones governed by this acceptance criterion are the responsibility of the COL applicant.	Not Applicable
DSRS 6.6, Rev 0: Inservice Inspection and Testing of Class 2 and 3 Components	II.13	Risk Informed ISI Program	Not Applicable	None.	Not Applicable
SRP 6.7, Draft Rev 3: Main Steam Isolation Valve Leakage Control System (BWR)	All	Various	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	This acceptance criterion is applicable but certain language in BTP 6-1, which would be applied by Acceptance Criterion B.1, refers to SSC 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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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	The intent of a portion of this acceptance criterion is applicable but the specific language refers to SSC 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.	6.2.2
BTP 6-1, (March 2007): pH for Emergency Coolant Water for Pressurized Water Reactors	B.3	Hydrogen Generation from Aluminum Corrosion	Conforms	This acceptance criterion is applicable but certain language in BTP 6-1, which would be applied by Acceptance Criterion B.3, refers to SSC 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	This guidance is applicable only to PWRs for which a postulated LOCA results in core uncovery. For the NuScale design, a LOCA does not result in core uncovery.	Not Applicable
BTP 6-3, Rev 3: Determination of Bypass Leakage Paths in Dual Containment Plants	All	Various	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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
BTP 6-4, Rev 3: Containment Purging During Normal Plant Operations	All (B.1 thru B.5)	Various	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 includes features that provide suitable means to prevent radiological release to the environs (see DSRS 6.2.4, AC II.13). (The NuScale containment vessel evacuation system valve closure times are addressed under SRP Section 6.2.4.)	Not Applicable
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	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
DSRS 7.0, Rev 0: Instrumentation and Controls - Introduction and Overview of Review Process	All	Various	Conforms	This DSRS 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. 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
DSRS Appendix 7.0-A, Rev 0: I&C - Hazard Analysis	-	I&C - Hazard Analysis	Conforms	None.	7.1.8
DSRS Appendix 7.0-B, Rev 0: I&C - System Architecture	-	I&C - System Architecture	Conforms	None.	7.0.3 7.0.4 7.1 7.2
DSRS Appendix 7.0-C, Rev 0: I&C - Simplicity	-	I&C - Simplicity	Conforms	None.	7.1.6 7.1.7 7.1.8

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 7.1.1, Rev 0: Fundamental Design Principles	All	Specific SRP Acceptance Criteria Applicable to I&C Systems Important to Safety are Listed in Table 7.0-1.	Conforms	None.	7.1 7.1.1
DSRS 7.1.2, Rev 0: Independence	II.1	Ensure compliance to current version of RG 1.75	Conforms	None.	7.1.2
DSRS 7.1.2, Rev 0: Independence	II.2	Ensure compliance to current version of RG 1.152	Conforms	None.	7.1.2
DSRS 7.1.3, Rev 0: Redundancy	-	Conformance with RG 1.53	Conforms	None.	7.1.3
DSRS 7.1.4, Rev 0: Predictability and Repeatability	-	Predictability and Repeatability	Conforms	None.	7.1.4
DSRS 7.1.5, Rev 0: Diversity and Defense in Depth	II.1	Methods for performing D3 analyses of reactor protection systems	Conforms	The D3 assessment of the NuScale I&C design is consistent with the guidelines in NUREG/CR-6303.	7.1.5
DSRS 7.1.5, Rev 0: Diversity and Defense in Depth	II.2	SECY-93-087	Conforms	Conformance to the applicable regulatory guidance from the staff requirement memorandum to SECY-93-087 is summarized in Section 7.1.5.	7.1.5
DSRS 7.1.5, Rev 0: Diversity and Defense in Depth	II.3	GL 85-06	Conforms	Conformance to 10 CFR 50.62 is summarized in Section 7.1.6. I&C systems are designed to the quality assurance program described in Section 17.5.	7.1.1 7.1.5 7.1.6
DSRS 7.1.5, Rev 0: Diversity and Defense in Depth	II.4	Conformance to RG 1.53	Conforms	None.	7.1.5
DSRS 7.1.5, Rev 0: Diversity and Defense in Depth	II.5	Conformance to RG 1.62	Conforms	See RG 1.62 in Table 1.9-2.	7.1.5
DSRS 7.1.5, Rev 0: Diversity and Defense in Depth	II.6	Conformance to IEEE Std. 7-4.3.2	Conforms	None.	7.1.1 7.1.2 7.1.5
DSRS 7.2.1 Rev. 0: Quality	II.1	Conformance to RG 1.28	Conforms	See RG 1.28 in Table 1.9-2.	7.2.1
DSRS 7.2.1 Rev. 0: Quality	II.2	Conformance to RG 1.152	Conforms	See RG 1.152 in Table 1.9-2.	7.2.1
DSRS 7.2.1 Rev. 0: Quality	II.3	Conformance to RG 1.168	Partially Conforms	See RG 1.168 in Table 1.9-2.	7.2.1
DSRS 7.2.1 Rev. 0: Quality	II.4	Conformance to RG 1.169	Partially Conforms	See RG 1.169 in Table 1.9-2.	7.2.1
DSRS 7.2.1 Rev. 0: Quality	II.5	Conformance to RG 1.170	Partially Conforms	See RG 1.170 in Table 1.9-2.	7.2.1
DSRS 7.2.1 Rev. 0: Quality	II.6	Conformance to RG 1.171	Partially Conforms	See RG 1.171 in Table 1.9-2.	7.2.1

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 7.2.1 Rev. 0: Quality	II.7	Conformance to RG 1.172	Partially Conforms	See RG 1.172 in Table 1.9-2.	7.2.1
DSRS 7.2.1 Rev. 0: Quality	II.8	Conformance to RG 1.173	Partially Conforms	See RG 1.173 in Table 1.9-2.	7.2.1
DSRS 7.2.2, Rev 0: Equipment Qualification	II.1	Conformance to IEEE Std 7- 4.3.2	Conforms	Digital I&C safety systems conform to the guidance in Section 5.4 of IEEE Std 7- 4.3.2-2003, IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations, as endorsed (with identified exceptions and clarifications) by RG 1.152, Rev. 3.	7.2.2
DSRS 7.2.2, Rev 0: Equipment Qualification	II.2	Conformance to RG 1.209	Conforms	None.	7.2.2
DSRS 7.2.2, Rev 0: Equipment Qualification	II.3	Conformance to RG 1.151	Partially Conforms	See RG 1.151 in Table 1.9-2.	7.2.2
DSRS 7.2.2, Rev 0: Equipment Qualification	II.4	Conformance to RG 1.180	Partially Conforms	See RG 1.180 in Table 1.9-2.	7.2.2
DSRS 7.2.2, Rev 0: Equipment Qualification	II.5	Conformance to RG 1.204	Partially Conforms	See RG 1.204 in Table 1.9-2.	7.2.2
DSRS 7.2.3, Rev 0: Reliability, Integrity, and Completion of Protective Action	II.1	Conformance to IEEE Std 7-4.3.2	Conforms	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 7-4.3.2-2003, as endorsed by RG 1.152 Rev. 3.	7.2.3
DSRS 7.2.4, Rev 0: Operating and Maintenance Bypasses	II.1	Conformance to RG 1.47	Conforms	None.	7.2.4
DSRS 7.2.5, Rev 0: Interlocks	II.1	Conformance to IEEE Std 7-4.3.2	Conforms	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 RG 1.152 Rev. 3.	7.2.5
DSRS 7.2.6, Rev 0: Derivation of System Inputs	All	Various	Conforms	There are no specific DSRS acceptance criteria in this section.	7.2.6
DSRS 7.2.7, Rev 0: Setpoints	II.1	Conformance to RG 1.105	Partially Conforms	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 7.2.7, Rev 0: Setpoints	II.2	NRC Regulatory Issue Summary (RIS) 2006-17	Conforms	The setpoint methodology conforms to ISA-67.04.01-2006, Setpoints for Nuclear Safety-Related Instrumentation, which addresses the issues identified in RIS 2006-17.	7.2.7
DSRS 7.2.7, Rev 0: Setpoints	II.3	Generic Letter (GL) 91-04	Conforms	The guidance of GL 91-04 is applicable to the setpoint methodology as described in TR-616-49121, NuScale Instrument Setpoint Methodology Technical Report (Ref. 7.2-27).	7.2.7
DSRS 7.2.8, Rev 0: Auxiliary Features	All	Various	Conforms	There are no specific DSRS acceptance criteria in this section.	7.2.8
DSRS 7.2.9, Rev 0: Control of Access, Identification, and Repair	II.1	Conformance to IEEE Std 7-4.3.2	Conforms	Digital I&C safety systems and components conform to the identification guidance in Section 5.11 of IEEE Std 7-4.3.2-2003.	7.2.9
DSRS 7.2.9, Rev 0: Control of Access, Identification, and Repair	II.2	Conformance to RG 1.75	Conforms	None.	7.2.9
DSRS 7.2.10, Rev 0: Interaction Between Sense and Command Features and Other Systems	All	Varies	Conforms	There are no specific DSRS 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
DSRS 7.2.11, Rev 0: Multi-Unit Stations	II.1	Conformance to RG 1.53	Conforms	None.	7.2.11
DSRS 7.2.12, Rev 0: Automatic and Manual Controls	II.1	Conformance to RG 1.62	Conforms	See RG 1.62 in Table 1.9-2.	7.2.12
DSRS 7.2.13, Rev 0: Displays and Monitoring	II.1	Conformance to RG 1.97	Partially Conforms	See RG 1.97 in Table 1.9-2.	7.2.13
DSRS 7.2.13, Rev 0: Displays and Monitoring	II.2	Conformance to RG 1.47	Conforms	None.	7.2.4 7.2.13 7.2.15
DSRS 7.2.13, Rev 0: Displays and Monitoring	II.3	SECY-93-087	Conforms	The main control room and remote shutdown station are designed to maintain alarm system reliability in accordance with item II.T of SECY-93-087.	7.2.13

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 7.2.14, Rev 0: Human Factors Considerations	All	Various	Conforms	There are no specific DSRS 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
DSRS 7.2.15, Rev 0: Capability for Test and Calibration	II.1	Conformance to IEEE Std 7-4.3.2	Conforms	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 7-4.3.2-2003.	7.2.15
DSRS 7.2.15, Rev 0: Capability for Test and Calibration	II.2	Conformance to RG 1.118	Partially Conforms	See RG 1.118 in Table 1.9-2.	7.2.15
DSRS 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	DSRS 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
DSRS 8.2, Rev 0: Offsite Power System	II.1	Compliance with GDC 5	Not Applicable	Conformance with GDC 5 is the responsibility of the COL applicant as described in Section 8.2.2.	Not Applicable
DSRS 8.2, Rev 0: Offsite Power System	II.2	Compliance with GDC 17	Departure	The NuScale design supports an exemption from GDC 17 that includes the associated requirements for the offsite power system.	8.2.3
DSRS 8.2, Rev 0: Offsite Power System	II.3	Compliance with GDC 18	Departure	The NuScale design supports an exemption from GDC 18 that includes the associated requirements for the offsite power system.	8.2.3
DSRS 8.2, Rev 0: Offsite Power System	II.4	Compliance with GDC 33	Departure	The NuScale design supports an exemption from GDC 33.	8.2.3
DSRS 8.2, Rev 0: Offsite Power System	II.4	Compliance with GDCs 34, 35, 38, 41, and 44	Departure	NuScale complies with a set of principal design criteria in lieu of these GDC.	8.2.3

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 8.2, Rev 0: Offsite Power System	II.5	Compliance with 10 CFR 50.63 - Passive Design	Conforms	The details regarding conformance with 10 CFR 50.63 are described in Section 8.4, Station Blackout.	8.2.3 8.4
DSRS 8.2, Rev 0: Offsite Power System	II.6	Compliance with 10 CFR 50.65(a)(4)	Not Applicable	Development of the maintenance rule (10 CFR 50.65) program is the responsibility of the COL applicant referencing the certified design.	Not Applicable
DSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.1	Compliance with GDC 2	Conforms	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
DSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.2	Compliance with GDC 4	Conforms	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
DSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.3	Compliance with GDC 5	Partially Conforms	Onsite AC power systems conform to GDC 5 to the extent described in Section 8.3.1.	8.1.4 8.3.1
DSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.4	Compliance with GDC 17	Departure	The NuScale design supports an exemption from GDC 17 that includes the associated requirements for the onsite AC power system.	8.3.1
DSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.5	Compliance with GDC 18	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
DSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II (No Number)	Compliance with GDC 33	Departure	The NuScale design supports an exemption from GDC 33.	8.1.4 8.3.1
DSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II (No Number)	Compliance with GDCs 34, 35, 38, 41, and 44	Departure	NuScale complies with a set of principal design criteria in lieu of these GDC.	8.1.4 8.3.1
DSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.6	Compliance with GDC 50	Conforms	The electrical design requirements associated with GDC 50 for electrical penetration assemblies (EPAs) are included in Section 8.3.	8.1 8.3
DSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.7	Compliance with 10 CFR 50.65(a)(4)	Not Applicable	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.	

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.8	Compliance with 10 CFR 50.55a(h)	Not Applicable	No onsite electrical AC power system equipment is required to conform to 10 CFR 50.55a(h) and IEEE Std. 603-1991.	Not Applicable
DSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.9	Compliance with 10 CFR 52.47(b)(1)	Conforms	None.	8.1 8.3
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 2	Conforms	None.	8.3.2
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 4	Conforms	None.	8.3.2
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 5	Conforms	None.	8.3.2
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 17	Departure	The NuScale design supports an exemption from GDC 17 that includes the associated requirements for the onsite DC power systems.	8.3.2
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 18	Departure	The NuScale design supports an exemption from GDC 18 that includes the associated requirements for the onsite DC power systems.	8.3.2
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 33	Departure	The NuScale design supports an exemption from GDC 33.	8.3.2
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 34, 35, 38, 41, and 44	Departure	Nuscale complies with a set of principal design criteria in lieu of these GDC.	8.3.2
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 50	Conforms	The electrical design requirements associated with GDC 50 for electrical penetration assemblies (EPAs) are included in Section 8.3.	8.1 8.3
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.1	Conformance with RG 1.32	Partially Conforms	As it applies to certain aspects of the EDSS design.	8.3.2
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.2	Conformance with RG 1.75	Conforms	As it applies to certain aspects of the EDSS design.	8.3.2
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.3	Conformance with RG 1.81	Partially Conforms	As it applies to certain aspects of the EDSS design.	8.3.2
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.4	Conformance with RG 1.118	Partially Conforms	As it applies to certain aspects of the EDSS design.	8.3.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.5	Conformance with RG 1.153	Partially Conforms	As it applies to certain aspects of the EDSS design.	8.3.2
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.6	Conformance with RG 1.153	Partially Conforms	As it applies to certain aspects of the EDSS design.	8.3.2
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.7	Conformance with RG 1.63	Partially Conforms	See RG 1.63 in Table 1.9-2.	8.1 8.3.2
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.8	Conformance with RG 1.160	Not Applicable	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
DSRS 8.4, Rev 0: Station Blackout	II.1	Compliance with 10 CFR 50.63 and the guidelines of RG 1.155	Partially Conforms	None.	8.4
DSRS 8.4, Rev 0: Station Blackout	II.2	Use of Alternate AC Power Sources and RTNSS for Plants of Passive Design	Partially Conforms	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. 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
DSRS 8.4, Rev 0: Station Blackout	II.3	Independence of SBO-related power sources	Partially Conforms	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
SRP Appendix 8-A, Rev1: General Agenda, Station Site Visits	All	Various	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	The NuScale 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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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	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	The analysis of grid stability is the responsibility of the COL applicant that references the NuScale design certification.	Not Applicable
SRP BTP 8-3, Rev 3: Stability of Offsite Power Systems	B.2	Grid Capacity	Not Applicable	The analysis of grid stability is the responsibility of the COL applicant that references the NuScale design certification.	Not Applicable
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	BTP 8-4 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.	Not Applicable
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	Not Applicable	This BTP does not apply to NuScale electric power systems as these systems are not engineered safety features and are not relied on to support engineered safety features.	Not Applicable
SRP BTP 8-6, Rev 3: Adequacy of Station Electric Distribution System Voltages	All	Criteria for evaluating voltage protection for the offsite power system to assure proper operation and sequencing of Class 1E loads	Not Applicable	For the NuScale design, the offsite power system does not supply power to Class 1E loads and does not support safety-related functions.	Not Applicable
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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP BTP 8-8, (Feb 2012): Onsite (Emergency Diesel Generators) and Offsite Power Sources Allowed Outage Time Extensions	All	Various	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	None.	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 to passive plant designs.	Not Applicable
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	None.	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	None.	9.1.1.3 9.1.1.1
DSRS 9.1.2, Rev 0: New and Spent Fuel Storage	II.1	Specific Criteria to Meet GDC 2	Conforms	None.	9.1.2.1 9.1.2.3
DSRS 9.1.2, Rev 0: New and Spent Fuel Storage	II.2	Specific Criteria to Meet GDC 4	Conforms	None.	9.1.2.1 9.1.2.3
DSRS 9.1.2, Rev 0: New and Spent Fuel Storage	II.3	Specific Criteria to Meet GDC 5	Conforms	None.	9.1.2.1 9.1.2.3
DSRS 9.1.2, Rev 0: New and Spent Fuel Storage	II.4	Specific Criteria to Meet GDC 61	Conforms	An ESF ventilation system is not required (see RG 1.52).	9.1.2.1 9.1.2.3
DSRS 9.1.2, Rev 0: New and Spent Fuel Storage	II.5	Specific Criteria to Meet GDC 63	Conforms	None.	9.1.2.1 9.1.2.3 9.1.2.5
DSRS 9.1.2, Rev 0: New and Spent Fuel Storage	II.6	Specific Criteria to Meet 10 CFR 20.1101(b)	Conforms	None.	9.1.2.1 9.1.2.2 9.1.2.3
DSRS 9.1.2, Rev 0: New and Spent Fuel Storage	II.7	Criticality Monitors and Subcriticality Margin	Conforms	None.	9.1.1

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 9.1.3, Rev 0: Spent Fuel Pool Cooling and Cleanup System	II.1	Specific Criteria to Meet GDC 2	Partially Conforms	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 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
DSRS 9.1.3, Rev 0: Spent Fuel Pool Cooling and Cleanup System	II.2	Specific Criteria to Meet GDC 4	Partially Conforms	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 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
DSRS 9.1.3, Rev 0: Spent Fuel Pool Cooling and Cleanup System	II.3	Specific Criteria to Meet GDC 5	Conforms	None.	9.1.3.1 9.1.3.3
DSRS 9.1.3, Rev 0: Spent Fuel Pool Cooling and Cleanup System	II.4	Specific Criteria to Meet GDC 61	Conforms	None.	9.1.3.1 9.1.3.2
DSRS 9.1.3, Rev 0: Spent Fuel Pool Cooling and Cleanup System	II.5	Specific Criteria to Meet GDC 63	Conforms	None.	9.1.3.1 9.1.3.5
DSRS 9.1.3, Rev 0: Spent Fuel Pool Cooling and Cleanup System	II.6	Specific Criteria to Meet 10 CFR 20.1101(b)	Conforms	None.	9.1.3.1 9.1.3.3

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 9.1.3, Rev 0: Spent Fuel Pool Cooling and Cleanup System	II.7	ITAAC for Design Certification Applications	Conforms	None.	14.3
DSRS 9.1.3, Rev 0: Spent Fuel Pool Cooling and Cleanup System	II.8	ITAAC for Combined License Applications	Not Applicable	This acceptance criterion is applicable only to COL applicants.	Not Applicable
SRP 9.1.4, Rev 4: Light Load Handling System and Refueling Cavity Design	II.1	Specific Criteria to Meet GDC 2	Conforms	None.	9.1.4
SRP 9.1.4, Rev 4: Light Load Handling System and Refueling Cavity Design	II.2	Specific Criteria to Meet GDC 5	Conforms	None.	9.1.4
SRP 9.1.4, Rev 4: Light Load Handling System and Refueling Cavity Design	II.3	Specific Criteria to Meet GDC 61	Conforms	None.	9.1.4
SRP 9.1.4, Rev 4: Light Load Handling System and Refueling Cavity Design	II.4	Specific Criteria to Meet GDC 62	Conforms	None.	9.1.4
SRP 9.1.5, Rev 1: Overhead Heavy Load Handling Systems	II.1	Specific Criteria to Meet GDC 1	Conforms	None.	9.1.5
SRP 9.1.5, Rev 1: Overhead Heavy Load Handling Systems	II.2	Specific Criteria to Meet GDC 2	Conforms	None.	9.1.5
SRP 9.1.5, Rev 1: Overhead Heavy Load Handling Systems	II.3	Specific Criteria to Meet GDC 4	Conforms	None.	9.1.5
SRP 9.1.5, Rev 1: Overhead Heavy Load Handling Systems	II.4	Specific Criteria to Meet GDC 5	Conforms	None.	9.1.5

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 9.2.1, Rev 5: Station Service Water System	II.1	Protection Against Natural Phenomena (GDC 2)	Conforms	The NuScale site cooling water system (SCWS) does not provide essential cooling to safety-related SSC and is not safety-related or important-to-safety. The applicability of GDC 2 to the SCWS reviewed under this acceptance criterion is limited to aspects ensuring that a failure of the nonsafety-related SCWS does not result in an adverse effect on a Seismic Category I SSC. For the NuScale design, this is provided by the design and construction of the nonsafety related SCWS to meet the provisions of RG 1.29, Staff Regulatory Guidance C.1.i.	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	The NuScale site cooling water system does not provide essential cooling to safety-related SSC and is 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	The NuScale site cooling water system does not provide essential cooling to safety-related SSC 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 SSC and associated safety functions.	9.2.7 (Used for Site Cooling Water System (SCWS))
SRP 9.2.1, Rev 5: Station Service Water System	II.4	Cooling Water System (GDC 44)	Not Applicable	The site cooling water system does not serve a safety-related cooling or accident mitigation function.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 9.2.1, Rev 5: Station Service Water System	II.5	Cooling Water System Inspection (GDC 45)	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.6	Cooling Water System Testing (GDC 46)	Not Applicable	The site cooling water system does not serve a safety-related cooling or accident mitigation function.	Not Applicable
SRP 9.2.2, Rev 4: Reactor Auxiliary Cooling Water System	II.1	Protection Against Natural Phenomena	Partially Conforms	The system function contemplated by this SRP criterion is applicable to the NuScale reactor component cooling water (RCCW) system. This criterion is based on RG 1.29 position C.1 for safety-related portions, and position C.2 for nonsafety-related portions. Position C.1 is not applicable since the RCCW is not safety related. The NuScale RCCW 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.	9.2.2
SRP 9.2.2, Rev 4: Reactor Auxiliary Cooling Water System	II.2	Environmental and Dynamic Effects	Partially Conforms	Additional information pertaining to impact of environmental and dynamic effects is provided in Sections 3.5 and 3.6.	9.2.2
SRP 9.2.2, Rev 4: Reactor Auxiliary Cooling Water System	II.3	Sharing of Structures, Systems, and Components	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	See comment above for Acceptance Criterion II.1.	Not Applicable
SRP 9.2.2, Rev 4: Reactor Auxiliary Cooling Water System	II.5	Cooling Water System Inspection	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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 9.2.4, Rev 3: Potable and Sanitary Water Systems	II.1	Control of Releases of Radioactive Materials to the PWSW	Partially Conforms	The NuScale potable and sanitary water systems do not interface with systems potentially containing radioactivity. The NuScale potable and sanitary water systems are designed such that failure will not result in flooding or other adverse impacts on essential SSC.	9.2.4
SRP 9.2.5, Rev 3: Ultimate Heat Sink	II.1	Protection Against Natural Phenomena	Partially Conforms	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 provides both spent fuel cooling and containment heat removal, and 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	None.	9.2.5
SRP 9.2.5, Rev 3: Ultimate Heat Sink	II.3	Cooling Water System	Partially Conforms	This acceptance criterion is applicable except for aspects related to the use of fiberglass piping (see RG 1.72). The NuScale design does not use fiberglass piping.	9.2.5
SRP 9.2.5, Rev 3: Ultimate Heat Sink	II.4	Cooling Water System Inspection	Conforms	None.	9.2.5
SRP 9.2.5, Rev 3: Ultimate Heat Sink	II.5	Cooling Water System Testing	Conforms	None.	9.2.5

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 9.2.6, Rev 3: Condensate Storage Facilities	II.1	Protection Against Natural Phenomena	Conforms	The NuScale design's 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 Staff Regulatory Guidance C.1.i of RG 1.29, no portion of the NuScale condensate storage system requires design and construction to withstand the safe-shutdown earthquake to prevent a failure that could adversely affect a Seismic Category I SSC.	9.2.6 10.4.7
SRP 9.2.6, Rev 3: Condensate Storage Facilities	II.2	Environmental and Dynamic Effects Design Basis	Conforms	None.	9.2.6 10.4.7
SRP 9.2.6, Rev 3: Condensate Storage Facilities	II.3	Sharing of Structures, Systems, and Components	Conforms	Sharing of the condensate storage facilities does not impair the ability of safety-related or risk-significant SSC 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	Conforms	None.	9.2.6 10.4.7
SRP 9.2.6, Rev 3: Condensate Storage Facilities	II.5	10 CFR 20.1406 Compliance	Conforms	None.	9.2.6 10.4.7
SRP 9.2.7, Rev 0: Chilled Water System	II.1	Quality Standards and Records	Not Applicable	The NuScale CHWS does not perform safety or containment isolation functions.	Not Applicable
SRP 9.2.7, Rev 0: Chilled Water System	II.2	Protection Against Natural Phenomena	Conforms	This criterion is based on RG 1.29. The CHWS is not classified as Seismic Category I. The CHWS complies with Staff Regulatory Guidance C.1.i in that the SSC whose failure could adversely affect Seismic Category I SSC are designed as Seismic Category II.	9.2.8
SRP 9.2.7, Rev 0: Chilled Water System	II.3	Environmental and Dynamic Effects	Conforms	None.	9.2.8

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 9.2.7, Rev 0: Chilled Water System	II.4	Sharing of Structures, Systems, and Components	Not Applicable	The NuScale CHWS is a nonsafety-system and does not perform safety functions.	Not Applicable
SRP 9.2.7, Rev 0: Chilled Water System	II.5	Cooling Water System	Not Applicable	The NuScale CHWS is a nonsafety-system and does not perform safety functions.	Not Applicable
SRP 9.2.7, Rev 0: Chilled Water System	II.6	Cooling Water System Inspection	Not Applicable	The NuScale CHWS is a nonsafety-system and does not perform safety functions.	Not Applicable
SRP 9.2.7, Rev 0: Chilled Water System	II.7	Cooling Water System Testing	Not Applicable	The NuScale CHWS is a nonsafety-system and does not perform safety functions.	Not Applicable
SRP 9.2.7, Rev 0: Chilled Water System	II.8	Minimization of Contamination	Conforms	The CHWS is at a higher pressure than the LRWS and GRWS where the systems interface, precluding introduction of radioactive contaminants into the CHWS.	9.2.8
SRP 9.3.1, Rev 2: Compressed Air System	II.1	Specific Criteria to Meet GDC 1	Not Applicable	NuScale compressed air systems are non-safety, non-risk-significant systems.	Not Applicable
SRP 9.3.1, Rev 2: Compressed Air System	II.2	Specific Criteria to Meet GDC 2	Not Applicable	NuScale compressed air systems are non-safety, non-risk-significant systems.	Not Applicable
SRP 9.3.1, Rev 2: Compressed Air System	II.3	Specific Criteria to Meet GDC 5	Conforms	None.	9.3.1
SRP 9.3.1, Rev 2: Compressed Air System	II.4	Specific Criteria to Meet 10 CFR 50.63	Partially Conforms	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 - are applicable. However, language that refers to reactor plant designs such as large LWRs is not relevant to the NuScale plant design. The NuScale plant design meets the intent of this guidance with its passive design and reduced reliance on AC power to cope 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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 9.3.2, Rev 3: Process and Post-Accident Sampling Systems	II.1	Sampling Capability	Partially Conforms	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	Technical Specifications	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.3	Process Sampling System Functional Design	Conforms	None.	9.3.2
SRP 9.3.2, Rev 3: Process and Post-Accident Sampling Systems	II.4	Seismic Design and Quality Group Classification	Conforms	None.	9.3.2
SRP 9.3.3, Rev 3: Equipment and Floor Drainage System	II.1	Protection Against Natural Phenomena	Conforms	None.	9.3.3
SRP 9.3.3, Rev 3: Equipment and Floor Drainage System	II.2	Environmental and Dynamic Effects	Conforms	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	No portions of the NuScale drain system penetrate the containment barrier.	9.3.3
DSRS 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	The only CVCS safety-related function precludes inadvertent boron dilution of the reactor coolant system.	9.3.4
DSRS 9.3.4, Rev 0: Chemical and Volume Control System (PWR) (Including Boron Recovery System)	II.2	Single Failure Criteria and GDC 5	Conforms	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
DSRS 9.3.4, Rev 0: Chemical and Volume Control System (PWR) (Including Boron Recovery System)	II.3	Minimization of contamination	Conforms	None.	9.3.4

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 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	The CVCS is located outside the RCPB.	9.3.4
DSRS 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	None.	9.3.4
DSRS 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	None.	9.3.4
DSRS 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	A portion of this acceptance criterion is applicable but the specific language refers to CVCS designs that are not relevant to the NuScale design. The NuScale CVCS design does not have holdup tanks that are subject to the vacuum conditions 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
DSRS 9.3.4, Rev 0: Chemical and Volume Control System (PWR) (Including Boron Recovery System)	II.8	ITAAC	Conforms	None.	9.3.4 14.3
SRP 9.3.5, Rev 3: Standby Liquid Control System (BWR)	All	Various	Not Applicable	This SRP section and its acceptance criteria are applicable only to BWRs.	Not Applicable
DSRS 9.3.6, Rev 0: Containment Evacuation and Flooding Systems	II.1	GDC 2	Conforms	None.	9.3.6
DSRS 9.3.6, Rev 0: Containment Evacuation and Flooding Systems	II.2	GDC 60	Conforms	None.	9.3.6

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 9.3.6, Rev 0: Containment Evacuation and Flooding Systems	II.3	TMI 10 CFR 50.34(f)	Conforms	None.	9.3.6
SRP 9.4.1, Rev 3: Control Room Area Ventilation System	II.1	Protection Against Natural Phenomena	Conforms	None.	9.4.1
SRP 9.4.1, Rev 3: Control Room Area Ventilation System	II.2	Environmental and Dynamic Effects	Conforms	None.	9.4.1
SRP 9.4.1, Rev 3: Control Room Area Ventilation System	II.3	Sharing of Structures, Systems, and Components	Conforms	Operation of the CRVS is part of normal plant operations. Up to 12 Modules modules share the same control room.	9.4.1
SRP 9.4.1, Rev 3: Control Room Area Ventilation System	II.4	Control Room	Conforms	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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 9.4.1, Rev 3: Control Room Area Ventilation System	II.6	Loss of All Alternating Current Power	Conforms	The intent of this acceptance criterion and its subtier 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 with its passive design and reduced reliance on AC power to cope with design basis events. Consistent with Commission policy, this coping capability eliminates 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 on compressed air tanks to pressurize the control room envelope in the event of an 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	None.	9.4.2
SRP 9.4.2, Rev 3: Spent Fuel Pool Area Ventilation System	II.2	Compliance with GDC 5	Conforms	None.	9.4.2
SRP 9.4.2, Rev 3: Spent Fuel Pool Area Ventilation System	II.3	Compliance with GDC 60	Conforms	This acceptance criterion is applicable except for aspects related to ESF atmosphere cleanup systems (see RG 1.52).	9.4.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 9.4.2, Rev 3: Spent Fuel Pool Area Ventilation System	II.4	Compliance with GDC 61	Conforms	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	None.	9.4.3
SRP 9.4.3, Rev 3: Auxiliary and Radwaste Area Ventilation System	II.2	Compliance with GDC 5	Conforms	None.	9.4.3
SRP 9.4.3, Rev 3: Auxiliary and Radwaste Area Ventilation System	II.3	Compliance with GDC 60	Not Applicable	The RWBV system does not filter exhaust. Exhaust is filtered by the RBV system.	Not Applicable
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	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 HVAC system (TBVS) is not relied on to control airborne radioactivity concentrations in the Turbine Building and gaseous effluents during normal operations (including anticipated operational occurrences) and after accidents that result in a radioactive material release. Furthermore, there are no requirements for TBVS performance needed to preclude adverse effects on safety-related functions during conditions of plant operation.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 9.4.5, Rev 3: Engineered Safety Feature Ventilation System	All	Various	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. Nonsafety-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 are not credited for meeting applicable offsite dose limits.	Not Applicable
SRP 9.5.1.1, Rev 0: Fire Protection Program	II.1	Fire Protection Probabilistic Risk Assessment (Including Appendix C)	Not Applicable	Development and implementation of a risk-informed, performance-based fire protection program is 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	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	This acceptance criterion is applicable except NuScale will use the current year subtier documents.	9.5.1
SRP 9.5.1.1, Rev 0: Fire Protection Program	II.4	Fire Protection for Permanently Shutdown and Decommissioning Reactor Plants	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	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	This acceptance criterion and its subtier guidance apply to COL applicants under 10 CFR 52. COL applicants referencing a certified design are responsible for implementing this guidance. Notwithstanding the above, NuScale, as an applicant for a design certification, considers 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	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, 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
SRP 9.5.1.1, Rev 0: Fire Protection Program	II.7	Operational Program and Proposed Implementation Milestones	Not Applicable	This acceptance criterion is the responsibility of the COL applicant.	Not Applicable
SRP 9.5.1.2, Rev 0: Risk-Informed, Performance-Based Fire Protection Program	All	Various	Not Applicable	Development and implementation of a risk-informed, performance-based fire protection program is 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
DSRS 9.5.2, Rev 0: Communication Systems	II.1	Emergency Facilities and Equipment	Partially Conforms	This acceptance criterion is the responsibility of the COL applicant.	9.5.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 9.5.2, Rev 0: Communication Systems	II.2	Onsite Technical Support Center and Operational Support Center	Partially Conforms	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. Communication systems serving these facilities in support of emergency response are the responsibility of the COL applicant.	9.5.2
DSRS 9.5.2, Rev 0: Communication Systems	II.3	Emergency Facilities and Equipment for Meeting 10 CFR 52.47(a)(8)	Partially Conforms	The NuScale design includes 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. Communication systems and equipment serving these facilities in support of emergency response are the responsibility of the COL applicant.	9.5.2
DSRS 9.5.2, Rev 0: Communication Systems	II.4	Design, Fabrication, Erection, Construction, Testing, and Inspection of SSC to Meet 10 CFR 50.55a	Not Applicable	None.	Not Applicable
DSRS 9.5.2, Rev 0: Communication Systems	II.5	ITAAC	Conforms	The aspects of this acceptance criterion within the scope of the NuScale design are applicable to the DCA. Aspects related to site-specific design, fabrication, erection, construction, testing, and inspection of SSC, and maintenance of records for activities throughout the life of the facility, are the responsibility of the COL applicant referencing the certified design.	Ch 14
DSRS 9.5.2, Rev 0: Communication Systems	II.6	ITAAC for a COL applicant	Not Applicable	COL applicant responsibility to prepare COL-specific ITAAC.	Not Applicable
DSRS 9.5.2, Rev 0: Communication Systems	II.7	Compliance with GDC 1	Conforms	Site-specific scope is 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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 9.5.2, Rev 0: Communication Systems	II.8	Compliance with GDC 2	Conforms	None.	9.5.2
DSRS 9.5.2, Rev 0: Communication Systems	II.9	Compliance with GDC 3	Conforms	None.	9.5.2
DSRS 9.5.2, Rev 0: Communication Systems	II.10	Compliance with GDC 4	Conforms	None.	9.5.2
DSRS 9.5.2, Rev 0: Communication Systems	II.11	Compliance with GDC 19	Departure	The NuScale design supports an exemption from GDC 19. As described in Section 3.1.2, the design complies with a NuScale-specific principal design criterion (PDC) in lieu of this GDC. Design documents meet requirements of PDC-19 for 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 off-site communication requirements even in the event of a single failure within a communication subsystem or the loss of the normal power source. The design addresses control room communications so that control room can maintain communications with site and offsite entities during normal and accident conditions.	9.5.2
DSRS 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	This acceptance criterion is applicable only to licensees subject to 10 CFR 73.45 and the general performance requirements of 10 CFR 73.20. The NuScale design does not reprocess spent fuel or use or transport special nuclear material.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 9.5.2, Rev 0: Communication Systems	II.13	Compliance with 10 CFR 73.46(f)	Conforms	Site-specific, programmatic aspects of physical security communication systems are the responsibility of the COL applicant referencing the certified design. Aspects of this acceptance criterion related to the physical design of the power reactor and communication systems are within the scope of the certified design and are applicable to the DCA.	9.5.2
DSRS 9.5.2, Rev 0: Communication Systems	II.14	Compliance with 10 CFR 73.55(e)(9)(vi)(B)	Conforms	Site-specific, programmatic aspects of physical security communication systems 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 (via Security Technical Report)
DSRS 9.5.2, Rev 0: Communication Systems	II.15	Compliance with 10 CFR 73.55(j)	Partially Conforms	Design focus pertains to addressing requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage and the communication requirements necessary for 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	None.	9.5.3
SRP 9.5.3, Rev 3: Lighting Systems	II.2	Emergency Lighting System(s)	Conforms	None.	9.5.3
SRP 9.5.3, Rev 3: Lighting Systems	II.3	Lighting Levels	Conforms	None.	9.5.3

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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	The NuScale plant design does not require or include safety-related emergency diesel generators that are 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	The NuScale plant design does not require or include safety-related emergency diesel generators that are 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	The NuScale plant design does not require or include safety-related emergency diesel generators that are subject to this SRP section. No AC or DC power is relied upon for the performance of NuScale plant safety functions.	Not Applicable
SRP 9.5.7, 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	The NuScale plant design does not require or include safety-related emergency diesel generators that are 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	The NuScale plant design does not require or include safety-related emergency diesel generators that are 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 SSC important to safety from the effects of turbine missiles with a turbine overspeed protection system (GDC 4)	Not Applicable	The NuScale plant design relies on the use of barriers for the protection of SSCs important to safety from the effects of turbine missiles.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 10.2, Rev 3: Turbine Generator	II.2	Inservice Inspection covering valves essential for overspeed protection.	Not Applicable	The NuScale plant design relies on the use of barriers for the protection of SSCs important to safety from the effects of turbine missiles.	Not Applicable
SRP 10.2, Rev 3: Turbine Generator	II.3	Prevention of Adverse Effects on Safety-Related SSC in the Turbine Building	Not Applicable	There are no safety-related SSC in the Turbine Building.	Not Applicable
DSRS 10.2.3, Rev 0: Turbine Rotor Integrity	II.1	Materials Selection	Not Applicable	Per DSRS 10.2.3, Section I and DSRS 3.5.1.3, Section I.1, plants that use barriers to protect essential SSCs specified in RG 1.115 do not have to rely on the turbine missile generation probabilities, including turbine rotor integrity.	Not Applicable
DSRS 10.2.3, Rev 0: Turbine Rotor Integrity	II.2	Fracture Toughness	Not Applicable	Per DSRS 10.2.3, Section I and DSRS 3.5.1.3, Section I.1, plants that use barriers to protect essential SSCs specified in RG 1.115 do not have to rely on the turbine missile generation probabilities, including turbine rotor integrity.	Not Applicable
DSRS 10.2.3, Rev 0: Turbine Rotor Integrity	II.3	Pre-Service Inspection	Not Applicable	Per DSRS 10.2.3, Section I and DSRS 3.5.1.3, Section I.1, plants that use barriers to protect essential SSCs specified in RG 1.115 do not have to rely on the turbine missile generation probabilities, including turbine rotor integrity.	Not Applicable
DSRS 10.2.3, Rev 0: Turbine Rotor Integrity	II.4	Turbine Rotor Design	Not Applicable	Per DSRS 10.2.3, Section I and DSRS 3.5.1.3, Section I.1, plants that use barriers to protect essential SSCs specified in RG 1.115 do not have to rely on the turbine missile generation probabilities, including turbine rotor integrity.	Not Applicable
DSRS 10.2.3, Rev 0: Turbine Rotor Integrity	II.5	Inservice Inspection	Not Applicable	Per DSRS 10.2.3, Section I and DSRS 3.5.1.3, Section I.1, plants that use barriers to protect essential SSCs specified in RG 1.115 do not have to rely on the turbine missile generation probabilities, including turbine rotor integrity.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 10.2.3, Rev 0: Turbine Rotor Integrity	II.6	10 CFR 52.47(b)(1) ITAAC	Not Applicable	Per DSRS 10.2.3, Section I and DSRS 3.5.1.3, Section I.1, plants that use barriers to protect essential SSCs specified in RG 1.115 do not have to rely on the turbine missile generation probabilities, including turbine rotor integrity.	Not Applicable
DSRS 10.3, Rev 0: Main Steam Supply System	II.1	Protection against natural phenomena (GDC 2)	Conforms	The NuScale main steam system (MSS) is not safety-related, but the portion of the system downstream of the main steam isolation valves (MSIV) inside the RXB includes the secondary MSIVs which act as backup to the MSIVs. Functionality is ensured by the design and construction of the MSS to the provisions of RG 1.29, Staff Regulatory Guidance C.1.i and C.2.	10.3.1
DSRS 10.3, Rev 0: Main Steam Supply System	II.2	Protection of SSC important to safety from the effects of turbine missiles (GDC 4)	Conforms	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
DSRS 10.3, Rev 0: Main Steam Supply System	II.3	Shared SSC important to safety perform required safety functions (GDC 5)	Conforms	None.	10.3.1
DSRS 10.3, Rev 0: Main Steam Supply System	II.4	MSS is capable of supporting core cooling or safe-shutdown (non-DBA) in the event of an SBO (10 CFR 50.63)	Partially Conforms	The intent of this acceptance criterion and its subtier guidance is applicable. The NuScale plant design meets the intent of this guidance with its passive design and reduced reliance on AC power to cope with design basis events.	10.3.1
DSRS 10.3, Rev 0: Main Steam Supply System	II.5	Protection of Important-to-Safety SSC from Tornado Missiles (RG 1.117, Appendix Positions 2 and 4)	Conforms	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	Not Applicable	The NuScale design contains no Class 2 or 3 components.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 10.3.6, Rev 3: Steam and Feedwater System Materials	II.2	Fracture Toughness of Class 2 and 3 Components	Not Applicable	The NuScale design contains no Class 2 or 3 components.	Not Applicable
SRP 10.4.1, Rev 3: Main Condensers	II.1	Prevent excessive releases of radioactivity to the environment (GDC 60)	Conforms	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	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	None.	10.4.3
SRP 10.4.4, Rev 3: Turbine Bypass System	II.1	Piping Failures (GDC 4)	Conforms	None.	10.4.4
SRP 10.4.4, Rev 3: Turbine Bypass System	II.2	Residual Heat Removal (GDC 34)	Departure	The NuScale design supports an exemption from the power provisions of GDC 34. As described in Section 3.1.4, the design complies with a NuScale-specific principal design criterion in lieu of this GDC.	10.4.4
SRP 10.4.4, Rev 3: Turbine Bypass System	II.3	MSIV Alternate Leakage Path	Not Applicable	BWR only.	Not Applicable
SRP 10.4.5, Rev 3: Circulating Water System	II.1	Flooding of SSC important to safety (GDC 4)	Conforms	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	BWR only.	Not Applicable
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	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
DSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.1	Seismic Events (GDC 2)	Conforms	None.	10.4.7

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.2	Fluid Instabilities (GDC 4)	Partially Conforms	The intent of this acceptance criterion and its subtier guidance - to satisfy GDC 4 related to protecting SSC 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
DSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.3	Sharing of Structures, Systems, and Components (GDC 5)	Conforms	None.	10.4.7
DSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.4	Heat Removal Capability (GDC 44)	Not Applicable	The CFWS is not a system used to transfer heat to an ultimate heat sink.	Not Applicable
DSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.5	Inspection (GDC 45)	Not Applicable	The CFWS is not a system used to transfer heat to an ultimate heat sink.	Not Applicable
DSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.6	Testing (GDC 46)	Not Applicable	The CFWS is not a system used to transfer heat to an ultimate heat sink.	Not Applicable
DSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.7	Flow Accelerated Corrosion	Conforms	None.	10.4.7
SRP 10.4.8, Rev 3: Steam Generator Blowdown System	All	Various	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	The NuScale design neither requires nor uses an auxiliary feedwater system. The NuScale decay heat removal system (DHRS) performs some functions similar to an auxiliary feedwater system. However, compared to an auxiliary feedwater system, the DHRS differs 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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	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	This guidance is applicable only to large PWRs that use Auxiliary Feedwater (AFW) system pumps powered by electrical and steam sources. The NuScale DHRS fulfills a similar function as the AFW system at a large PWR. The NuScale DHRS design does not use pumps: it 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	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	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	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 DHRS that fulfills a similar function as a typical AFW system. However, the NuScale steam generator design precludes potential water hammer issues without providing DHRS 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	None.	5.4.1
DSRS 11.1, Rev 0: Coolant Source Terms	II.1	RG 1.110	Partially Conforms	See RG 1.110 in Table 1.9-2.	11.1
DSRS 11.1, Rev 0: Coolant Source Terms	II.2	RG 1.112	Partially Conforms	See RG 1.112 in Table 1.9-2.	11.1

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 11.1, Rev 0: Coolant Source Terms	II.3	RG 1.140	Partially Conforms	RG 1.140 in Table 1.9-2.	11.1
DSRS 11.1, Rev 0: Coolant Source Terms	II.4	DC/COL-ISG-5	Not Applicable	The NuScale design is similar to large PWRs in the existing fleet for effluent release calculations. However, an alternate methodology is used because the existing PWRGALE code was developed in the 1980s for evaluation of the PWR reactors of that time and does not address the NuScale plant design.	Not Applicable
DSRS 11.1, Rev 0: Coolant Source Terms	II.5	normal operation and AOO sources of radioactive liquid and gaseous effluents	Conforms	None.	11.1
DSRS 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	The NuScale design is similar to large PWRs in the existing fleet for effluent release calculations. However, an alternate methodology is used because the existing PWRGALE code was developed in the 1980s for evaluation of the large PWR reactors of that time and does not address the NuScale plant design. Some aspects of ANSI/ANS 18.1 are used for the coolant source terms.	11.1
DSRS 11.1, Rev 0: Coolant Source Terms	II.7	Decontamination factors used to reduce gaseous effluent releases to the environment	Partially Conforms	Decontamination factors are consistent with the NuScale Technical Report, Effluent Release (GALE Replacement) Methodology and Results, Revision 0 (TR-1116-52065).	11.1
DSRS 11.1, Rev 0: Coolant Source Terms	II.8	Decontamination factors applied to reduce liquid effluent releases to the environment	Partially Conforms	Decontamination factors are consistent with the NuScale Technical Report, Effluent Release (GALE Replacement) Methodology and Results, Revision 0 (TR-1116-52065).	11.1
DSRS 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	See RG 1.110 in Table 1.9-2.	11.1
DSRS 11.1, Rev 0: Coolant Source Terms	II.10	Primary and secondary coolant source terms, used in characterizing liquid and gaseous effluents	Conforms	None.	11.2 11.3

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 11.1, Rev 0: Coolant Source Terms	II.11	If neutron activation products are expected in reactor pool water and secondary coolant	Conforms	None.	11.1
DSRS 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	The NuScale design is similar to large PWRs in the existing fleet for effluent release calculations. However, an alternate methodology is used because the existing PWRGALE code was developed in the 1980s for evaluation of the large PWR reactors of that time and does not address the NuScale plant design.	11.1
DSRS 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	The design basis coolant source term for NuScale is partially based on a failed fuel fraction much less than 0.25 percent, which is described in NuScale's Technical Report, Effluent Release (GALE Replacement) Methodology and Results, Revision 0 (TR-1116-52065).	11.1
DSRS 11.1, Rev 0: Coolant Source Terms	II.14	calculational technique or any source term parameter	Conforms	None.	11.1
DSRS 11.2, Rev 0: Liquid Waste Management System	II.1	Capability to Meet Dose Design Objectives	Partially Conforms	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
DSRS 11.2, Rev 0: Liquid Waste Management System	II.2	Design for Anticipated Processing Requirements	Conforms	None.	11.2.2
DSRS 11.2, Rev 0: Liquid Waste Management System	II.3	Seismic Design of Structures Housing Liquid Waste Management System Components	Conforms	None.	11.2.2
DSRS 11.2, Rev 0: Liquid Waste Management System	II.4	Provisions to Control Leakage and Facilitate Operation and Maintenance	Conforms	None.	11.2.2
DSRS 11.2, Rev 0: Liquid Waste Management System	II.5	Automatic control features	Conforms	None.	11.2 11.5 11.6

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 11.2, Rev 0: Liquid Waste Management System	II.6	Exhaust ventilation system	Conforms	None.	11.3
DSRS 11.2, Rev 0: Liquid Waste Management System	II.7	Criteria for Early Site Permit Applications	Not Applicable	This acceptance criterion is applicable only to applicants for an early site permit.	Not Applicable
DSRS 11.3, Rev 0: Gaseous Waste Management System	II.1	Capability to Meet Dose Design Objectives	Partially Conforms	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
DSRS 11.3, Rev 0: Gaseous Waste Management System	II.2	Design for Anticipated Processing Requirements	Conforms	None.	11.3.2
DSRS 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	None.	11.3.1
DSRS 11.3, Rev 0: Gaseous Waste Management System	II.4	Features to Minimize Contamination, Facilitate Decommissioning, and Minimize Generation of Radwaste	Partially Conforms	This acceptance criterion is applicable except for aspects that govern site-specific activities that are the responsibility of the COL applicant.	11.3.2
DSRS 11.3, Rev 0: Gaseous Waste Management System	II.5	Design, Testing, and Maintenance of HEPA Filters and Charcoal Adsorbers	Conforms	None.	11.3.1 11.3.4
DSRS 11.3, Rev 0: Gaseous Waste Management System	II.6	Automatic control features	Conforms	None.	11.3.7
DSRS 11.3, Rev 0: Gaseous Waste Management System	II.7	Design to Withstand Effects of Hydrogen Explosion	Conforms	None.	11.3.2
DSRS 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	None.	11.3.3
DSRS 11.3, Rev 0: Gaseous Waste Management System	II.9	Criteria for Early Site Permit Applications	Not Applicable	This acceptance criterion is applicable only to applicants for an early site permit.	Not Applicable
DSRS 11.3, Rev 0: Gaseous Waste Management System	II.10	Relevant RGs, ISG, and BTP	Partially Conforms	As described above in acceptance criteria II.1, II.3, II.4, II.5, and II.8.	As listed above in acceptance criteria II.1, II.3, II.4, II.5, and II.8.

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 11.4, Rev 0: Solid Waste Management System	II.1	Design Parameters Based on Expected Radionuclide Distributions and Concentrations	Conforms	None.	Table 11.4-1 Table 11.4-5 thru Table 11.4-9
DSRS 11.4, Rev 0: Solid Waste Management System	II.2	Sizing of Processing Equipment	Conforms	None.	11.4.2
DSRS 11.4, Rev 0: Solid Waste Management System	II.3	Liquid and Wet Waste Stabilization in Accordance with Process Control Program	Partially Conforms	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
DSRS 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	The development and implementation of a PCP is the responsibility of the COL applicant.	Not Applicable
DSRS 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	The development and implementation of a PCP and ODCM are the responsibility of the COL applicant.	Not Applicable
DSRS 11.4, Rev 0: Solid Waste Management System	II.6	Waste Containers, Shipping Casks, and Waste Packaging	Partially Conforms	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
DSRS 11.4, Rev 0: Solid Waste Management System	II.7	Onsite Waste Storage Facilities	Partially Conforms	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
DSRS 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	None.	3.8 11.4.1 Table 11.4-1 11.4.2
DSRS 11.4, Rev 0: Solid Waste Management System	II.9	Provisions to Control Leakage and Facilitate Operation and Maintenance	Partially Conforms	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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 11.4, Rev 0: Solid Waste Management System	II.10	Features to Minimize Contamination, Facilitate Decommissioning, and Minimize Generation of Radwaste	Partially Conforms	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
DSRS 11.4, Rev 0: Solid Waste Management System	II.11	Storage Facility Design for Long Term Onsite Storage (Including Appendix 11.4A)	Not Applicable	The NuScale design has no long term storage facility for solid radioactive waste. This is a COL applicant responsibility.	Not Applicable
DSRS 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	This acceptance criterion governs site-specific, programmatic aspects of the PCP development and implementation that are the responsibility of the COL applicant. This guidance is applicable to the extent necessary to ensure that the COL applicant referencing the certified design can satisfy the guidance.	11.4.2 11.4.3
DSRS 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	This acceptance criterion governs site-specific, programmatic aspects of the PCP development and implementation that are the responsibility of the COL applicant. This guidance is applicable to the extent necessary to ensure that the COL applicant referencing the certified design can satisfy the guidance.	11.4.1 11.4.2
DSRS 11.4, Rev 0: Solid Waste Management System	II.14	Processing and Disposing of Mixed Wastes	Partially Conforms	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. This guidance is applicable to the extent necessary to ensure that the COL applicant referencing the certified design can satisfy the guidance contained therein.	11.4.2
DSRS 11.4, Rev 0: Solid Waste Management System	II.15	All Effluent Releases Associated with Operation of the SWMS	Partially Conforms	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 11.4, Rev 0: Solid Waste Management System	II.16	Operational Programs	Not Applicable	The information governed by this acceptance criterion is site-specific and is the responsibility of the COL applicant.	Not Applicable
DSRS 11.4, Rev 0: Solid Waste Management System	II.17	Automatic control features	Not Applicable		Not Applicable
DSRS 11.4, Rev 0: Solid Waste Management System	II.18	Design of exhaust ventilation systems	Conforms	None.	11.4.2
DSRS 11.4, Rev 0: Solid Waste Management System	II.19	Seismic design of structures housing SWMS	Conforms	None.	11.4.1
DSRS 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	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
DSRS 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	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
DSRS 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	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
DSRS 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	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
DSRS 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	This acceptance criterion is applicable except for the administrative and procedural controls that are the COL applicant's responsibility.	11.5

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 11.5, Rev 0: Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	II.6	Operational Programs	Not Applicable	The information governed by this acceptance criterion is site-specific and is the responsibility of the COL applicant.	Not Applicable
DSRS 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	None.	11.5
DSRS 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	None.	11.5 11.6
DSRS 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	None.	11.5 11.6
DSRS 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	None.	11.6 12.3
DSRS 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	This acceptance criterion is applicable except for aspects of its subtier regulation 10 CFR 50.34(f)(2)(xxvi) that address 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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 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	None.	9.3.2 11.6
DSRS 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	This acceptance criterion is applicable except for site-specific, programmatic aspects regarding software reviews, which are the COL applicant's responsibility.	11.6 Ch 17
DSRS 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	The RETS/SREC and ODCM are COL applicant responsibilities.	Not Applicable
DSRS 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	See RG 4.21 in Table 1.9-2.	11.5 11.6 12.3.6
DSRS 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	None.	5.2.5 9.3.4 9.3.6 11.5 11.6

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 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	None.	11.5 11.6 12.3
DSRS 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	None.	13.4 14.2 14.3 Ch 16
DSRS 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	None.	11.5 11.6 12.3
DSRS 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	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	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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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	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	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	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
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	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	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	None.	11.3.3

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
BTP 11-6, Rev 4: Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	B.1	Failure Mechanism and Radioactivity Releases	Not Applicable	COL applicant.	Not Applicable
BTP 11-6, Rev. 4: Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	B.2	Mitigating Design Features	Not Applicable	COL applicant.	Not Applicable
BTP 11-6, Rev 4: Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	B.3	Radioactive Source Term	Partially conforms	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	The development of representative site parameters under this acceptance criterion (and SRP Section 2.4.13) is site-specific and 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	The development of representative site parameters under this acceptance criterion is site-specific and applicable to COL applicant.	Not Applicable
BTP 11-6, Rev 4: Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	B.6	SRP Dose Acceptance Criteria	Not Applicable	This acceptance criterion is the responsibility of the COL applicant.	Not Applicable
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	Compliance with this guidance is the responsibility of the COL applicant.	Not Applicable
SRP 12.1, Rev 4: Assuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable	II.1	Policy Considerations	Partially Conforms	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	None.	12.1.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 12.1, Rev 4: Assuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable	II.3	Operational Considerations	Not Applicable	This guidance governs site-specific operational programs, plans, and procedures that are the responsibility of the COL applicant.	Not Applicable
SRP 12.1, Rev 4: Assuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable	II.4	Radiation Protection Considerations	Not Applicable	See comment above for Acceptance Criterion II.3.	Not Applicable
DSRS 12.2, Rev 0: Radiation Sources	II.1	RG 1.183	Partially Conforms	See RG 1.183 in Table 1.9-2.	12.2.1
DSRS 12.2, Rev 0: Radiation Sources	II.2	RG 1.7	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
DSRS 12.2, Rev 0: Radiation Sources	II.3	RG 1.112	Partially Conforms	See RG 1.112 in Table 1.9-2.	12.2.1
DSRS 12.2, Rev 0: Radiation Sources	II.4	NUREG-0737, Task Action Plan Item II.B.2	Conforms	None.	12.3 12.4
DSRS 12.2, Rev 0: Radiation Sources	II.5	ANSI/ANS Standard 18.1	Conforms	None.	11.1
DSRS 12.2, Rev 0: Radiation Sources	II.6	Radiation Sources for 10 CFR 50.49 (EQ)	Conforms	None.	12.2 Ch 3
DSRS 12.2, Rev 0: Radiation Sources	II.7	RG 1.143	Partially Conforms	See RG 1.143 in Table 1.9-2.	11.2 11.3 11.4 11.6
DSRS 12.2, Rev 0: Radiation Sources	II.8	RG 1.26, RG 1.29 and RG 1.117	Conforms	None.	3.2
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.1	RG 1.7	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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.2	RG 1.52	Not Applicable	See RG 1.52 in Table 1.9-2.	Not Applicable
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.3	RG 1.69	Partially Conforms	See RG 1.69 in Table 1.9-2.	12.3.2
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.4	RG 1.97	Partially Conforms	See RG 1.97 in Table 1.9-2.	7.2.13 12.3.4
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.5	RG 1.183	Partially Conforms	See RG 1.183 in Table 1.9-2.	12.2
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.6	RG 8.2	Not Applicable	See RG 8.2 in Table 1.9-2.	Not Applicable
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.7	RG 8.8	Partially Conforms	See RG 8.8 in Table 1.9-2.	12.3.1
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.8	RG 8.10	Not Applicable	See RG 8.10 in Table 1.9-2.	Not Applicable
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.9	RG 8.15	Not Applicable	See RG 8.15 in Table 1.9-2.	Not Applicable
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.10	RG 8.19	Conforms	None.	12.4
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.11	RG 8.25	Not Applicable	See RG 8.25 in Table 1.9-2.	Not Applicable
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.12	RG 8.38	Partially Conforms	See RG 8.38 in Table 1.9-2.	12.3.1
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.13	ANSI/ANS/HPSSC-6.8.1-1981	Conforms	None.	12.3.4

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.14	ANSI/HPS N13.1-2011	Conforms	None.	12.3.4
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.15	ANSI/ANS-6.4-2006	Conforms	None.	12.3.2
DSRS 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	The portion of this guidance that pertains to the design phase is applicable to the DCA.	12.3.4
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.17	RG 1.140	Partially Conforms	See RG 1.140 in Table 1.9-2.	12.3.3
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.18	RG 1.89	Partially Conforms	See RG 1.89 in Table 1.9-2.	3.11
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.19	RG 4.21	Partially Conforms	See RG 4.21 in Table 1.9-2.	12.3.6
DSRS 12.3-12.4, Radiation Protection Design Features	II.20	RG 1.45	Partially Conforms	See RG 1.45 in Table 1.9-2.	5.2.5
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.21	NEI 97-06	Conforms	None.	Ch 5
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.22	RG 1.143	Partially Conforms	See RG 1.143 in Table 1.9-2.	Ch 11
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.23	BTP 11-3 and SECY-94-198	Partially Conforms	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
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.24	RG 1.97	Partially Conforms	See RG 1.97 in Table 1.9-2.	12.3.4

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.25	RG 1.12	Partially Conforms	See RG 1.12 in Table 1.9-2.	12.3.1
DSRS 12.5, Rev 0: Operational Radiation Protection Program	All	Various	Not 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.	Not Applicable
SRP 13.1.1, Rev 5: Management and Technical Support Organization	All	General and Specific Requirements	Not Applicable	COL applicant.	Not Applicable
SRP 13.1.2 - 13.1.3, Rev 6: Operating Organization	All	Operating Organization	Not Applicable	COL applicant.	Not Applicable
SRP 13.2.1, Rev 3: Reactor Operator Requalification Program; Reactor Operator Training	All	General and Specific Requirements	Not Applicable	COL applicant.	Not Applicable
SRP 13.2.2, Rev 3: Non-Licensed Plant Staff Training	All	Various	Not Applicable	COL applicant.	Not Applicable
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	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.2	Onsite and Offsite Emergency Response Plans	Not Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.3	Emergency Classification and Action Level Scheme	Not Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.4	Meteorological Criteria	Not Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.5	Upgrading Emergency Response Facilities	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	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.7	Protective Action Recommendations	Not Applicable	COL applicant.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 13.3, Rev 3: Emergency Planning	II.8	Alternatives to NUREG-0654/FEMA-REP-1, Rev 1,	Not Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.9	State, Tribal, and Local Government Planning and Preparedness	Not Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.10	Emergency Planning Zones	Not Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.11	Evacuation Time Estimates	Not Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.12	Emergency Response Data System	Partially Conforms	The NuScale design includes an emergency response data system. Site-specific aspects are the responsibility of the COL applicant that references the NuScale certified design.	13.3
SRP 13.3, Rev 3: Emergency Planning	II.13	Acceptability of Emergency Plans	Not 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	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	Guidance applies to early site permit applicants.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.16	Early Site Permit Criteria - Preliminary Analysis of Evacuation Times	Not Applicable	Guidance applies to early site permit applicants.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.17	Physical Characteristics Unique to Proposed Site	Not Applicable	Guidance applies to early site permit applicants.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.18	Copies of Letters of Agreement or Other Certifications	Not Applicable	Guidance applies to early site permit applicants.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.19	Emergency Preparedness Information and Plans Associated with Early Site Permit Application	Not Applicable	Guidance applies to early site permit applicants.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.20	Complete and Integrated Emergency Plans Associated with Early Site Permit Application	Not Applicable	Guidance applies to early site permit applicants.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.21	ITAAC Associated with Early Site Permit Application	Not Applicable	Guidance applies to early site permit applicants.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.22	ITAAC Associated with Design Certification Application	Not Applicable	Emergency planning ITAAC are not part of the NuScale DCA.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 13.3, Rev 3: Emergency Planning	II.23	ITAAC Associated with Combined License Application	Not Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.24	Generic Emergency Planning ITAAC	Not Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.25	Design and Implementation of Emergency Response Facilities	Partially Conforms	The NuScale design includes a technical support center. The operational support center and the emergency operations facility are the responsibility of the COL applicant that references the NuScale certified design.	13.3
SRP 13.3, Rev 3: Emergency Planning	II.26	Safety Parameter Display System	Conforms	Safety parameter displays are provided in the technical support center. The emergency operations facility is the responsibility of the COL applicant that references the NuScale design certification.	13.3
SRP 13.3, Rev 3: Emergency Planning	II.27	Reactor Coolant System and Containment Sampling	Departure	The NuScale design supports an exemption from 10 CFR 50.34(f)(2)(viii).	9.3.2
SRP 13.3, Rev 3: Emergency Planning	II.28	Containment Monitoring and Continuous Sampling from Potential Accident Release Points	Partially Conforms	Programmatic aspects of containment and effluent monitoring are the responsibility of the COL applicant.	9.3.2, 11.5
SRP 13.3, Rev 3: Emergency Planning	II.29	NRC Notifications and Communications	Not 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	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.31	Operational Programs	Not 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	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	COL applicant.	Not Applicable
SRP 13.5.1.2, Draft Rev 0: Administrative Procedures - Initial Test Program	All	Various	Not Applicable	Draft SRP section was never finalized. Content was subsumed into SRP Section 14.2.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 13.5.2.1, Rev 2: Operating and Emergency Operating Procedures	All	Various	Not Applicable	COL applicant.	Not Applicable
SRP 13.5.2.2, Draft Rev 0: Maintenance and Other Operating Procedures	All	Various	Not Applicable	NRC never finalized guidance in this SRP. Instead, applicable guidance was relocated to SRP 17.5.	Not Applicable
SRP 13.6.1, Rev 1: Physical Security - Combined License and Operating Reactors	All	Various	Not Applicable	COL applicant.	Not Applicable
SRP 13.6.2, Rev 2: Physical Security - Design Certification	All	Various	Conforms	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	ESP applicant.	Not Applicable
SRP 13.6.4, Rev 1: Access Authorization	II (no number)	10 CFR 73.56	Not Applicable	COL applicant.	Not Applicable
SRP 13.6.6, Rev 0: Cyber Security Plan	All	Various	Not Applicable	COL applicant.	Not Applicable
SRP 13.7.1, Rev 0: Fitness for Duty (Operational)	All	Various	Not Applicable	COL applicant.	Not Applicable
SRP 13.7.2, Rev 0: Fitness for Duty (Construction)	All	Various	Not Applicable	COL applicant.	Not Applicable
DSRS 14.2, Rev 0: Initial Plant Test Program - Design Certification and New COL applicants	II.1	Summary of Test Program and Objectives	Conforms	None.	14.2
DSRS 14.2, Rev 0: Initial Plant Test Program - Design Certification and New COL applicants	II.2	Test Programs Conformance with Regulatory Guides	Conforms	None.	14.2
DSRS 14.2, Rev 0: Initial Plant Test Program - Design Certification and New COL applicants	II.3	Initial Test Program Administrative Procedures	Partially Conforms	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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 14.2, Rev 0: Initial Plant Test Program - Design Certification and New COL applicants	II.4	Initial Startup Tests	Partially Conforms	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
DSRS 14.2, Rev 0: Initial Plant Test Program - Design Certification and New COL applicants	II.5	Individual Test Descriptions/ Abstracts	Conforms	None.	14.2
DSRS 14.2, Rev 0: Initial Plant Test Program - Design Certification and New COL applicants	II.6	Initial Test Program Acceptance Criteria	Partially Conforms	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	This SRP section is applicable only to extended power uprate license amendment requests.	Not Applicable
SRP 14.3, (March 2007): Inspections, Tests, Analyses, and Acceptance Criteria	II.1	Acceptability of the Scope of ITAAC	Partially Conforms	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	None.	14.3
SRP 14.3.2, Rev 0: Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria	All	Various	Not 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	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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 14.3.5, Rev 0: Instrumentation and Controls - Inspections, Tests, Analyses, and Acceptance Criteria	All	Various	Not 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	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	Methodology for developing ITAAC is provided in SRP 14.3.	Not Applicable
SRP 14.3.8, Rev 0: Radiation Protection - Inspections, Tests, Analyses, and Acceptance Criteria	All	Various	Not 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	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	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	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	Partially Conforms	The COL applicant addresses Physical Security Hardware ITAAC outside of the nuclear island and structures.	14.3.12
DSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.1	Categorization of Transients and Accidents	Conforms	None.	15.0

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.2	Categorization According to Frequency of Occurrence	Partially Conforms	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
DSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.3	Categorization According to Type	Conforms	None.	15.0
DSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.4.A	Analysis Acceptance Criteria for AOOs	Conforms	None.	15.0.0
DSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.4.B	Analysis Acceptance Criteria for IEs and Postulated Accidents	Partially Conforms	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 degrees F.	15.0.0
DSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.5	Plant Characteristics Considered in the Safety Evaluation	Conforms	None.	15.0
DSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.6	Assumed Protection and Safety Systems Actions	Conforms	None.	15.0
DSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.7	Evaluation of Individual Initiating Events	Conforms	None.	15.0
DSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.8.A	Identification of Causes and Frequency Classification	Conforms	None.	15.0
DSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.8.B	Sequence of Events and Systems Operation	Partially Conforms	This acceptance criterion is applicable except for Item B.vi, which is applicable to COL applicants.	15.0

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 15.0, Rev 0: Introduction - Transient and Accident Analyses	I.8.C	Core, System, and Barrier Performance	Partially Conforms	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
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	The NuScale design uses a modified version of the alternative source term (AST) methodology to evaluate radiological consequences of accidents.	Not Applicable
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	The NuScale design utilizes a modified version of the AST methodology to evaluate radiological consequences of accidents.	Not Applicable
SRP 15.0.2, (March 2007): Review of Transient and Accident Analysis Methods	II.1	Evaluation Model	Departure	The NuScale design supports an exemption from selected portions of 10 CFR 50 Appendix K. 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	None.	15.0.2
SRP 15.0.2, (March 2007): Review of Transient and Accident Analysis Methods	II.3	Code Assessment	Departure	The NuScale design supports an exemption from selected portions of 10 CFR 50 Appendix K. 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	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	None.	15.0.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 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	None.	15.0.3
DSRS 15.0.3, Rev 0: Design Basis Accident Radiological Consequence Analyses for the NuScale SMR Design	II.2	Control Room Radiological Habitability	Conforms	None.	6.4 9.4 13.3 15.0.3
DSRS 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	Dose acceptance criterion met for TSC when AC power is available. TSC function is transferred to the main control room when AC is not available.	15.0.3
DSRS 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	None.	15.1.1-15.1.4
DSRS 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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 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	None.	15.1.1-15.1.4
DSRS 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	NuScale has determined that critical heat flux more accurately describes plant phenomena than departure from nucleate boiling.	15.1.1-15.1.4
DSRS 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	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
DSRS 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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 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	None.	15.1.1-15.1.4
DSRS 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	None.	15.1.1-15.1.4
DSRS 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	None.	15.1.1-15.1.4
DSRS 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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 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	None.	15.1.1-15.1.4
DSRS 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	None.	15.1.5
DSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.2 Specific Criteria	Evaluation of Core Damage Potential	Conforms	NuScale has determined that critical heat flux more accurately describes plant phenomena than departure from nucleate boiling.	15.1.5
DSRS 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	None.	15.1.5
DSRS 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	None.	15.1.5
DSRS 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	None.	15.1.5
DSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.2 Assumptions	Loss of Offsite Power	Conforms	None.	15.1.5
DSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.3 Assumptions	Postulated Steam Line Break Effects	Conforms	None.	15.1.5
DSRS 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	None.	15.1.5

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.5 Assumptions	Maximum-Worth Rod Fully Withdrawn	Conforms	None.	15.1.5
DSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.6 Assumptions	Core Burnup	Conforms	None.	15.1.5
DSRS 15.1.5, Rev 0: Steam System Piping Failures Inside and Outside of Containment	II.7 Assumptions	Initial Core Flow	Conforms	NuScale has determined that critical heat flux more accurately describes plant phenomena than departure from nucleate boiling.	15.1.5
DSRS 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	None.	15.1.5
DSRS 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	None.	15.1.5
DSRS 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	Operator Action is not required to mitigate the consequences of a steam line break.	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	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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	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 and Items 1 and 2 on Page 15.1.5-11, Exposure Guidelines for Calculated Doses	Partially Conforms	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
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	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	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
DSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.1 Specific Criteria	Reactor Coolant Pressure	Conforms	None.	15.1.6
DSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.2 Specific Criteria	Cladding Integrity	Conforms	None.	15.1.6
DSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.3 Specific Criteria	AOOs Should Not Generate More Serious Condition	Conforms	None.	15.1.6
DSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.4 Specific Criteria	Instruments Spans and Setpoints use RG 1.105	Conforms	None.	15.1.6
DSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.5 Specific Criteria	Identify Limiting Single Failure	Conforms	None.	15.1.6

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.1 Analytical Parameters	Initial Power Level is 102%	Conforms	None.	15.1.6
DSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.2 Analytical Parameters	Conservative Scram Used	Conforms	None.	15.1.6
DSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.3 Analytical Parameters	Core Burnup	Conforms	None.	15.1.6
DSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.4 Analytical Parameters	Maximize Heat Transfer from RCS to Containment and Reactor Pool	Conforms	None.	15.1.6
DSRS 15.1.6, Rev 0: Loss of Containment Vacuum	II.5 Analytical Parameters	Setpoint Inaccuracies use Guidance in RG 1.105	Conforms	None.	15.1.6
DSRS 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	None.	15.2.1-15.2.5
DSRS 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	The NuScale design does not have a steam pressure regulator.	15.2.1-15.2.5
DSRS 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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 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	The NuScale design does not have a steam pressure regulator.	15.2.1-15.2.5
DSRS 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	None.	15.2.1-15.2.5
DSRS 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	None.	15.2.1-15.2.5
DSRS 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	None.	15.2.1-15.2.5
DSRS 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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 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	None.	15.2.1-15.2.5
DSRS 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	None.	15.2.1-15.2.5
DSRS 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	None.	15.2.1-15.2.5
DSRS 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	None.	15.2.1-15.2.5
DSRS 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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 15.2.6, Rev 0: Loss of Nonemergency AC Power to the Station Auxiliaries	II.1	Reactor Coolant and Main Steam System Pressures	Conforms	None.	15.2.6
DSRS 15.2.6, Rev 0: Loss of Nonemergency AC Power to the Station Auxiliaries	II.2	Fuel Cladding Integrity	Conforms	None.	15.2.6
DSRS 15.2.6, Rev 0: Loss of Nonemergency AC Power to the Station Auxiliaries	II.3	Incidents of Moderate Frequency	Conforms	NuScale categorizes events as AOO and IE.	15.2.6
DSRS 15.2.6, Rev 0: Loss of Nonemergency AC Power to the Station Auxiliaries	II.4	Requirements of GDC 10 and GDC 15	Conforms	None.	15.2.6
DSRS 15.2.6, Rev 0: Loss of Nonemergency AC Power to the Station Auxiliaries	II.5	Most Limiting Plant System Single Failure	Conforms	None.	15.2.6
DSRS 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	None.	15.2.6
DSRS 15.2.7, Rev 0: Loss of Normal Feedwater Flow	II.1	Fuel and System Pressure Parameters met	Conforms	None.	15.2.7
DSRS 15.2.7, Rev 0: Loss of Normal Feedwater Flow	II.2	Events of Moderate Frequency	Conforms	NuScale categorizes events as AOO and IE.	15.2.7
DSRS 15.2.7, Rev 0: Loss of Normal Feedwater Flow	II.3	Analytical Model and Methods	Conforms	None.	15.2.7
DSRS 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	None.	15.2.8
DSRS 15.2.8, Rev 0: Feedwater System Pipe Break Inside and Outside Containment (PWR)	II.2	Evaluation of Core Damage Potential	Conforms	For slower reactivity insertions, NuScale uses a heat generation rate limit to ensure that fuel centerline melting limits are met.	15.2.8
DSRS 15.2.8, Rev 0: Feedwater System Pipe Break Inside and Outside Containment (PWR)	II.3	Calculated Site Boundary Doses	Conforms	None.	15.2.8

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 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	None.	15.2.8
DSRS 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	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	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	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	Critical heat flux (CHF) is more appropriate terminology for NuScale phenomena than departure from nucleate boiling (DNBR).	15.4.1
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	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	The SRP states that this criterion applies to BWRs. NuScale uses the 95/95 MCHFR approach to ensure no cladding or fuel failures.	Not Applicable
SRP 15.4.2, Rev 3: Uncontrolled Control Rod Assembly Withdrawal at Power	II.1.A	Thermal Margin Limits	Conforms	CHF is more appropriate terminology for NuScale phenomenon than DNBR.	15.4.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 15.4.2, Rev 3: Uncontrolled Control Rod Assembly Withdrawal at Power	II.1.B	Fuel Centerline Temperatures	Conforms	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	The SRP states that this criterion applies to BWRs. NuScale uses the 95/95 MCHFR approach to ensure no cladding or fuel failures.	Not Applicable
SRP 15.4.3, Rev 3: Control Rod Misoperation (System Malfunction or Operator Error)	II.1	Thermal Margin Limits	Conforms	CHF is more appropriate terminology for NuScale phenomenon than DNBR.	15.4.3
SRP 15.4.3, Rev 3: Control Rod Misoperation (System Malfunction or Operator Error)	II.2	Fuel Centerline Temperatures	Conforms	For slower reactivity insertions, NuScale uses a heat generation rate limit to meet fuel centerline melting limits.	15.4.3
SRP 15.4.3, Rev 3: Control Rod Misoperation (System Malfunction or Operator Error)	II.3	Uniform Cladding Strain	Conforms	None.	15.4.3
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	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 design does not require or include reactor coolant pumps. 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 DSRS 15.4.6 - Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR).	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	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	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 design does not require or include reactor coolant pumps. 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 DSRS 15.4.6 - Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR).	Not Applicable
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	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 design does not require or include reactor coolant pumps. 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 DSRS 15.4.6 - Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR).	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	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	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 design does not require or include reactor coolant pumps. 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 DSRS 15.4.6 - Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR).	Not Applicable
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	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 design does not require or include reactor coolant pumps. 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 DSRS 15.4.6 - Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR).	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	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	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 design does not require or include reactor coolant pumps. 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 DSRS 15.4.6 - Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR).	Not Applicable
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	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	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	None.	15.4.6
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	Operator action is not required to mitigate an inadvertent boron dilution event.	Not Applicable
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	None.	15.4.6

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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	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	Safety analysis demonstrates that there are no fuel failures.	15.4.7
SRP 15.4.8, Rev 3: Spectrum of Rod Ejection Accidents (PWR)	II.1	Availability of Monitoring Instrumentation	Conforms	None.	15.4.8
SRP 15.4.8, Rev 3: Spectrum of Rod Ejection Accidents (PWR)	II.2	Effects of Postulated Reactivity Accidents	Conforms	None.	15.4.8
SRP 15.4.8, Rev 3: Spectrum of Rod Ejection Accidents (PWR)	II.3	Radiation Dose Limits	Conforms	None.	15.4.8
SRP 15.4.8.A, Rev 1: Radiological Consequences of a Control Rod Ejection Accident (PWR)	All	Various	Partially Conforms	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, apply 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	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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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	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
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	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	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	This SRP section and its acceptance criteria are applicable only to BWRs.	Not Applicable
DSRS 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	None.	15.0 15.5.1-15.5.2
DSRS 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	None.	15.5.1-15.5.2
DSRS 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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 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	None.	15.5.1-15.5.2
DSRS 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	None.	15.5.1-15.5.2
DSRS 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	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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	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	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	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	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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	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	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	This SRP section and its acceptance criteria are applicable only to BWRs.	Not Applicable
DSRS 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	Departure	The NuScale design supports an exemption from selected portions of 10 CFR 50 Appendix K. 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
DSRS 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	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.	15.6.5
DSRS 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	None.	15.6.5

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 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	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	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	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
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	None.	15.0.3 15.6.5

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	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.2	Calculation of Radiological Consequences	Partially Conforms	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	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
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	This SRP section and its acceptance criteria are applicable only to BWRs.	Not Applicable
DSRS 15.6.6, Rev 0: Inadvertent Operation of ECCS	II.1	RCS pressure below 110 percent design value.	Conforms	None.	15.6.6
DSRS 15.6.6, Rev 0: Inadvertent Operation of ECCS	II.2	Maintain minimum DNBR.	Conforms	NuScale evaluated CHF as it is more appropriate than DNBR for the NuScale design.	15.6.6

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 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	None.	15.6.6
SRP 15.7.3, Rev 2: Radioactive Release from a Subsystem or Component	All	Various	Partially Conforms	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	This acceptance criterion specifies radiological analysis acceptance criteria that are superseded by SRP Section 15.0.3.	Not Applicable
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	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 SSC. 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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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	The NuScale design does not rely on ESF ventilation systems to mitigate the consequences of a design basis accident. Nonsafety-related normal ventilation systems provide atmosphere cleanup capability, as necessary, that meets the design, testing, and maintenance guidelines 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 receive no credit in the determination of the radiological consequences of an accident.	Not Applicable
SRP 15.7.4, Rev 1: Radiological Consequences of Fuel Handling Accidents	II.5	Radiation Detection in Containment	Partially Conforms	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 for fuel handling operations inside containment are not relevant 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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 15.7.5, Rev 2: Spent Fuel Cask Drop Accidents	All	Various	Partially Conforms	One of the principal functions of the NuScale reactor building crane (RBC) is to move 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 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 15.7.5, Rev 2: Spent Fuel Cask Drop Accidents	II.1	Acceptability of Site and Dose Mitigating ESF Systems	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 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	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 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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 15.7.5, Rev 2: Spent Fuel Cask Drop Accidents	II.3	Dose Model and Modeling Assumptions	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 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.4	ESF Grade Atmosphere Clean-Up System in Spent Fuel Storage Area	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 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	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 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	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	NuScale is characterized as an evolutionary plant (See the acceptance criteria in II.3).	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 15.8, Rev 2: Anticipated Transients without Scram	II.3.A.i	Provide a diverse scram system	Partially Conforms	The NuScale design relies on diversity within the module protection system (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
SRP 15.8, Rev 2: Anticipated Transients without Scram	II.3.A.ii	or, Demonstrate that the ATWS event consequences are acceptable	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	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	NuScale conforms to the second criterion option of reducing the probability of a failure to scram. This is achieved with a diverse RPS instead of a diverse scram system as discussed above.	15.8
DSRS 15.9.A, Rev 0: Thermal Hydraulic Stability Review Responsibilities	II.1	No requirements	-	None.	-
DSRS 15.9.A, Rev 0: Thermal Hydraulic Stability Review Responsibilities	II.2	Meeting Requirements of GDC 12	Conforms	None.	4.4.7
DSRS 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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 15.9.A, Rev 0: Thermal Hydraulic Stability Review Responsibilities	II.4	Detect and Suppress Method: Exclusion zone and buffer region methodology	Not Applicable	Exclusion zone option is not used in the NuScale design. Reactor trip signals prevent violation of CHF limits before unstable flow oscillations can develop. Protective action occurs prior to development of oscillation.	Not Applicable
DSRS 15.9.A, Rev 0: Thermal Hydraulic Stability Review Responsibilities	II.5	Detect and Suppress Method Trip of reactor before SAFDL violation	Partially Conforms	Existing reactor trip signals provide an exclusion zone that prevents violation of SAFDL limits from other causes, which is already more limiting than the exclusion zone needed to preclude flow instabilities.	4.4.4 4.4.7
DSRS 15.9.A, Rev 0: Thermal Hydraulic Stability Review Responsibilities	II.6	Backup options if licensing solutions declared inoperable	Not Applicable	Detect and Suppress options are not employed. Existing technical specifications for RTS provide controls on allowable unavailabilities of protective trips. Backup options are not required.	Not Applicable
DSRS 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	RTS system trips reactor prior to conditions that could initiate flow instabilities. Stabilities are not detected and suppressed.	4.4.7
DSRS 15.9.A, Rev 0: Thermal Hydraulic Stability Review Responsibilities	II.8	Detect and Suppress system to monitor process variables and systems.	Not Applicable	RTS system trips reactor prior to conditions that could initiate flow instabilities. Stabilities are not detected and suppressed.	Not Applicable
DSRS 15.9.A, Rev 0: Thermal Hydraulic Stability Review Responsibilities	II.9	Stability-related instrumentation functionality should be demonstrated by analysis.	Conforms	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
DSRS 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	None.	4.4.4 4.4.7 15.9.A
DSRS 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	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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 16.0, Rev. 0: Technical Specifications	All (No Number)	Acceptance Criteria for Technical Specifications	Partially Conforms	This DSRS 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. Notwithstanding the above, pursuant to 10 CFR 52.47(a)(11) and consistent with DSRS 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 DSRS - 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 used to draft the specifications. 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 DSRS 16.0, technical justification for such differences is provided.	Ch 16

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 16.1, Rev 1: Risk-Informed Decision Making: Technical Specifications	II.1	Traditional Engineering Guidelines	Partially Conforms	This guidance is for revisions being made to existing technical specifications (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	This guidance applies to revisions being made to existing TS, including deviation from generic or 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	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 NQA-1) 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
SRP 17.2, Rev 2: Quality Assurance During the Operations Phase	All	Various	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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 17.3, Rev 0: Quality Assurance Program Description	All	Various	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	None.	17.4
SRP 17.4, Rev 1: Reliability Assurance Program (RAP)	II.B	COL Applicant	Not 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	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	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
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	None.	17.5

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.D	Procurement Document Control	Conforms	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	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	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	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	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.I	Control of Special Processes	Not 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	The provisions specific to inservice, modification, etc. are the responsibility of 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) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.K	Test Control	Conforms	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	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	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	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.O	Nonconforming Materials, Parts, or Components	Conforms	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	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	None.	17.5

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New COL applicants	II.R	Audits	Conforms	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	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	None.	17.5
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	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	None.	17.5
SRP 17.6, Rev 2: Maintenance Rule	All	Various	Not Applicable	This SRP section and its acceptance criteria govern a site-specific operational program that is the responsibility of the COL applicant.	Not Applicable
SRP 18.0, Rev 2: Human Factors Engineering	II.A	Review of the HFE Aspects of a New Plant	Conforms	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	This acceptance criterion is applicable to existing reactor licensees that request NRC approval of control room modifications.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 18.0, Rev 2: Human Factors Engineering	II.C	Review of the HFE Aspects of Modifications Affecting Risk Important Human Actions	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	This appendix supersedes DI&C ISG-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.2.B	Review Criteria for Phase 2 (Preliminary Validation)	Conforms	This appendix supersedes DI&C ISG-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	This appendix supersedes DI&C ISG-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	This appendix supersedes DI&C ISG-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	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 to PRAs used by a licensee to support license amendments for an operating reactor.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 19.2, Initial Issuance: Review of Risk Information Used to Support Permanent PlantSpecific Changes to the Licensing Basis: General Guidance	All	Various	Not Applicable	Applicable to licensees, plant-specific proposals for changes to the licensing basis.	Not Applicable
SRP 19.3, Rev 0: Regulatory Treatment of Non-Safety Systems for Passive Advanced Light Water Reactors	All	Various	Conforms	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 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 by NEI 06-12 and includes a statement that 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	None.	19.5

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

ISG Section/ Title	AC	AC Title / Description	Conformance Status	Comments	Section
DC/COL-ISG-1: Seismic Issues of High Frequency Ground Motion	1	Seismic Issues addressed in this Interim Staff Guidance	-	This section points out to the guidance provided in Sections 2, 3, 4, and 5.	3.7
DC/COL-ISG-1	2	Ground Motion Definitions	Conforms	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	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	The NuScale 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	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	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	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
DC/COL-ISG-4: Definition of Construction and on Limited Work Authorizations	All	Various	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

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

ISG Section/ Title	AC	AC Title / Description	Conformance Status	Comments	Section
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	The NuScale design is similar to large PWRs in the existing fleet for effluent release calculations. However, 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 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	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. 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	All	Normal and Extreme Winter Precipitation Events and their Resulting Live Roof Loads	Conforms	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 2.3.	2.3 3.4 3.8
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	None.	Ch 16
DC/COL-ISG-8	para 2-4 (p. 4 & 5)	Second, third, and fourth paragraphs under heading Final Interim Staff Guidance, specifying compliance options for COL applicants	Not 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	This guidance is applicable except for aspects that are BWR-specific.	3.9.5

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

ISG Section/ Title	AC	AC Title / Description	Conformance Status	Comments	Section
DC/COL-ISG-11: Finalizing Licensing-basis Information	All	Licensing-Basis Information Freeze Point; Changes That Should Not be Considered for Deferral	Partially Conforms	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	Partially Conforms	Site-specific aspects that are the responsibility of the COL applicant.	11.2.3
DC/COL-ISG-13	2	Mitigating Design Features	Partially Conforms	This guidance is applicable except for site-specific aspects that are the responsibility of the COL applicant.	11.2.3
DC/COL-ISG-13	3	Radioactive Source Term (Including Attachment A)	Partially Conforms	Site-specific aspects 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	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	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	Partially Conforms	Site-specific aspects are the responsibility of the COL applicant.	11.2.3
DC/COL-ISG-13	7	Specifications on Tank Waste Radioactivity Concentration Levels	Partially Conforms	Site-specific aspects (e.g., development and implementation of the ODCM) are the responsibility of the COL applicant.	11.2.2
DC/COL-ISG-13	8	Evaluation Findings for Combined License Reviews	Not Applicable	This acceptance criterion is explicitly directed towards the review of combined license applications.	Not Applicable

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

ISG Section/ Title	AC	AC Title / Description	Conformance Status	Comments	Section
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	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	This guidance is for COL applicants.	Not Applicable
ESP/DC/COL-ISG-15	No Num (p11-23)	Anticipated NRC Revisions of NUREG0800, SRP Chapter 1.0	Partially Conforms	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 throughout the FSAR.	Ch 1
DC/COL-ISG-16: Compliance with 10 CFR 50.54(hh)(2) and 10 CFR 52.80(d)	All	-	Not Applicable	Requirements in 10 CFR 50.54(hh)(2) were moved to 10 CFR 50.155(b)(2).10 CFR 50.54(hh)(2) is not applicable to design certification applicants; however 10 CFR 52.80(d) requires COL applicants to include a description of the equipment upon which mitigating strategies rely to comply with 10 CFR 50.155(b)(2) to maintain or restore core cooling, containment, and SFP cooling capabilities.	Not Applicable
DC/COL-ISG-17: Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Interaction Analyses	All	-	Not Applicable	This ISG is applicable to the review of seismic design information submitted to support combined license (COL) applications.	Not Applicable
DC/COL-ISG-19: Gas Accumulation Issues in Safety Related Systems	All	Various	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). 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

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

ISG Section/ Title	AC	AC Title / Description	Conformance Status	Comments	Section
DC/COL-ISG-20: Implementation of a Probabilistic Risk Assessment-Based Seismic Margin Analysis for New Reactors	All	Various	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
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	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 instead of 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	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	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	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	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	Comments	Section
DC/COL-ISG-27: Specific Environmental Guidance for Light Water Small Modular Reactor	All	Various	Not 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	Provides guidance for DC and COL applicants to conform to PRA Standard.	19.1
Digital I&C-ISG-01: Cyber Security	5.	Staff Position	Not Applicable		Not Applicable
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)	Not Applicable	Digital I&C-ISG-02 is not applicable to the NuScale DCA. See DSRS 7.1.5 in Table 1.9-3 which provides information on the Diversity and Defense-in-Depth review.	Not Applicable
Digital I&C-ISG-02	3	BTP 7-19 Position 4 Challenges - Staff Position (Page 6)	Not Applicable	Digital I&C-ISG-02 is not applicable to the NuScale DCA. See DSRS 7.1.5 in Table 1.9-3 which provides information on the Diversity and Defense-in-Depth review.	Not Applicable
Digital I&C-ISG-02	4	Effects of Common Cause Failure (CCF) - Staff Position (Pages 8 and 9)	Not Applicable	Digital I&C-ISG-02 is not applicable to the NuScale DCA. See DSRS 7.1.5 in Table 1.9-3 which provides information on the Diversity and Defense-in-Depth review.	Not Applicable
Digital I&C-ISG-02	6	Echelons of Defense - Staff Position (Page 12)	Not Applicable	Digital I&C-ISG-02 is not applicable to the NuScale application. See DSRS 7.1.5 in Table 1.9-3 which provides information on Diversity and Defense-in-Depth review.	Not Applicable
Digital I&C-ISG-02	7	Single Failure - Staff Position (Page 14)	Not Applicable	Digital I&C-ISG-02 is not applicable to the NuScale application. See DSRS 7.1.5 in Table 1.9-3 which provides information on Diversity and Defense-in-Depth review.	Not Applicable
Digital I&C-ISG-03: Risk-Informed Digital Instrumentation and Controls	4	Staff Position	Not Applicable	Digital I&C-ISG-03 is not applicable to the NuScale DCA. See DSRS 7.0 in Table 1.9-3 which provides an overview of the I&C review process.	Not Applicable
Digital I&C-ISG-04: Highly Integrated Control Rooms & Digital Communication Systems	1	Interdivisional Communications - Staff Position (Pages 4 through 8)	Not Applicable	Digital I&C-ISG-04 is not applicable to the NuScale DCA; however, it is addressed as part of topical report NuScale Power, LLC, TR-1015-18653-P-A, "Design of the Highly Integrated Protection System Platform Topical Report."	Not Applicable

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

ISG Section/ Title	AC	AC Title / Description	Conformance Status	Comments	Section
Digital I&C-ISG-04	2	Command Prioritization - Staff Position (Pages 8 through 10)	Not Applicable	Digital I&C-ISG-04 is not applicable to the NuScale DCA; however, it is addressed as part of topical report NuScale Power, LLC, TR-1015-18653-A, "Design of the Highly Integrated Protection System Platform Topical Report."	Not Applicable
Digital I&C-ISG-04	3	Multidivisional Control and Display Stations - Staff Position (Pages 11 through 16)	Not Applicable	Digital I&C-ISG-04 is not applicable to the NuScale DCA; however, it is addressed as part of topical report NuScale Power, LLC, TR-1015-18653-A, "Design of the Highly Integrated Protection System Platform Topical Report."	Not Applicable
Digital I&C-ISG-04	3.1	Independence and Isolation (Pages 11 through 13)	Not Applicable	Digital I&C-ISG-04 is not applicable to the NuScale DCA; however, it is addressed as part of topical report NuScale Power, LLC, TR-1015-18653-A, "Design of the Highly Integrated Protection System Platform Topical Report."	Not Applicable
Digital I&C-ISG-04	3.2	Human Factors Considerations (Pages 13 through 15)	Not Applicable	Digital I&C-ISG-04 is not applicable to the NuScale DCA; however, it is addressed as part of topical report NuScale Power, LLC, TR-1015-18653-A, "Design of the Highly Integrated Protection System Platform Topical Report."	Not Applicable
Digital I&C-ISG-04	3.3	Diversity and Defense-in-Depth (D3) Considerations (Page 15)	Not Applicable	Digital I&C-ISG-04 is not applicable to the NuScale DCA; however, it is addressed as part of topical report NuScale Power, LLC, TR-1015-18653-A, "Design of the Highly Integrated Protection System Platform Topical Report."	Not Applicable
Digital I&C-ISG-05: Highly Integrated Control Rooms - Human Factors	1	Computer-Based Procedures - Staff Position (Pages 3 through 7)	Partially Conforms	This position is applicable except for site-specific operational elements of subtier NUREG-0899 that are the responsibility of the COL applicant.	18.7
Digital I&C-ISG-05	2	Minimum Inventory - Staff Position (Pages 9 through 11)	Partially Conforms	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	This acceptance criterion is superseded by Chapter 18 Appendix 18-A.	Not Applicable
Digital I&C-ISG-05	3.1.B	Phase 1: Analysis - Review Criteria (Pages 15 through 16)	Not Applicable	This acceptance criterion is superseded by Chapter 18 Appendix 18-A Criterion 1.B.	Not Applicable
Digital I&C-ISG-05	3.2.B	Phase 2: Preliminary Validation - Review Criteria (Page 18)	Not Applicable	This acceptance criterion is superseded by Chapter 18 Appendix 18-A Criterion 2.B.	Not Applicable
Digital I&C-ISG-05	3.3.B	Phase 3: Integrated System Validation - Review Criteria (Pages 19 through 20)	Not Applicable	This acceptance criterion is superseded by Chapter 18 Appendix 18-A.Criterion 3.B.	Not Applicable

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

ISG Section/ Title	AC	AC Title / Description	Conformance Status	Comments	Section
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	This acceptance criterion is superseded by Chapter 18 Appendix 18-A.Criterion 4.B.	Not Applicable
Digital I&C-ISG-06: Licensing Process	All	Various	Not Applicable	This guidance is for 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	This guidance is for 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	This guidance governs site-specific programmatic and design aspects of emergency planning that are the responsibility of the COL applicant referencing the NuScale design.	Not Applicable
NSIR/DPR-ISG-02: Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants	All	Various	Not Applicable	Applicable to license holder during decommissioning activities.	Not Applicable
NSIR/DPR-ISG-03: Review of Security Exemptions/License Amendment Requests for Decommissioning Nuclear Power Plants	All	Various	Not 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	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	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

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

ISG Section/ Title	AC	AC Title / Description	Conformance Status	Comments	Section
JLD-ISG-12-04, Draft: Performing a Seismic Margin Assessment in Response to the March 2012 Request for Information Letter	All	Various	Not Applicable	This ISG is for response to the March 2012 50.54(f) request for information letter. DC/COL-ISG-020 remains the NRC's 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	This ISG is for 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	This ISG is for 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	The information in this guidance is site-specific and is the responsibility of the COL applicant.	Not Applicable
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	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	One fuel assembly design is used in the criticality analysis.	9.1.1
DSS-ISG-2010-01	2	Depletion Analysis	Conforms	The analysis does not take credit for burnup.	9.1.1
DSS-ISG-2010-01	3a	Axial Burnup Profile	Conforms	The analysis does not take credit for burnup.	9.1.1
DSS-ISG-2010-01	3b	Rack Model	Conforms	The rack model analysis is appropriate for conditions.	9.1.1
DSS-ISG-2010-01	3c	Interfaces	Conforms	The analysis does not take credit for zoning or a loading pattern.	9.1.1
DSS-ISG-2010-01	3d	Normal Conditions	Conforms	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

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

ISG Section/ Title	AC	AC Title / Description	Conformance Status	Comments	Section
DSS-ISG-2010-01	3e	Accident Conditions	Conforms	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	The analysis considers area of applicability in the code validation.	9.1.1
DSS-ISG-2010-01	4b	Trend Analysis	Conforms	The analysis includes a trend analysis in the code validation.	9.1.1
DSS-ISG-2010-01	4c	Statistical Treatment	Conforms	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	The analysis does not take credit for burnup.	9.1.1
DSS-ISG-2010-01	4e	Code-to-Code Comparisons	Conforms		9.1.1
DSS-ISG-2010-01	5a	Precedents	Conforms	The analysis does not rely upon cited precedents.	9.1.1
DSS-ISG-2010-01	5b	References	Conforms	Cited references are publicly available and are referenced in SFP criticality analyses.	9.1.1
DSS-ISG-2010-01	5c	Assumptions	Conforms	Assumptions used 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	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	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	This rule requires an evaluation of proposed PWR auxiliary feedwater (AFW) systems. The NuScale plant design does have an AFW system like a typical 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	The NuScale reactor design differs from large PWRs because 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	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	This requirement applies only to BWRs.	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	This requirement applies 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
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	This requirement applies only to BWRs.	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	Comments	Section
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	This requirement applies only to BWRs.	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	This requirement applies only to BWRs.	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	This requirement applies only to BWRs.	Not Applicable
50.34(f)(1)(xi)	Provide an evaluation of depressurization methods (II.K.3.45)	Not Applicable	This requirement applies only to BWRs.	Not Applicable
50.34(f)(1)(xii)	Perform an evaluation of alternative hydrogen control systems	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
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	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	The plant procedure improvement program specified by this requirement (and development of plant procedures) is the responsibility of the COL applicant referencing the certified design.	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	None.	18.7
50.34(f)(2)(iv)	Provide a plant safety parameter display console (I.D.2)	Conforms	The NuScale safety display and indication system is integrated into the control room human-system interface design rather than having a separate console.	7.1 7.2.13 18.7.2

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	Comments	Section
50.34(f)(2)(v)	Provide for automatic indication of the bypassed and operable status of safety systems (I.D.3)	Conforms	None.	7.1 7.2.4 7.2.13
50.34(f)(2)(vi)	Provide the capability of high point venting of noncondensible 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	The venting of noncondensible gases is unnecessary to ensure long term core cooling capability.	5.4.4
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	The NuScale design does not contain vital areas, as defined by NUREG-0737, Item II.B.2, other than the areas for initiating combustible gas monitoring, main control room and technical support center. Protection of necessary equipment from radiation is reasonably assured through demonstrating equipment survivability.	12.4 19.2
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)	Departure	The NuScale design does not rely on primary coolant or containment samples to assess the extent of potential core damage. The NuScale design relies upon radiation monitors under the bioshield and core exit temperature indications for this assessment. The NuScale design supports an exemption from 10 CFR 50.34(f)(2)(viii) design criterion for obtaining and analyzing post-accident samples of the reactor coolant system and containment without exceeding prescribed radiation dose limits.	9.3.2 11.5 12.4
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	Pursuant to 10 CFR 52.47(a)(8) and 10 CFR 50.34(f), Paragraph (f)(2)(ix) 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	Comments	Section
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	This requirement is applicable to the DCA except for aspects specifying PORV block valve testing and consideration of ATWS conditions in the testing program. The NuScale design does not use power-operated relief valves. The ATWS provision is not technically relevant to the NuScale design. This aspect of the regulation relates to reactor designs that rely on the relief and safety valves to mitigate the consequences of an ATWS event. The NuScale design supports an exemption from 10 CFR 50.62(c)(1) because the NuScale design relies on protection system diversity to prevent an ATWS, rather than design features to mitigate the condition. As a result, the module response to an ATWS is not analyzed in FSAR Section 15.8, such that the performance of the relief and safety valves is not relied upon to meet the ATWS safety criteria. Therefore, consideration of ATWS conditions in the relief and safety valve test program is not necessary to ensure acceptable performance.	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	None.	5.2 6.3.1 7.1 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	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	The NuScale design does not have an AFW system as would be found at a typical LWR. Also, while the DHRS performs some of the functions of an AFW system at a PWR, the NuScale DHRS 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 DHRS operation involves passive natural circulation flow, with flow characteristics that vary with system conditions, which makes DHRS flow a less useful measurement for the NuScale design. Control room indication for system parameters other than DHRS flow are more appropriate to ensure operators have the information necessary to adequately monitor DHRS operation and reactor core cooling. These parameters include DHRS pressure, valve position indication, and reactor coolant system pressure and temperature. 10 CFR 50.34(f)(2)(xii) is not considered applicable to the NuScale DHRS. Because the language and intent of 10 CFR 50.34(f)(1)(ii) do not apply, the requirement is not applicable to the NuScale design. An exemption would be unnecessary because 10 CFR 50.34(f)(1)(ii) only applies to the technically relevant portions of the TMI requirements.	Not Applicable
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	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.	5.4.5 8.3.1 8.3.2

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	Comments	Section
50.34(f)(2)(xiv) 50.34(f)(2)(xiv)(A) 50.34(f)(2)(xiv)(B) 50.34(f)(2)(xiv)(C) 50.34(f)(2)(xiv)(D) 50.34(f)(2)(xiv)(E)	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	The NuScale design conforms to 50.34(f)(2)(xiv)(A), (B), (C), and (D). NuScale is requesting an exemption to TMI requirement 10 CFR 50.34(f)(2)(xiv)(E). For 50.34(f)(2)(xiv)(D), the high containment pressure analytical limit is above the highest allowable containment pressure for leak detection operability. Therefore, the set point for initiating containment isolation is compatible with normal operation. Additionally, the containment high pressure analytical limit is subatmospheric, therefore, any pressure setpoint up to and including the analytical limit will prevent a release to the environs. For 50.34(f)(2)(xiv)(E), the NuScale design differs from that of a traditional large water reactor design of a TMI-era vintage because reactor core uncovery, and resulting core damage, cannot occur without reaching the low low pressurizer level containment isolation setpoint. The pressurizer is an integral part of the reactor vessel, located well above the reactor core, and not connected to the reactor core by piping. Design basis events meet their thermal and hydraulic acceptance criteria without reliance on isolating the CES on a high radiation signal. No design basis event results in degraded or damaged core conditions. Section 19.2 analyses demonstrate severe accident conditions, with resultant core damage, also result in generation of reliable containment isolation signals, without reliance on isolation on high containment radiation. An in-containment event resulting in core damage or degradation also results in containment isolation on low low pressurizer level and high containment pressure. An event that leads to core damage or degradation also results in containment isolation on low low pressurizer level. These features provide a reliable alternative means to prevent radiological release from the CES to the environs.	5.2.5 6.2.4 7.1.5 7.2.13 9.3.6 19.2

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	Comments	Section
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	The NuScale containment vessel is smaller than a typical containment building, does not contain sub-compartments and does not 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.	Not Applicable
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	This requirement applies 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	None.	6.2.1 7.1.1 7.2.13 9.3.2 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	None.	4.3.2 6.3 7.0.4 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	Comments	Section
50.34(f)(2)(xix)	Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage (II.F.3)	Conforms	None.	7.1.1 7.1.2 7.2.13 19.2
50.34(f)(2)(xx)	Provide power supplies for pressurizer relief valves, block valves, and level indicators (II.G.1)	Departure	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 from the portions of 10 CFR 50.34(f)(2)(xx) related to pressurizer level indicators.	5.4.5 7.2.13 8.1.4 8.3.1 8.3.2
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	This requirement applies only to BWR designs.	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	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	This requirement applies only to B&W plant designs.	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	This requirement applies only to BWR designs.	Not Applicable
50.34(f)(2)(xxv)	Provide an onsite Technical Support Center and onsite Operational Support Center (III.A.1.2)	Partially Conforms	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	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.	5.4 6.3.1 9.3.2 9.3.4
50.34(f)(2)(xxvii)	Provide for monitoring of in-plant radiation and airborne radioactivity (III.D.3.3)	Conforms	None.	11.5 11.6 12.3.4

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	Comments	Section
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	None.	6.4.1 6.4.4 15.0.3
50.34(f)(3)(i)	Provide administrative procedures for evaluating operating, design, and construction experience (I.C.5)	Not Applicable	This requirement is the responsibility of the COL applicant.	Not Applicable
50.34(f)(3)(ii)	Ensure that the QA list required by Criterion II in Appendix B to 10 CFR 50 includes all SSC important to safety (I.F.1)	Conforms	None.	3.2 17.4
50.34(f)(3)(iii)	Establish a QA Program based on the specified considerations (I.F.2)	Partially Conforms	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.	17.5
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)	Departure	This requirement is not technically relevant to the NuScale design. 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 differs from a typical LWR containment structure because of its high-pressure capability. A 3-foot opening relative to the NuScale containment is unnecessary. As discussed in Section 6.2.1.1.1, the calculated peak containment for design basis events remains less than the CNV internal design pressure. As discussed in Section 19.2.3, peak containment pressures do not challenge containment integrity for any analyzed severe accident progression. (Refer to TR-0716-50424, Section 2.8).	6.2 19.2
50.34(f)(3)(v)	Preliminary Design Information - Containment Integrity (II.B.8)	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

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	Comments	Section
50.34(f)(3)(vi)	For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations (II.E.4.1)	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	This requirement is applicable only to applicants and holders of reactor facility licenses.	Not Applicable
Issue 191	Assessment of Debris Accumulation on PWR Sump Performance	Conforms	None.	6.3 18.2.3
Issue 193	Boiling Water Reactor Emergency Cooling Water System (ECCS) Suction Concerns	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	This is applicable to currently-operating plants.	Not Applicable
Issue 204	Flooding of Nuclear Power Plant Sites Following Upstream Dam Failures	Not 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	Comments	Section
Generic Letter 88-14	Instrument Air Supply System Problems Affecting Safety-Related Equipment	Conforms	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	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	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	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	The NuScale Power Plant design 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.	Not Applicable
Generic Letter 2007-01	Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients.	Partially Conforms	As described in Chapter 8, the electrical power systems do not include power cables that provide power to equipment with risk-significant or safety-related functions. The scope of compliance with the issues addressed by GL 2007-01 is limited to power cables within the scope of 10 CFR 50.65. Conformance is achieved for cable monitoring by the COL holder applying the guidance of RG 1.218 as discussed in Chapter 8.	8.1 8.2 8.3
Generic Letter 2008-01	Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems	Partially Conforms	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.	5.4 Ch 6
Bulletin 2007-01	Security Officer Attentiveness	Not 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	Comments	Section
Bulletin 2011-01	Mitigating Strategies	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 compliance with 10 CFR 50.54(hh)(2) is addressed in Section 20.2.	Not Applicable
Bulletin 2012-01	Design Vulnerability in Electric Power System	Partially Conforms	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	Comments	Section
SECY-89-013	Design Requirements Related to the Evolutionary Advanced Light Water Reactors	Conforms	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	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 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	Incorporated into 10 CFR 52 and implementing NRC guidance documents.	-
SECY-90-377	Requirements for Design Certification under 10 CFR Part 52	Conforms	Incorporated into 10 CFR 52 and implementing NRC guidance documents.	-
SECY-91-074	Prototype Decisions for Advanced Reactor Designs	Conforms	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 WaterReactor (LWR) Certification Issues	Not Applicable	SECY-91-078 pertains to evolutionary ALWR designs and is not directly applicable to passive plant designs.	Not Applicable
SECY-91-178	ITAAC for Design Certifications and Combined Licenses	Conforms	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	Incorporated into 10 CFR 52 and implementing NRC guidance documents.	-
SECY-91-229	Severe Accident Mitigation Design Alternatives for Certified Standard Designs	Conforms	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	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	Incorporated into NRC Orders, regulatory guidance, and pending rulemaking.	14.3.6
SECY-92-092	The Containment Performance Goal, External Events Sequences, and the Definition of Containment Failure for Advanced LWRs	Conforms	Incorporated into NRC Orders, regulatory guidance, and pending rulemaking.	-

Table 1.9-7: Conformance with Advanced and Evolutionary Light Water Reactor Design Issues (SECYs and Associated SRMs) (Continued)

Doc ID	Title	Conformance Status	Comments	Section
SECY-93-087	Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs	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	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 8.2 8.3 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	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	Incorporated into 10 CFR 52 and implementing NRC guidance documents.	8.1 8.2 8.3 8.4 19.3
SECY-96-128	Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design	Partially Conforms	Section IV of this SECY applies.	19.3
SECY-14-038	Performance-Based Framework for Nuclear Power Plant Emergency Preparedness Oversight	Not 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	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	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	None.	15.0.3 15.10
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	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	Design does not use external loops and no drain down condition for refueling.	Not Applicable
I.D	Station Blackout (SBO): Position on methods to mitigate the effects of a loss of all AC power.	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.	Not Applicable
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	None.	Appendix 9A
I.F	Intersystem LOCA: Position on acceptable design practices and preventative measures to minimize the probability of an ISLOCA.	Conforms	None.	9.3.4 19.2.2
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		6.2.5
I.H	Core Debris Coolability: Acceptability criteria for cooling area and quenching ability regarding corium interaction with concrete.	Conforms	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	None.	19.2.3
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	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	None.	19.2.4

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	Comments	Section
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	None.	19.2.3
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	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	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	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	The NuScale electrical system design conforms to the passive plant guidance of SECY-94-084, Section G.	Not Applicable
II.C	Seismic Hazard Curves and Design Parameters: Position on use of proposed generic bounding seismic hazard curves and performance of seismic PRA.	Conforms	None.	19.1.5
II.D	Leak-Before-Break: Position on use of leak-before-break concept.	Conforms	LBB is applied to the MS and FW lines inside containment.	3.6.3
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	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.	Partially Conforms	The FSAR uses the maximum tornado wind speed of 230 mph found in RG 1.76 Revision 1 rather than the outdated 300 mph guidance found in SECY-93-087.	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	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	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.	Departure	The NuScale design supports an exemption from 10 CFR 50.34(f)(2)(viii).	9.3.2

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	Comments	Section
II.J	Level of Detail: Position on a design certification submittal with depth of detail similar to that in an FSAR.	Conforms	None.	All FSAR Sections
II.K	Prototyping: No guidance provided; information only	Conforms	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	None.	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	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	None.	19.1
II.O	Severe Accident Mitigation Design Alternatives (SAMDAs): Position on the consideration of SAMDA as part of the final design approval/design certification of an advanced reactor.	Conforms	None.	19.2.6
II.P	Generic Rulemaking Related to Design Certification: No guidance provided; information only.	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	None.	7.1.5
II.R	Multiple SG Tube Failures: Position on requiring that analysis of multiple SG Tube Failures of 2 to 5 SG tubes be included in the application for design certification of passive ALWRs.	Conforms	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	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	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	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	None.	15.0.0
III.C	Thermal-Hydraulic Stability of the SBWR	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	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	The provisions of this SECY are met by using the nonsafety-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 habitably in accordance with regulatory standards.	Conforms	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	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	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	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 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.