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December 31, 2022 Docket No. 52-050

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of the NuScale Standard Design Approval Application Part 2 – Final Safety Analysis Report, Chapter 1, "Introduction and General Description of the Plant," Revision 0

REFERENCES:

1. NuScale letter to NRC, "NuScale Power, LLC Submittal of Planned Standard Design Approval Application Content," dated February 24, 2020 (ML20055E565)
2. NuScale letter to NRC, "NuScale Power, LLC Requests the NRC staff to conduct a pre-application readiness assessment of the draft, 'NuScale Standard Design Approval Application (SDAA),'" dated May 25, 2022 (ML22145A460)
3. NRC letter to NuScale, "Preapplication Readiness Assessment Report of the NuScale Power, LLC Standard Design Approval Draft Application," Office of Nuclear Reactor Regulation dated November 15, 2022 (ML22305A518)
4. NuScale letter to NRC, "NuScale Power, LLC Staged Submittal of Planned Standard Design Approval Application," dated November 21, 2022 (ML22325A349)

NuScale Power, LLC (NuScale) is pleased to submit Chapter 1 of the Standard Design Approval Application, "Introduction and General Description of the Plant," Revision 0. This chapter supports Part 2, "Final Safety Analysis Report," (FSAR) of the NuScale Standard Design Approval Application (SDAA), described in Reference 1. NuScale submits the chapter in accordance with requirements of 10 CFR 52 Subpart E, Standard Design Approvals. As described in Reference 4, the enclosure is part of a staged SDAA submittal. NuScale requests NRC review, approval, and granting of standard design approval for the US460 standard plant design.

From July 25, 2022 to October 26, 2022, the NRC performed a pre-application readiness assessment of available portions of the draft NuScale FSAR to determine the FSAR's readiness for submittal and for subsequent review by NRC staff (References 2 and 3). Chapter 1 was not included in the scope of the readiness assessment.

Enclosure 1 contains SDAA Part 2 Chapter 1, "Introduction and General Description of the Plant," Revision 0, nonpublic version. As this version contains Security Related Information (SRI), NuScale requests that the nonpublic version (Enclosure 1), be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. Enclosure 2 contains the public version.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Mark Shaver at 541-360-0630 or at mshaver@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 31, 2022.

Sincerely,

Carrie Fosaaen
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Enclosure 1: SDAA Part 2 Chapter 1, “Introduction and General Description of the Plant,” Revision 0, (nonpublic)
Enclosure 2: SDAA Part 2 Chapter 1, “Introduction and General Description of the Plant,” Revision 0, (public)

Enclosure 1:

SDAA Part 2 Chapter 1, “Introduction and General Description of the Plant,” Revision 0, (nonpublic)

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Enclosure 2:

SDAA Part 2 Chapter 1, “Introduction and General Description of the Plant,” Revision 0, (public)

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NuScale US460 Plant
Standard Design Approval Application

Chapter One
Introduction and General Description of the Plant

Final Safety Analysis Report

Revision 0

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CHAPTER 1 INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

1.1 Introduction

This document represents the Final Safety Analysis Report (FSAR) required under 10 CFR 52.137(a) to be provided as part of an application for a standard design approval under 10 CFR 52, Subpart E and is referred to as such throughout. It describes the NuScale Power, LLC US460 Standard Plant design, including the design bases and limits on its operation; a safety analysis of the structures, systems, and components and of the facility as a whole; and the information prescribed in 10 CFR 52.137(a) that is relevant to the NuScale US460 Standard Plant design.

A NuScale Power Module (NPM), shown in Figure 1.2-4 and Figure 1.2-5, is a collection of systems, sub-systems, and components that together constitute a modularized, movable, nuclear steam supply system.

The NuScale advanced small modular reactor plant design is scalable, such that from one (1) to six (6) NPMs operate within a single Reactor Building (RXB). The design meets the 10 CFR 52.1 definition of “modular design.” The information provided in this FSAR includes the design of an individual NPM, as well as plant design and interfaces for up to a 6-NPM facility. In general, chapters describe a single module. Multi-module information is noted where warranted (e.g., shared systems or analyses of seismic events, which potentially affect multiple modules).

The design features

- no alternating current (AC) or direct current (DC) power requirements for safe shutdown and cooling.
- compact helical-coil steam generators with reactor pressure on the outside of the tubes.
- a high-strength steel containment partially immersed in a pool of water.
- sub-atmospheric containment pressure during normal operation.
- a small core with a correspondingly small source term.
- comprehensive digital instrumentation and controls.

Important features of a multi-module plant include

- a scalable plant design, which allows incremental plant capacity growth.
- a compact nuclear island.
- the ability to operate in "island mode."
- black start.

1.1.1 Plant Location

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

COL Item 1.1-1: An applicant that references the NuScale Power Plant US460 standard design will identify the site-specific plant location.

1.1.2 Schedule

COL Item 1.1-2: An applicant that references the NuScale Power Plant US460 standard design will provide the schedules for completion of construction and commercial operation of each power module.

1.1.3 Format and Content

1.1.3.1 Standard Review Plan - NuScale Design Specific Review Standard

A NuScale Design-Specific Review Standard (DSRS) was 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, Conformance with Regulatory Criteria.

The format and content of this FSAR generally follow the format and content of NUREG-0800 and the DSRS.

1.1.3.2 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.

1.1.3.3 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.

1.1.3.4 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 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 standard design is provided to the NRC separately in the form of topical or technical reports, or in response to requests for additional information. Topical and technical reports that are incorporated by reference are listed in Table 1.6-1 and Table 1.6-2, respectively.

1.1.3.5 Acronyms and Abbreviations

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

1.1.4 Multi-Module Regulatory Considerations

1.1.4.1 Compliance with GDC 5 of 10 CFR 50 Appendix A

The design complies with GDC 5. Other than the ultimate heat sink (UHS), safety-related systems are functionally independent and are not shared among NPMs. The UHS is designed to perform its required safety-related functions during a design-basis event (DBE) in one NPM and a controlled shutdown of the remaining NPMs.

Portions of the RXB and Control Building (CRB) are Seismic Category I structures that contain and support safety-related systems for multiple NPMs. The RXB and CRB structures are not adversely impacted during DBEs. Operation of the nonsafety-related shared systems, including credible failures of these systems during DBEs, does not adversely affect safety-related NPM functions.

The operating configurations include from 1 to 6 NPMs in the operating modes permitted by the technical specifications.

Operating configurations are considered in the design and no restrictions are required to ensure plant safety. The shared systems have been evaluated for interface requirements and system interactions. The plant design provides protection of safety systems in the event of failures in shared systems and the performance of safety-related functions is ensured during DBEs.

1.1.4.2 Multi-Module Considerations During Phased Construction and Startup

The design relies on passive safety-related systems that are module-specific and a shared safety-related UHS. With the exception of the UHS, the shared systems are nonsafety-related and non-risk-significant, and shared system interactions do not result in a loss of NPM safety-related functions. Construction and phased expansion of NPMs do not result in operating configurations that are materially different than that assumed in the safety analysis and the independence of NPM safety-related systems is maintained. The analysis of shared system interactions described in Section 19.1 applies to the operating NPMs during installation of subsequent NPMs. Consequently, restrictions in operating configurations or interface requirements are not necessary to ensure the safe operation of operating NPMs during installation, testing, or startup of subsequent NPMs.

Table 1.1-1: Acronyms and Abbreviations

Acronym or Abbreviation	Description
ABS	auxiliary boiler system
ABVS	Annex Building HVAC system
ABWR	advanced boiling water reactor
AC	alternating current
ACCS	air cooled condenser system
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
APL	actuation and priority logic
APWR	advanced pressurized water reactor
AQ	augmented quality
ARM	area radiation monitor
ARO	all rods out
ARS	acceleration response spectra
ASAI	application specific action item
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
ATJC	articulating traveling jib crane
ATWS	anticipated transient without scram
AVT	all-volatile treatment
AWH	auxiliary wet hoist
AWS	American Welding Society
AWWA	American Water Works Association
BAS	boron addition system
BAST	boric acid storage tank

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
BDBE	beyond-design-basis event
BDBEE	beyond-design-basis external event
BDG	backup diesel generator
BMS	battery monitor system
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
CBT	calibration and test bus
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
CCT	condensate collection tank
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
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
CITF	containment isolation test fixture
CIV	containment isolation valve

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
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
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
CPS	condensate polisher resin regeneration 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
CSI	containment system isolation
CSS	containment sampling system
CST	condensate storage tank
CTB	calibration and test bus
CTG	combustion turbine generator
CUB	Central Utility Building
CUBV	Central Utility Building ventilation
CVAP	Comprehensive Vibration Assessment Program
CVCS	chemical and volume control system
CVCSI	chemical and volume control system isolation
D3	diversity and defense-in-depth
DAC	design acceptance criteria
DAS	distributed antenna system
DAS	diverse actuation system
DAW	dry active waste
DB	double building
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
DCH	direct containment heating
DCR	demand-to-capacity ratio

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
DCS	distributed control system
DDC	distributed Doppler coefficient
DDG	dry dock gate
DDT	deflagration-to-detonation transition
DE	dose equivalent
DEA	deaerator
DFL	dynamic fluid load
DGB	Diesel Generator Building
DGBVS	Diesel Generator Building HVAC system
DHRSAV	DHRS Actuation Valve
DHRS	decay heat removal system
DIM	display interface module
DMA	dimethylamine
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOE	Department of Energy
DOT	Department of Transportation
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
DWO	Density Wave Oscillation
DWS	demineralized water system
DSWI	demineralized water system isolation
EAB	exclusion area boundary
EAL	Emergency Action Level
ECCS	emergency core cooling system
ECL	effluent concentration limit
EDAS	augmented DC power system
EDAS-C	EDAS-common
EDAS-MS	EDAS-module-specific
EDL	equivalent dead load
EDMG	extensive damage mitigation guidelines
EDNS	normal DC power system
EDV	engineering design verification
EFDS	equipment and floor drainage system
EFPD	effective full-power day
EFPY	effective full-power year
EHVS	high voltage AC electrical distribution 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

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
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
ERS	equipment requirement specification
ESAS	emergency safeguards actuation system
ESB	emergency core cooling system supplemental boron
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	function allocation
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
FFSMA	free field strong motion accelerator
FHA	Fire Hazards Analysis
FHA	fuel handling accident
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
FN	ferrite number

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
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
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
FWRV	feedwater regulating valve
FWS	condensate and feedwater system
FWTS	feedwater treatment system
GAC	granulated activated charcoal
GDC	General Design Criterion
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

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
HP	high pressure
HP	horsepower
HP-FWH	high pressure feedwater heater
HPM	human performance monitoring
HPME	high pressure melt ejection
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 and control air system
IBC	International Building Code
ICI	in-core instrumentation
ICIS	in-core instrumentation system
ICS	integrated control system
ICV	individual cell voltage
ID	inside diameter
IDD	interface design description
IE	infrequent event, initiating event
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
IR	intermediate range
ISA	integrated safety analysis
ISA	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

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
IVR	in-vessel retention
JLD	Japan Lessons-Learned Directorate
LBA	lower block assembly
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
LOCV	loss of condenser vacuum
LODC	loss of direct current
LOEL	loss of external load
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
LRLTT	lower riser lifting and torque tool
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
LSSS	limiting safety system setting
LTC	load manual tap changers
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
MAP	module access platform
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
MHS	module heatup system
MIB	monitoring and indication bus
MIC	microbiologically induced corrosion
MIR	module inspection rack
MIT	Massachusetts Institute of Technology

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
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
MS	main steam
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
NDRC	National Defense Research Council
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	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
NRHX	non-regenerative heat exchanger
NSA	neutron source assembly

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
NSAC	Nuclear Safety Analysis Center
NSAR	nonsafety-related with augmented requirements
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
OM	Operational and Maintenance
OOS	out of service
ORE	occupational radiation exposure
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
PCP	Process Control Program
PCS	plant control system
PCT	peak cladding temperature
PCWS	pool cooling and cleanup system
PDC	power distribution center
PDC	Principal Design Criteria
PDIL	power dependent insertion limit
PDIT	differential pressure indicating transmitter
PFT	process feed tank
PGA	peak ground acceleration
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
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

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
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	personal protective equipment
PPS	plant protection system
PRA	Probabilistic Risk Assessment
PRHA	Pipe Rupture Hazards Analysis
PRV	pressure relief valve
PSCIV	primary system containment isolation valves
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
QHO	quantitative health objective
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

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
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
RL	response level
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
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
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
RXCS	reactor core system
RXF	reactor fuel assembly
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

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
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	steel-plate composite
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
SDA	Standard Design Approval
SDAA	Standard Design Approval Application
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
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
SJAE	steam jet air ejector
SLB	steam line break
SLP	site layout plan
SM	single module
SMA	seismic margin assessment

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
SMACNA	Sheet Metal and Air Conditioning Contractors' National Association
SME	subject matter expert
SMR	small modular reactor
SMS	seismic monitoring system
SMSIBV	secondary main steam isolation bypass valve
SMSIV	secondary main steam isolation valve
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	source range
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
SSD	site-specific design
SSE	safe shutdown earthquake
SSI	secondary system isolation
SSI	soil-structure interaction
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	site water intake/discharge structure
SWV	shear wave velocity
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

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
TDS	total dissolved solids
TED	turbine exhaust duct
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
TJC	traveling jib crane
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
TSS	top support structure
TSTF	Technical Specification Task Force
TT	turbine trip
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
UL	Underwriters Laboratories, Inc.
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
UX	horizontal displacement in the North-South direction
UY	horizontal displacement in the East-West direction
UZ	vertical displacement
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
WAMNS	wide area mass notification system
WDT	watchdog timer

Table 1.1-1: Acronyms and Abbreviations (Continued)

Acronym or Abbreviation	Description
WMCR	waste management control room
WSW	wet solid waste
WTB	Waste Treatment Building
XPC	extended passive cooling
ZOI	Zone of Influence
ZPA	zero period acceleration

1.2 General Plant Description

This section summarizes the US460 standard design and provides a general description of the overall facility. The description includes

- principal design criteria, operating characteristics, and safety considerations.
- engineered safety features and emergency systems.
- instrumentation, controls, and electrical systems.
- a steam and power conversion system.
- fuel handling and storage systems.
- plant cooling water systems.
- radioactive waste management systems.

The term "NuScale Power Plant" refers to the entire facility, which includes up to six NuScale Power Modules (NPMs) and the associated balance of plant support systems and structures. Each NPM can be safely operated independent of other NPMs. The plant includes design features that ensure the independence and protection of safety-related systems during design-basis events (DBEs). Except for the ultimate heat sink (UHS), safety-related systems are module-specific and functionally independent of shared systems.

Some nonsafety-related systems are shared among NPMs. These shared systems are designed for operational reliability and availability to minimize restrictions on NPM operating configurations during normal modes of operation. Shared system failures are considered as part of the safety analysis provided in Chapter 15 and the Probabilistic Risk Assessment provided in Chapter 19.

A future applicant may 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. The scopes of the standard design and site-specific design are shown in Figure 1.2-1. The basic systems associated with power generation are shown in Figure 1.2-2. 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.

An NPM, shown in Figure 1.2-4, is a collection of systems, sub-systems, and components that make up a modularized, movable, nuclear steam supply system (NSSS). Each NPM comprises 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 plant is designed for up to six NPMs with the associated primary and secondary systems and components necessary to produce power and maintain the facility.

1.2.1.1 Facility Description

Process Overview

The reactor core transfers heat into the reactor coolant, which flows upward through the core and lower and upper riser assemblies. The heated coolant exits the upper riser assembly and is redirected downward into the SG region. As the reactor coolant transfers heat to the SGs, it cools and becomes denser, promoting 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 1.2-5).

On the secondary side, preheated feedwater is pumped into the tube side of the SGs where it boils. As 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 a microgrid, utility grid, or service load.

Steam that exits or bypasses the turbine is directed to the air-cooled condenser, which removes heat and condenses the steam. The condensate is pumped through condensate polishing equipment to the inlet of the feedwater pumps. A small amount of steam is extracted from turbine stages to preheat the feedwater and increase plant efficiency.

1.2.1.1.1 Principal Design Criteria

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

- reliable, passive safety systems that do not rely on electrical power to fulfill their safety functions
- the ability to cope with DBEs without the need for operator actions
- safety features that ensure 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 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 a passive heat sink for long-term decay heat removal
- a 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 overall characteristics of the design.

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 Section 4.2 and Section 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 a CNV that is partially immersed in the reactor pool. The reactor core is located below the helical-coil SGs inside the RPV.

Reactor Core

Control rod assemblies are used to control reactivity during normal plant operation, shutdown, and scram events.

The fuel assembly design is similar to existing pressurized water reactor designs. The fuel is uranium dioxide, UO_2.

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 heater bundles installed above the integral steam plenum baffle plate. Pressure in the RCS is reduced using spray provided by the chemical and volume control system (CVCS).

Steam Generator

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

Reactor Pressure Vessel

The RPV is an approximately cylindrical pressure vessel with an inside diameter ranging between 8 and 9 feet. The RPV comprises a lower head and lower shell section, a mating flange, an upper shell section, and an upper head. The lower head is torispherical, while the upper head is a truncated torisphere that includes a flat sub-assembly which supports the control rod drive mechanism support structure.

The RPV upper head includes penetrations for the pressurizer spray lines, the pressurizer high point vent line, reactor vent valves, reactor safety valves, and in-core instrumentation. The RPV upper shell houses the SGs, upper vessel internals, and a portion of the pressurizer region (including the pressurizer heater penetrations, heaters, and baffle plate). The upper shell also includes the feedwater and steam plenums, penetrations for the RCS injection and discharge lines, and penetrations for the reactor recirculation valves.

The RPV lower shell houses the vessel internals core support assembly and the reactor core. The lower shell is bolted to the upper shell by a mating flange, and the mating flange allows removal of the lower shell to support refueling operations. There are no penetrations in the lower section of the RPV.

Containment Vessel

The CNV is a cylindrical, steel pressure vessel housing the RPV, control rod drive mechanisms, and associated NSSS piping and components.

The safety functions of the CNV are to

- provide a barrier to contain mass, energy, and fission product release by closure of the containment isolation valves (CIVs) upon containment isolation signal.
- provide a sealed containment and thermal conduction for the condensation of steam that provides makeup water to the RCS.

- transfer core heat from the reactor coolant in containment to the UHS during ECCS operation.
- provide structural support for safety-related structures, systems, and components.
- provide electrical penetration assemblies for safety-related reactor instrumentation cables through the CNV.
- provide the required pressure boundary for DHRS operation.

Following an actuation of the ECCS, steam is vented from the RPV through the reactor vent valves. Steam in contact with the inside surface of the CNV is passively cooled and condensed. This passive process maintains containment pressure and temperature within design limits for at least 72 hours for DBEs. As described in Section 19.3, nonsafety-related SSC are not relied upon to perform a RTNSS B function for a period after 72 hours up to 7 days following an accident to ensure long-term safety.

1.2.1.1.3 Safety Considerations

The integral design of the NPM eliminates external coolant loop piping, which eliminates large-break loss-of-coolant accident (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. These characteristics result in a design with a core damage frequency that is lower than those of the current light water reactor fleet.

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 Containment System

The containment is described in Section 1.2.1.1.2 and Section 6.2.

1.2.1.2.2 Emergency Core Cooling System

The ECCS provides a passive means of decay heat removal in the event of a LOCA. The ECCS consists of two independent reactor vent valves and two independent reactor recirculation valves (Figure 1.2-7). All four 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 naturally circulated through the core. When reactor coolant temperature is reduced to below the boiling point, core cooling continues via conduction through the CNV to the reactor pool. The cooling function of the

ECCS is entirely passive. Section 6.3 provides design and operational information for the ECCS.

1.2.1.2.3 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 fission products in the CNV atmosphere are depleted through the passive process of aerosol deposition. Section 6.5 provides information for this engineered safety feature.

1.2.1.3 Instrumentation, Controls, and Electrical Systems

The instrumentation and controls architectural design philosophy incorporates clear interconnection interfaces, separation between safety and nonsafety systems, and simplification of system functions. The instrumentation and controls 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 - 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, 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 NPMs and are not specific to an individual NPM
- in-core instrumentation system - monitors various parameters within the reactor core and RCS and sends the parameter values to the module control system for display and evaluation

Under normal operating conditions the AC electrical power distribution system supplies power to equipment required for startup, normal operation, and shutdown of the plant. As described in Section 8.3, onsite or offsite electrical power systems are not required to actuate safety-related systems or to cope with DBEs.

The electrical power systems are described below:

- The high voltage AC electrical distribution system provides power from the turbine generators and the backup power supply system to high voltage AC buses and contains the switchyard.
- The medium voltage AC electrical distribution system provides power to buses servicing medium voltage loads.
- The low voltage AC electrical distribution system provides power to buses servicing low voltage loads.
- The augmented DC power system provides DC power to select plant loads.
- The normal DC Power System provides power to nonsafety-related loads.
- The backup power supply system provides backup power for plant loads.

1.2.1.4 Steam and Power Conversion System

The steam and power conversion system associated with an NPM primarily consists of the main steam system, the turbine generator system, the air cooled condenser system, and the condensate and feedwater system as shown in Figure 1.2-2. Chapter 10 provides design and operational information for the steam and power conversion system.

1.2.1.5 Fuel Handling and Storage Systems

The fuel handling and reactor maintenance areas are located in the Reactor Building (RXB) and include space for the spent fuel pool (SFP), refueling pool, and dry dock. The pools are shown in Figure 1.2-8.

1.2.1.6 Plant Cooling Water Systems

The plant cooling water systems include several nonsafety-related systems that support plant operation. These systems include the following:

- The reactor component cooling water system is a closed-loop cooling system that transfers heat from various plant components to the site cooling water system (Section 9.2.2).
- The pool cooling and cleanup system contains subsystems that provide water level control and temperature control for the interconnected reactor pool, refueling pool, and SFP (Section 9.1.3).
- The chilled water system provides cooling for heating ventilation and air conditioning and radioactive waste equipment (Section 9.2.8).
- The site cooling water system is a two-loop system comprising a closed-loop subsystem that interfaces with plant loads, and an open-loop system that rejects heat to the environment via cooling towers (Section 9.2.7).

1.2.1.7 Radioactive Waste Management Systems

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

1.2.1.8 Control Room Habitability System

The control room habitability system (CRHS) ensures that plant operators are adequately protected against the effects of accidental releases of radioactive gases. The CRHS is a passive system that provides clean, compressed, breathable air to the main control room (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.

1.2.2 General Arrangement of Major Structures and Equipment

Figure 1.2-1 presents the layout of a NuScale US460 Power Plant.

1.2.2.1 Reactor Building

As shown in Figure 1.2-1, the RXB is approximately central to the site. Figure 1.2-3 and Figure 1.2-8 through Figure 1.2-17 are RXB drawings. The RXB houses the NPMs and systems and components required for plant operation and shutdown. It is designed with considerations for the effects of aircraft impact, environmental conditions, postulated DBEs (internal and external), and design-basis threats. The RXB also provides radiation protection to plant operations and maintenance personnel.

1.2.2.1.1 Fuel Handling and Reactor Maintenance Areas

The fuel handling and reactor maintenance areas are located in the RXB and include space for the SFP, refueling pool, and dry dock as shown in Figure 1.2-8.

Space is provided 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. 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 personnel in those areas.

A fuel receiving area and a jib crane for loading new fuel assemblies into the new fuel elevator are located near the SFP. The area provides equipment 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 before removal for dry storage and for temporary short-term storage for new fuel assemblies.

1.2.2.2 Control Building

The CRB contains the MCR and the technical support center (TSC). Figure 1.2-18 through Figure 1.2-21 are CRB drawings.

1.2.2.2.1 Main Control Room

The MCR contains control panels for all installed NPMs. Monitoring and control of multiple NPMs can be performed from a control room panel.

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 MCR contains alarms, displays, and controls for monitoring and control by the operators. The control room supervisor station provides an overview of all NPMs using multiple monitors. Monitor displays are designed using human factors analysis to enhance simplicity. The display layout and design provides 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 setpoints that control the NPM or plant functions
- take corrective actions if an NPM or plant system does not operate as intended

1.2.2.2.2 Technical Support Center

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

1.2.2.3 Radioactive Waste Building

The Radioactive Waste Building (RWB) houses equipment and systems for processing radioactive gaseous, liquid, and solid waste and for preparing waste for off-site shipment. Figure 1.2-22 through Figure 1.2-28 are RWB drawings. The building houses equipment to prepare low-level radioactive waste for compaction to reduce volume and provides temporary storage for radioactive waste. Heating, ventilation, and air conditioning equipment for high-efficiency particulate air filtration is located in the RWB. The building is designed to maintain radiation exposures to 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, and for LOCA events prior to ECCS actuation. The system, as shown in Figure 1.2-6, 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 submerged in the reactor pool. In the event of a steam generator tube failure, the affected SG is isolated and the DHRS provides cooling through the intact SG.

Upon receipt of an actuation signal, feedwater and main steam isolation valves, feedwater regulating valves, and secondary main steam isolation valves are closed and the DHRS actuation valves open. Reactor coolant continues to circulate through the RPV, removing decay heat from the core. As water from the DHRS condenser travels through the SG tubes, it absorbs decay heat from the reactor coolant and is converted to steam. The steam then flows back to the DHRS condenser where it transfers 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 5.4.3 provides design and operational information for the DHRS.

1.2.2.4.2 Ultimate Heat Sink

The UHS is a pool of water located in the RXB below plant grade level. The UHS consists of the reactor pool, the refueling pool, and the SFP. The pools are shown in Figure 1.2-8. During normal plant operations, heat is removed from the pool through the pool cooling and cleanup system.

In a DBA involving a sustained loss of 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 UHS.

1.2.2.4.3 Chemical and Volume Control System

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 the CVCS. Injection pumps provide borated water or demineralized water that is delivered into the RCS, and excess reactor coolant is letdown to the liquid radioactive waste system.

Section 9.3.4 provides design and operational information for the CVCS.

1.2.2.5 Other Site Structures

Other site structures include the Turbine Generator Building, Annex Building, Security Buildings, Central Utility Building, and Diesel Generator Buildings.

1.2.3 Plant Features of Special Interest

Human Factors Considerations

The plant design minimizes human error through fail-safe design functionality, allows multi-modular control capability from a single control room with automation design, employs digital display design and soft control technology to enhance usability, and considers minimization of operator workload.

The Human Factors Engineering Program satisfies specific regulatory requirements and guidance, and leverages human performance and operating experience from nuclear and non-nuclear industries.

Chapter 18 describes the Human Factors Engineering Program.

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

Overall Plant	
Number of power modules	up to six
Power Module	
Number of reactors	One
Thermal power rating	250 MWth
Nominal gross electrical output	77 MWe
RCS normal operating pressure	2000 psia
Steam generator number	Two
Steam generator type	Vertical helical tube
Steam cycle	Rankine-subcritical regenerative with superheat
Turbine type	3600 rpm, condensing, with extraction
Reactor Core	
Fuel	UO_2
Refueling interval	18 months

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 DBA remains below containment design pressure	Containment integrity ensured, minimizing the potential for radioactive releases during postulated accidents
RPV inside the CNV	During an accident, water lost from the 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 CNV	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 (250 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 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

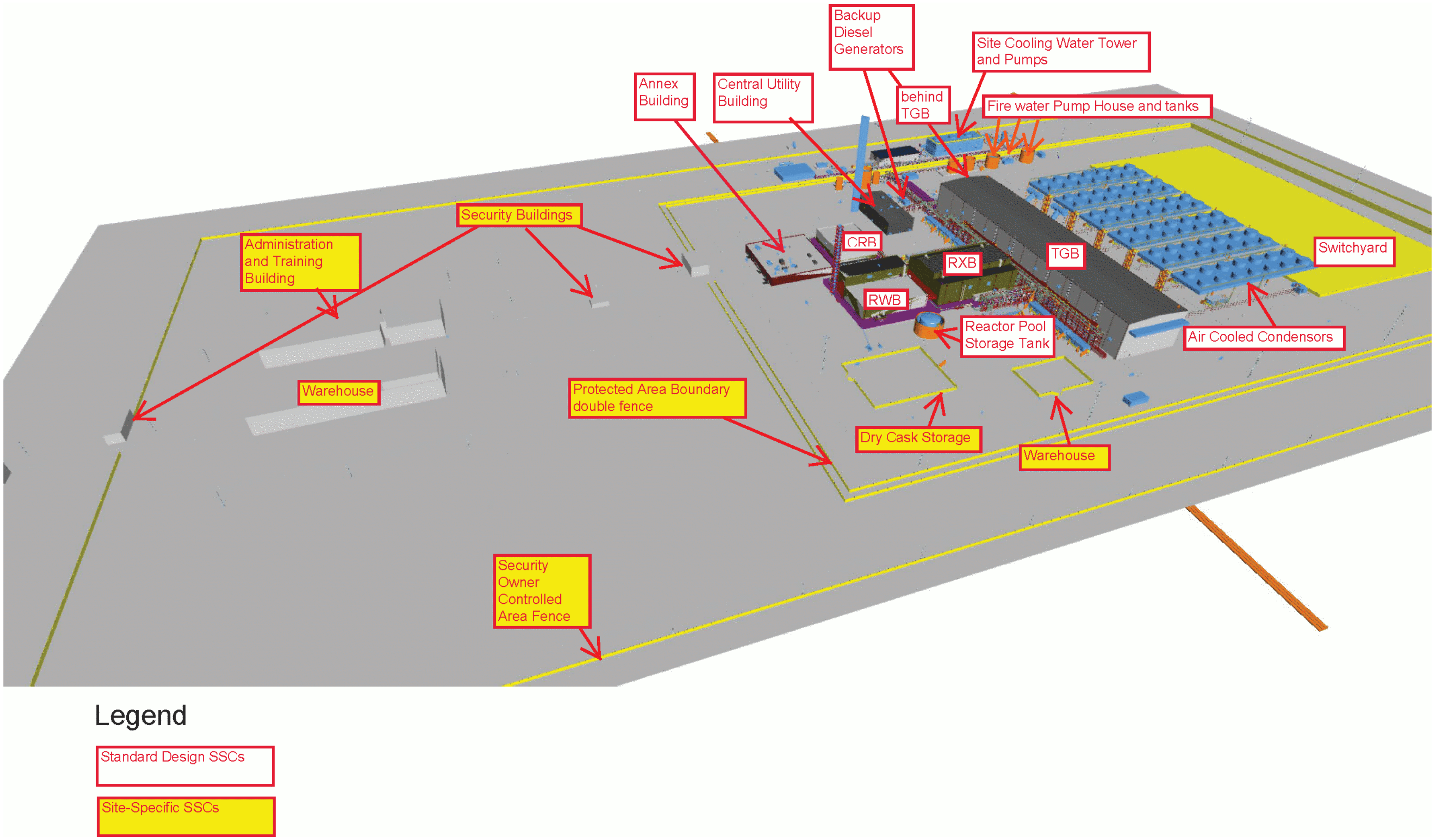
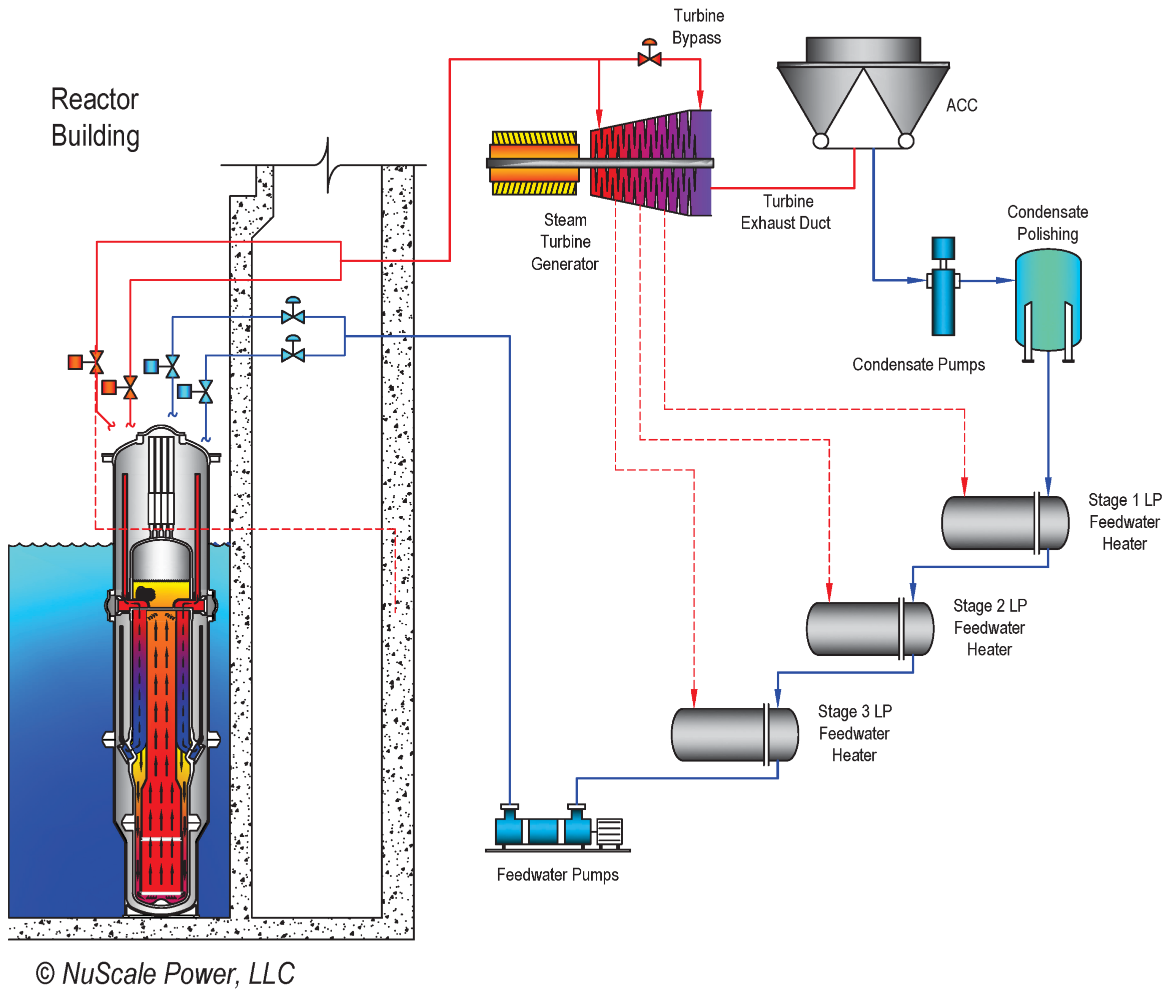
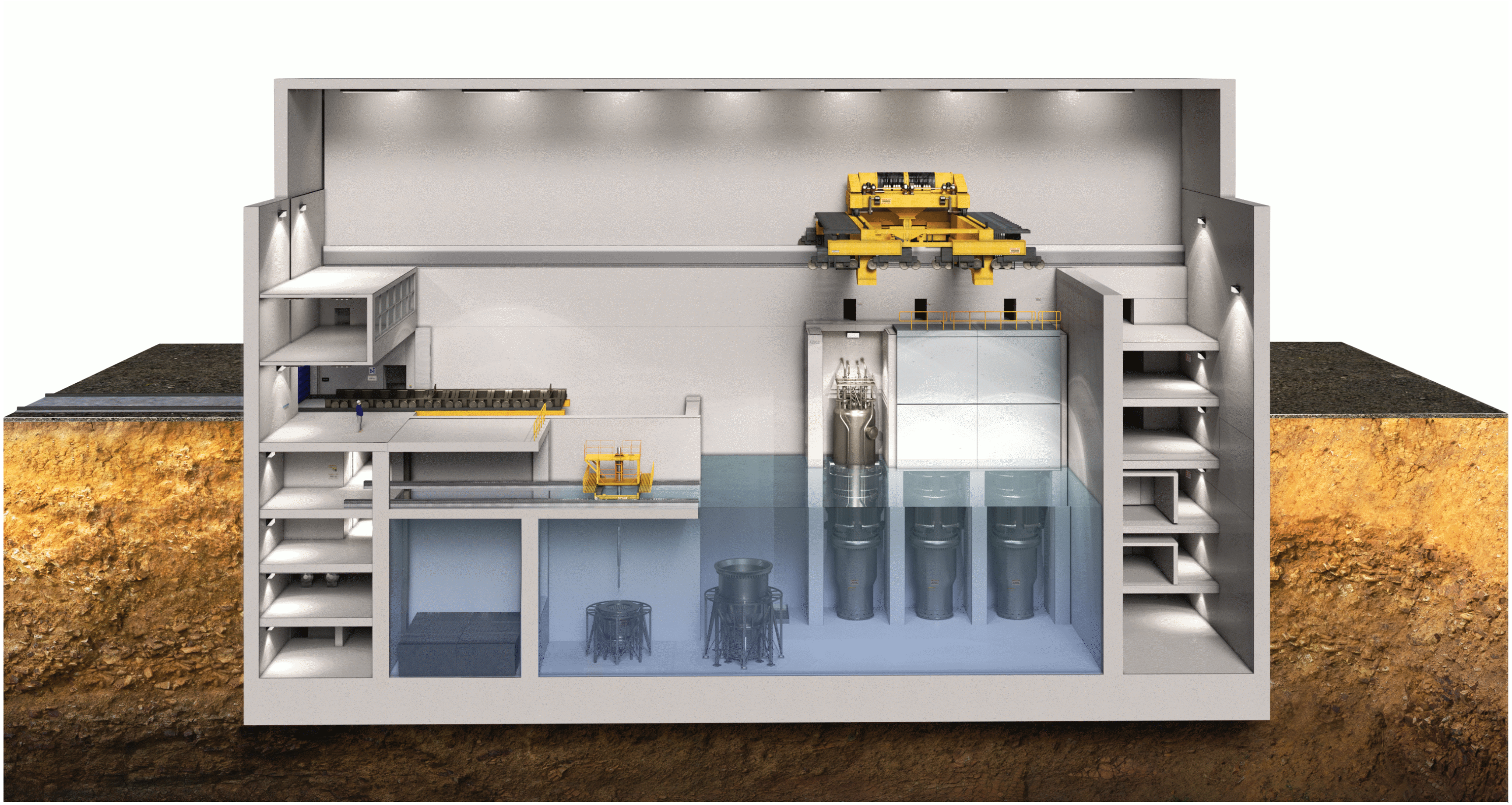


Figure 1.2-2: Schematic of a Single NuScale Power Module and Associated Secondary Equipment



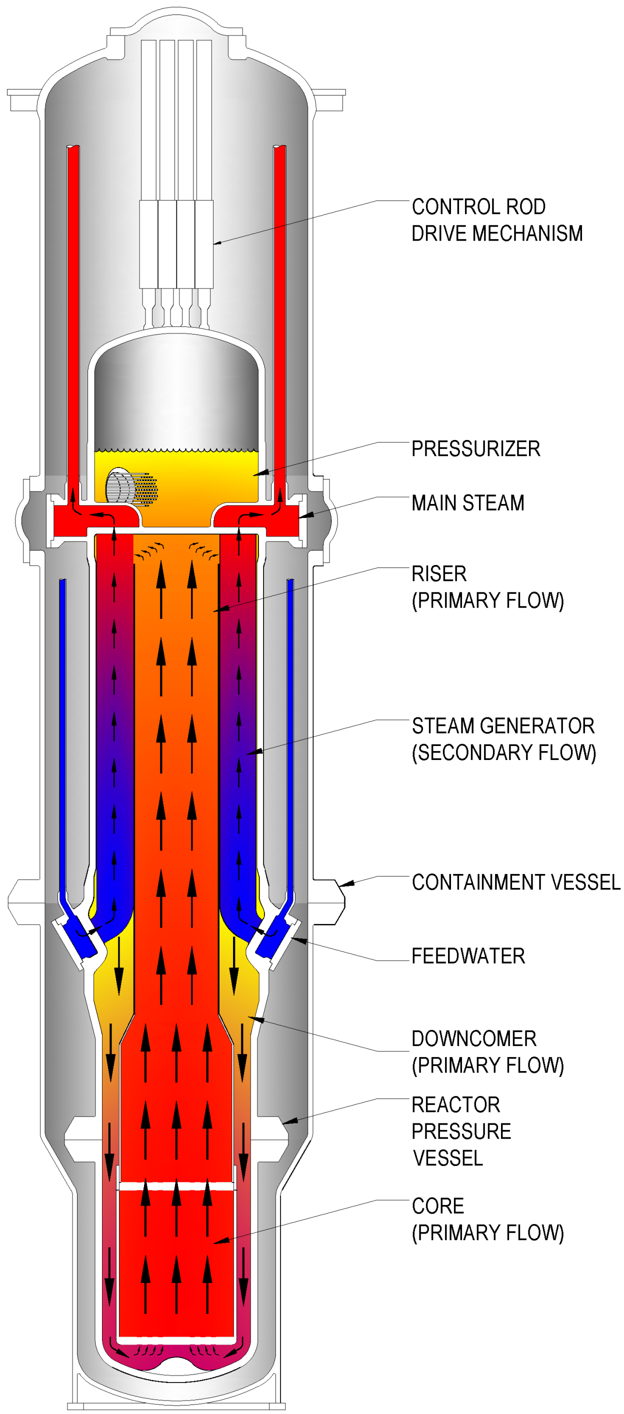
This is a representative figure and does not portray actual proportions, layout, or geometries.

Figure 1.2-3: Cutaway Illustration of 6 Module Configuration



Not to Scale

Figure 1.2-4: Cutaway View of NuScale Power Module



Not to Scale

Figure 1.2-5: Steam Generator and Reactor Flow

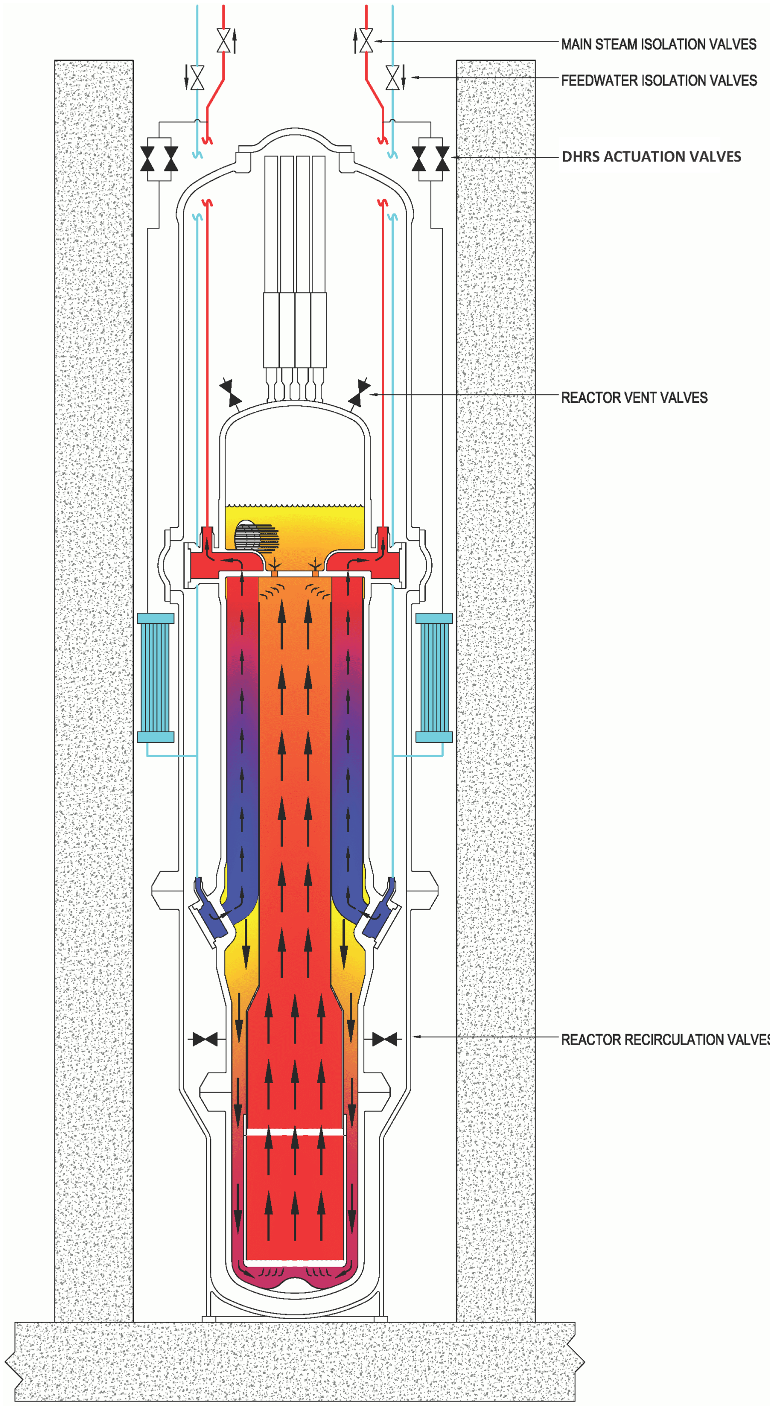
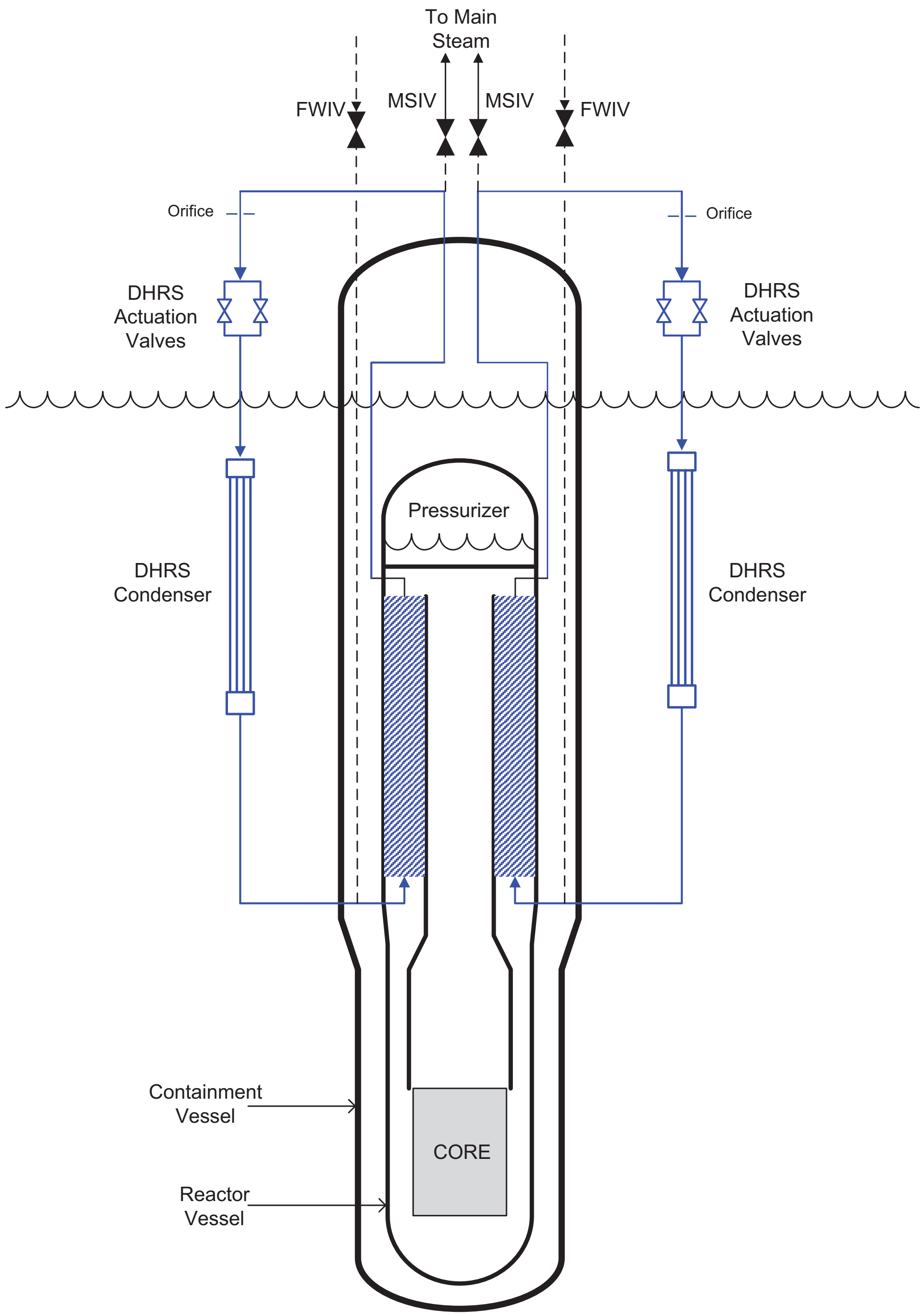
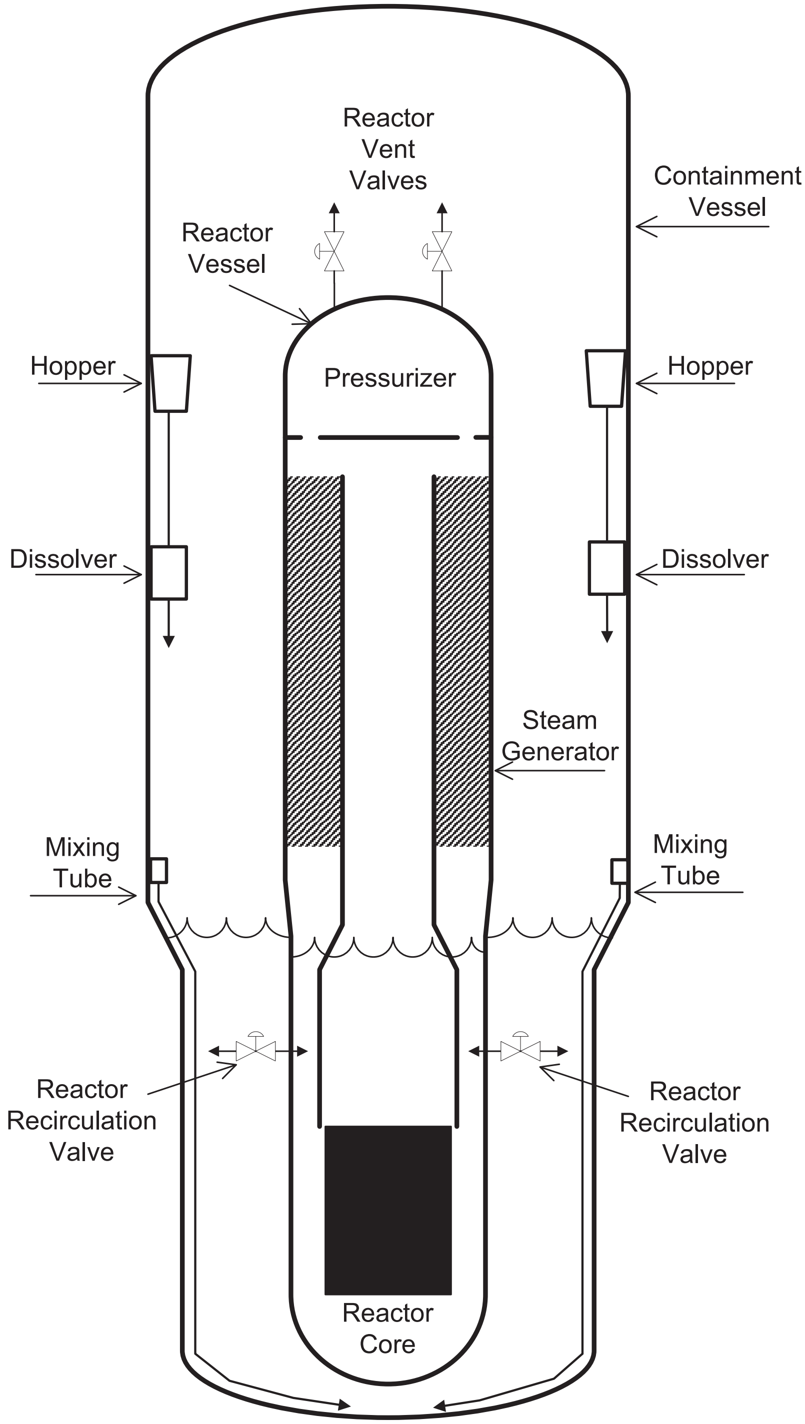


Figure 1.2-6: Decay Heat Removal System



Not to Scale

Figure 1.2-7: Emergency Core Cooling System



Not to Scale

Figure 1.2-8: Reactor Building 25’-0” Elevation

{{ Withheld - See Part 9 }}

Figure 1.2-9: Reactor Building 40'-0" Elevation

{{ Withheld - See Part 9 }}

Figure 1.2-10: Reactor Building 55’-0" Elevation

{{ Withheld - See Part 9 }}

Figure 1.2-11: Reactor Building 70’-0" Elevation

{{ Withheld - See Part 9 }}

Figure 1.2-12: Reactor Building 85’-0" Elevation

{{ Withheld - See Part 9 }}

Figure 1.2-13: Reactor Building 100'-0" Elevation

{{ Withheld - See Part 9 }}

Figure 1.2-14: Reactor Building 126'-0" Elevation

{{ Withheld - See Part 9 }}

Figure 1.2-15: Reactor Building 146’-6” Elevation

{{ Withheld - See Part 9 }}

Figure 1.2-16: Reactor Building East-West Section View

{{ Withheld - See Part 9 }}

Figure 1.2-17: Reactor Building North-South Section View

{{ Withheld - See Part 9 }}

Figure 1.2-18: Control Building 100'-0" Elevation

{{ Withheld - See Part 9 }}

Figure 1.2-19: Control Building 125’-0” Elevation

{{ Withheld - See Part 9 }}

Figure 1.2-20: Control Building North-South Section View

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Figure 1.2-21: Control Building East-West Section View

{{ Withheld - See Part 9 }}

Figure 1.2-22: Radioactive Waste Building 70’-0" Elevation

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Figure 1.2-23: Radioactive Waste Building 82'-0" Elevation

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Figure 1.2-24: Radioactive Waste Building 100'-0" Elevation

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Figure 1.2-25: Radioactive Waste Building 119’-0” Elevation

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Figure 1.2-26: Radioactive Waste Building 145'-0" Elevation

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Figure 1.2-27: Radioactive Waste Building North-South Section View

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Figure 1.2-28: Radioactive Waste Building East-West Section View

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1.3 Comparison with Other Facilities

Table 1.3-1 provides a comparison of a NuScale Power Module, from the NuScale Power Plant US460 standard design, to a typical pressurized water reactor. Values provided are nominal and for comparison only.

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 NPM
Design life (years)	40	60
Nominal gross electrical output (MWe)	1,186	77
Core thermal output (MWt)	3,411	250
Reactor Core		
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	3.9
Number of control rod assemblies	53	16
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	2,000
Hot leg temperature (°F)	618	598
Reactor coolant pumps	4	0
Reactor Vessel		
Vessel inner diameter (in.)	173	104
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 (sq. ft)	55,000	Approximately 18,000
Number of tubes	5,626	1,380
Pressurizer		
Internal volume (cu. ft)	1,800	Approximately 580
Surge nozzle nominal diameter (in.)	14	None
Containment		
Type	Prestressed Concrete Containment Vessel	Steel Pressure Vessel
Inner diameter (ft)	140	Approximately 14
Height (ft)	205 (inner)	Approximately 76 (outer)
Other Parameters and Features		
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
Residual heat removal pumps	2	None
Accumulators	4	None
I&C system type	Analog	Digital
Emergency diesel generators	2	None

Table 1.3-1: NuScale Plant Comparison with Other Facilities (Continued)

NuScale Plant Parameter or Feature (per NPM)	Typical PWR	NuScale NPM
Turbine type	1800 rpm, Tandem Compound Six Flow	3,600 rpm, 16 stage, Single Flow Condensing
Emergency feedwater pumps	3	None
Charging pumps (chemical and volume control system pumps)	2	2
Volume control tank	1	0
Reactor component cooling water pumps	4	3 total for 6 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	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	
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

NuScale Power, LLC (NuScale), has the overall design responsibility for the NuScale Power Plant US460 standard design.

1.4.1 Principal Consultants

Fluor Corporation (Fluor) provided the balance-of-plant design described in the Standard Design Approval Application.

COL Item 1.4-1: An applicant that references the NuScale Power Plant US460 standard design 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 US460 standard design. The testing program described in this section provides data to support the final safety analyses.

1.5.1 NuScale Testing Programs

1.5.1.1 Critical Heat Flux Testing - Preliminary Fuel Design

The NuScale Power Module (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 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 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 7 feet (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, 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. 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.

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 fuel design, which employs Framatome 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 were used to develop NuScale NSP4 CHF correlation. 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 Framatome 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 Framatome Karlstein Thermal Hydraulics (KATHY) facility located in Karlstein, Germany. The test data were used to develop the NSP4 correlation for the NuFuel HTP2™ fuel design (Reference 1.5-1). Testing described in Section 1.5.1.1 was used to support the supplement to the CHF topical report, which expanded the range of applicability of the NSP4 CHF correlation (Reference 1.5-2).

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 steam generator (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 that supported development of power spectral density spectra that are 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.

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 and third set of SG tests were 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 the Societa Informazioni Esperienze Termoidrauliche (SIET) test facility 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. Specifically for SG density wave oscillation, tests were performed and resulting data have been used to validate NRELAP5 to model phenomena and characteristics that are important to density wave oscillation.

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) 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. 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 NIST is a scaled representation of the NPM reactor, containment, and reactor pool systems. The NIST 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 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, located on the Oregon State University campus in Corvallis, OR, in support of NuScale's US460 standard design. These tests include

- facility characterization tests used to develop the NRELAP5 model of the NIST.
- loss-of-coolant accident (LOCA) tests used to validate NRELAP5 for LOCA and containment analyses.
- non-LOCA (anticipated operational occurrence) tests used to validate NRELAP5 for non-LOCA analyses.
- long-term cooling tests used to validate NRELAP5 for extended passive cooling analyses.

Data obtained from the NIST tests identified above have been used to successfully validate the NRELAP5 code for LOCA and containment peak pressure, non-LOCA, and extended passive cooling applications.

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

- control rod drive mechanism penetrations in the reactor vessel head
- integrated steam plenum

- upper control rod drive shaft supports in the upper riser section
- a control rod drive shaft alignment cone located at the top of the control rod assembly (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 Framatome 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 an 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.

1.5.1.7 Emergency Core Cooling System Valve Demonstration Testing

The NPM design utilizes 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 is 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 RRV main valve.

Test programs were developed to demonstrate and evaluate 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 the test programs was to

- demonstrate the ECCS valves function at reactor operating pressures and temperatures.
- demonstrate the ECCS valves function in borated reactor coolant.
- evaluate functional performance and design characteristics of the ECCS valves.

Initial 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 before 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.

Subsequent testing was performed at the National Technical Services facility in Huntsville, Alabama using full scale prototypic ECCS valves. The types of tests performed with respect to the safety related functions of the ECCS valves included main valve actuation, IAB functionality, and valve flow capacity. Tests were performed through the range of reactor operating pressures and temperatures. These tests demonstrated the ECCS valves are capable of proper functionality as required for the NPM.

1.5.1.8 Emergency Core Cooling System Supplemental Boron Dissolution Testing

The ECCS supplemental boron (ESB) system provides boron to the recirculating coolant in containment during ECCS actuation. The boron addition is provided to add negative reactivity to keep the reactor subcritical during some design-basis events. This boron addition is achieved passively by dissolving solid boron oxide staged in two supplemental boron dissolvers located on the inside of the containment vessel wall. During ECCS actuation a portion of condensate generated on the containment vessel inner wall is captured by condensate channels and redirected through the dissolver. The dissolution of the boron oxide creates a boric acid and water solution that is added to the recirculating coolant.

A test program was developed to measure fundamental characteristics of boron oxide dissolution. This information was used to determine conservative assumptions for the ESB boron oxide dissolution rate which is used as part of the boron transport methodology. The boron transport methodology demonstrates that sufficient boron is transported back to the RPV to meet subcriticality requirements.

Testing was performed in the NIST facility at Oregon State University in Corvallis, OR. The test program was designed to measure dissolution of the boron oxide due to water flow through a scaled test section representing the dissolver basket.

The tests measured the average dissolution rate based on the total mass of boron oxide dissolved and the time required to dissolve it. The boron dissolution rates, measured over varying conditions, are used to support boron transport analyses.

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.

1.5-2 NuScale Power, LLC, "Applicability Range Extension of NSP4 CHF Correlation," TR-107522-P, Revision 1.

1.6 Material Referenced

Topical reports and technical reports that are incorporated by reference as part of the NuScale Power Plant US460 standard design approval 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
MN-122626, Revision 0	NuScale Power, LLC Quality Assurance Program Description	17.5
TR-1015-18653-P-A, Revision 2	Design of the Highly Integrated Protection System Platform	7, 15
TR-131981-P, Revision 0	Methodology for the Determination of the Onset of Density Wave Oscillations	3.9, 5.4

Table 1.6-2: NuScale Referenced Technical Reports

Report Number	Title	FSAR Section
TR-118318, Revision 0	NuScale Design of Physical Security Systems	9.5, 13.6, 14.2
TR-122844-P, Revision 0	NuScale Instrument Setpoint Methodology Technical Report	7.0, 7.2
TR-121353-P, Revision 0	NuScale Comprehensive Vibration Assessment Program Analysis Technical Report	3.9, 14.2
TR-121515-P, Revision 0	NuScale Power Module Seismic Analysis	3.7, 3.12
TR-130877-P, Revision 0	Pressure and Temperature Limits Methodology	5.3
TR-121517-P, Revision 0	NuScale Power Module Short-Term Transient Analysis	3.9
TR-130721-P, Revision 0	Use of Austenitic Stainless Steel for US460 Standard Design Reactor Pressure Vessel	5.2, 5.3
TR-121354-P, Revision 0	NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report	3.9, 14.2
TR-121507-P, Revision 0	Pipe Rupture Hazards Analysis	3.6

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.

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

COL Item 1.7-1: An applicant that references the NuScale Power Plant US460 standard design will list site-specific electrical and instrumentation and control drawings, 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.

Figure 1.7-3a through Figure 1.7-3d show a legend of the symbols and characters used in piping and instrumentation diagrams.

COL Item 1.7-2: An applicant that references the NuScale Power Plant US460 standard design 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	Pressurizer Trip Breaker Arrangement
Figure 7.0-8	Module Protection System Gateway Diagram
Figure 7.0-9	Module Protection System Power Distribution
Figure 7.0-10	Neutron Monitoring System Ex-Core Block Diagram
Figure 7.0-11	Plant Protection System Block Diagram
Figure 7.0-12	Safety Display and Indication System Boundary
Figure 7.0-13	Safety Display and Indication Hub
Figure 7.0-14	Module Control System Internal Functions and External Interfaces
Figure 7.0-1	Plant Control System Internal Functions and External Interfaces
Figure 8.3-1	High Voltage AC Electrical Distribution System
Figure 8.3-2a	Medium Voltage Alternating Current Electrical Distribution System (Common Portion)
Figure 8.3-2b	Medium Voltage Alternating Current Electrical Distribution System (Module-Specific Portion)
Figure 8.3-3	Augmented Direct Current Power System (Common)
Figure 8.3-4a and Figure 8.3-4b	Augmented Direct Current Power System (Module Specific)
Figure 11.5-2	Process and Effluent Radiation Monitoring System Instrumentation and Control 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 6.2-3	Containment System Piping and Instrumentation Diagram
Figure 6.2-6	Containment Isolation Valve Actuator Hydraulic Schematic
Figure 6.3-1	Emergency Core Cooling System
Figure 6.3-3	Simplified Reactor Vent Valve Diagram
Figure 6.3-4	Simplified Reactor Recirculation Valve Diagram
Figure 9.1.3-1	Pool Cooling and Cleanup System Diagram
Figure 9.2.5-1	Ultimate Heat Sink Configuration
Figure 9.2.6-1	Condensate Storage Facility
Figure 9.2.8-1	Chilled Water 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 9.3.7-1	Containment Flooding and Drain 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 10.2-1	Turbine Generator System Schematic
Figure 10.3-1	Main Steam System Piping and Instrumentation Diagram
Figure 11.3-1	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
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 1.7-1a: Electrical Symbols

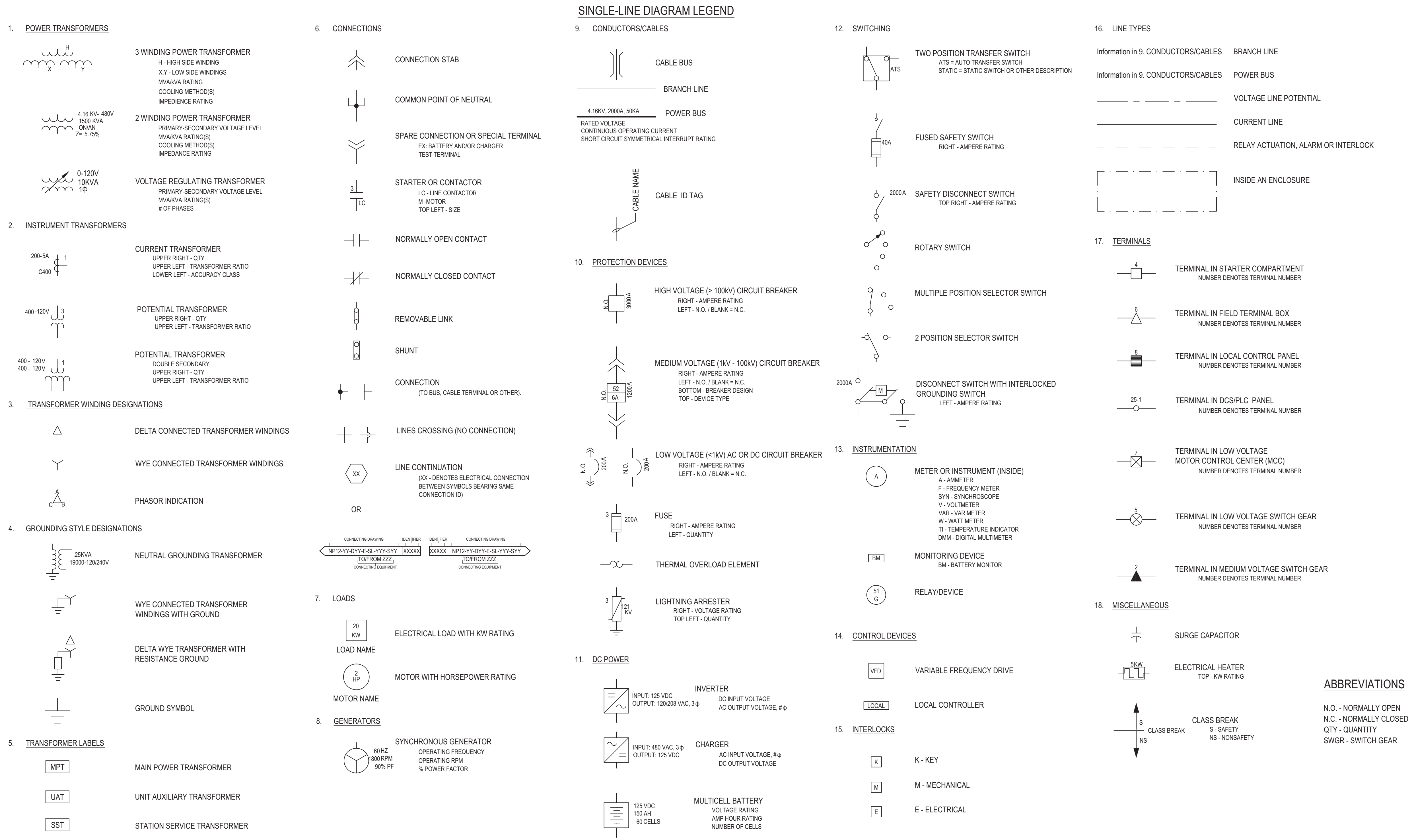


Figure 1.7-1b: Electrical Symbols

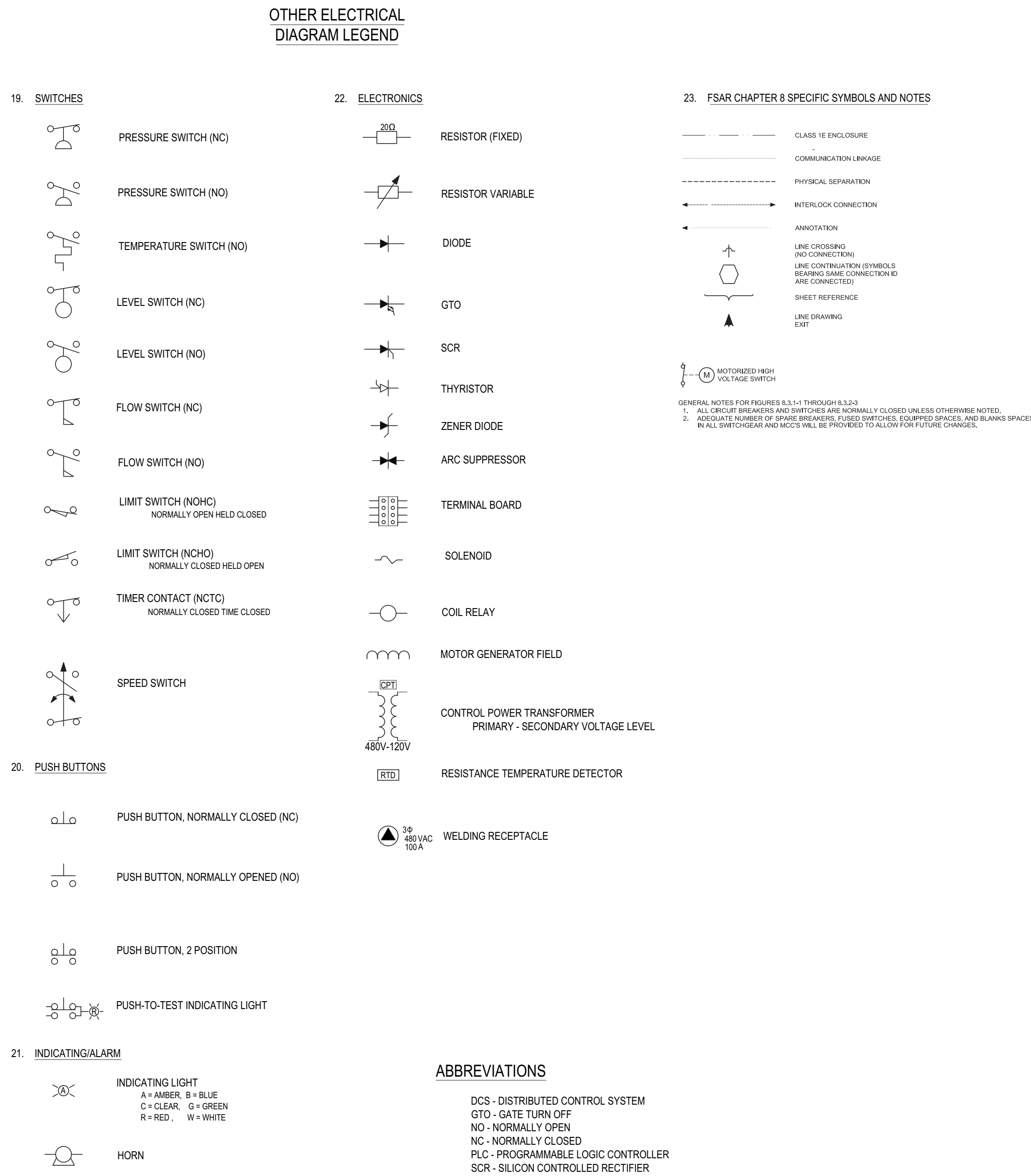


Figure 1.7-2: Instrumentation and Controls Symbol Legend

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

SYMBOL	LOGIC FUNCTION	DESCRIPTION
f(x)	UNSPECIFIED FUNCTION	THE OUTPUT VALUE IS A NONLINEAR OR UNSPECIFIED FUNCTION OF THE INPUT. THE FUNCTION IS DEFINED IN A NOTE OR OTHER TEXT.
(+) Σ (−)	SUMMATION	THE OUTPUT VALUE IS THE ALGEBRAIC SUM OF THE INPUTS.

Figure 1.7-3a: Piping and Instrumentation Diagram Legends

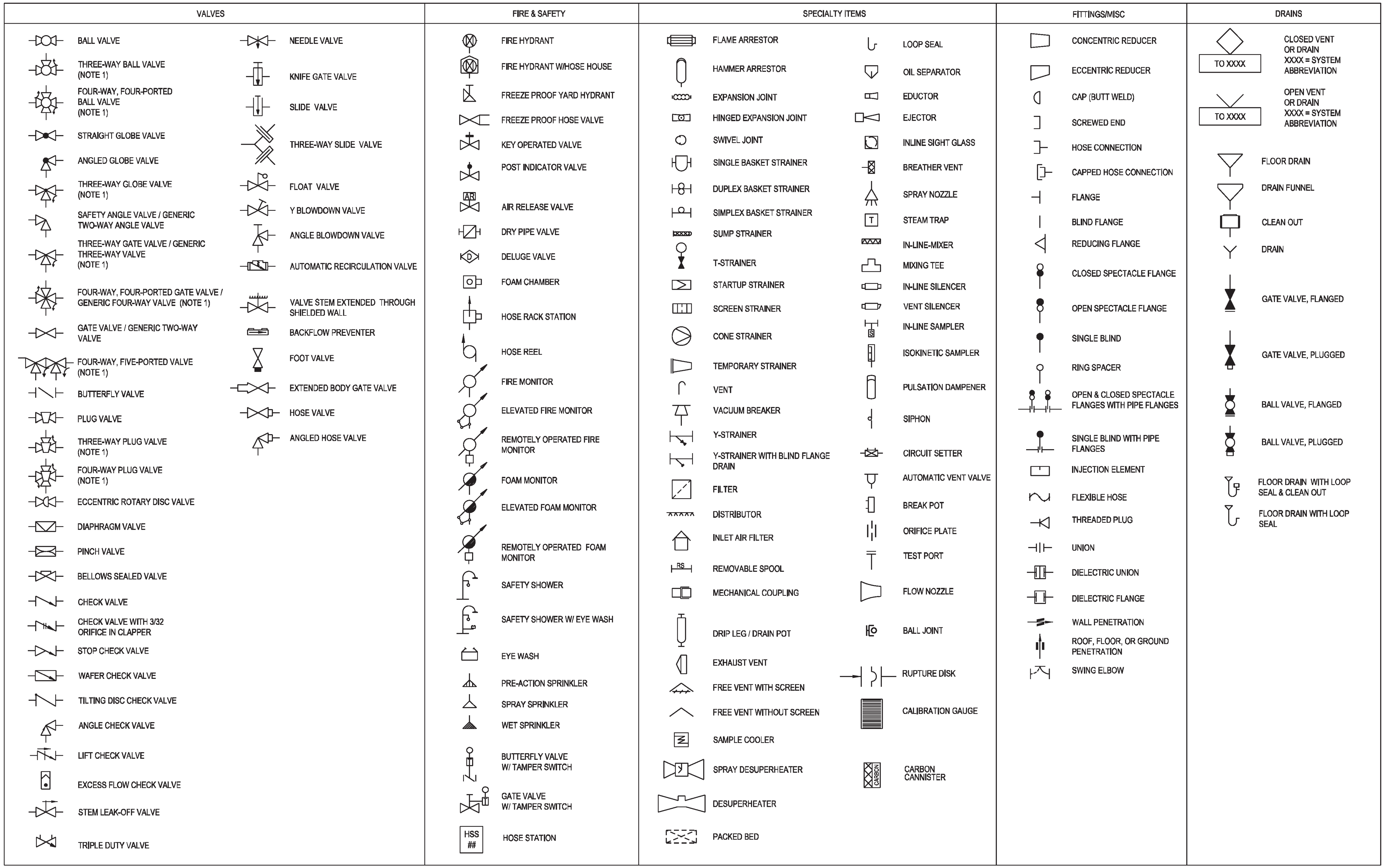


Figure 1.7-3b: Piping and Instrumentation Diagram Legends

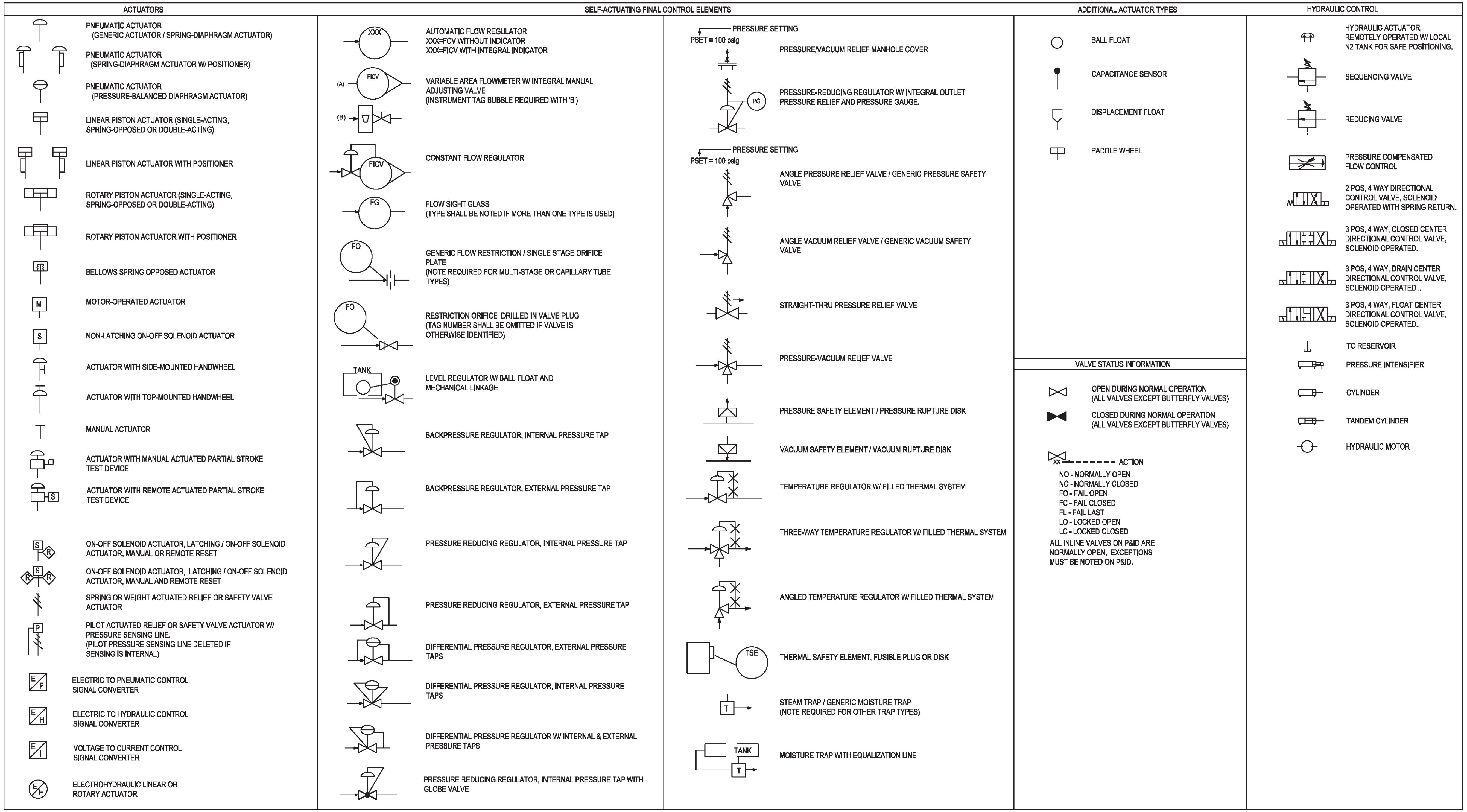


Figure 1.7-3c: Piping and Instrumentation Diagram Legends

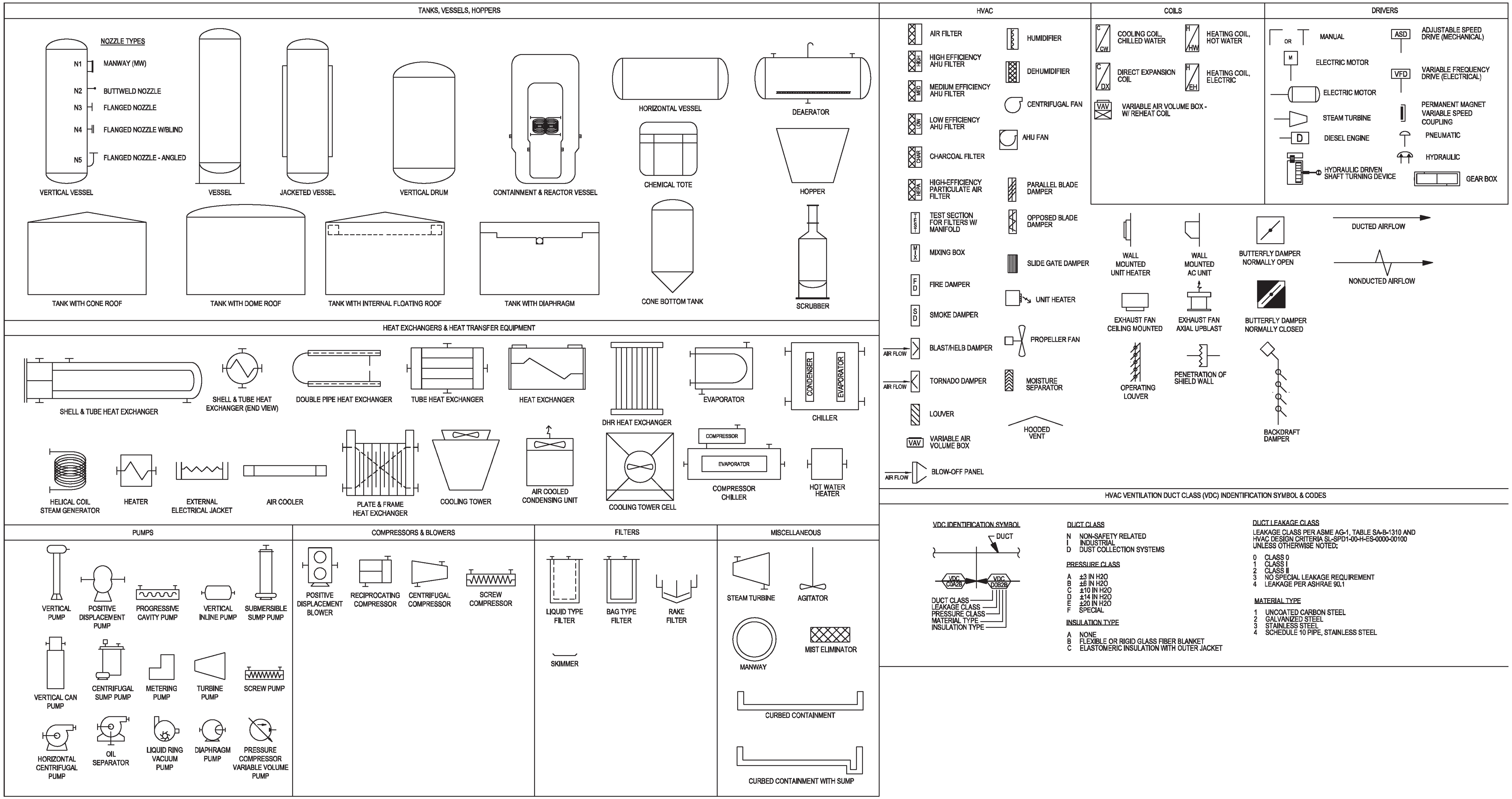
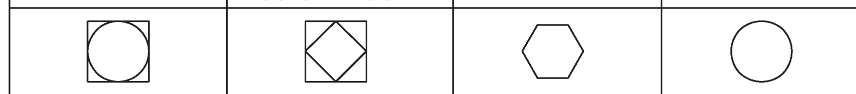
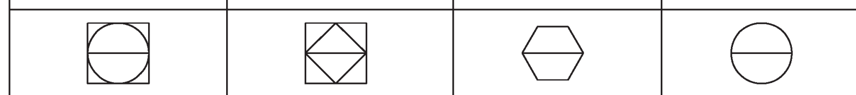
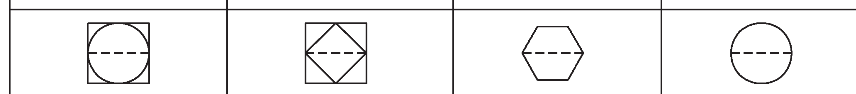
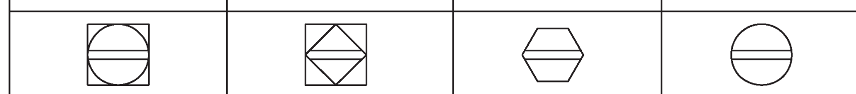
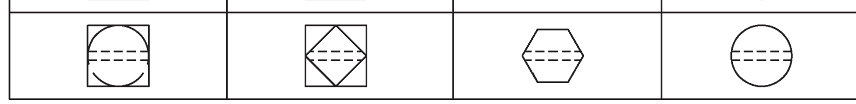


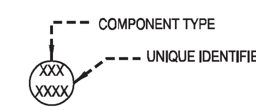
Figure 1.7-3d: Piping and Instrumentation Diagram Legends

INSTRUMENTATION DEVICE AND FUNCTION SYMBOLS (NOTE 1)				
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

FIRST LETTER: MEASURED VARIABLE	
LETTER	VARIABLE
A	ANALYSIS
B	BURNER/COMBUSTION
C	USER'S CHOICE
D	USER'S CHOICE
E	VOLTAGE
F	FLOW RATE
G	USER'S CHOICE
H	HAND
I	CURRENT
J	POWER
K	TIME
L	LEVEL
M	USER'S CHOICE
N	USER'S CHOICE
O	USER'S CHOICE
P	PRESSURE/VACUUM
Q	QUANTITY
R	RADIATION
S	SPEED/FREQUENCY
T	TEMPERATURE
U	MULTI VARIABLE
V	VIBRATION/MACHINERY ANALYSIS
W	WEIGHT/FORCE
X	UNCLASSIFIED
Y	EVENT/STATE/PRESENCE
Z	POSITION

SUBSEQUENT LETTERS: DEVICE FUNCTION(S)	
LETTER	FUNCTION
A	ALARM/SWITCH
C	CONTROLLER, CLOSED
D	DIFFERENTIAL
E	PRIMARY ELEMENT
F	RATIO
G	GAUGE, GLASS, OR OTHER VIEWING DEVICE
H	HIGH ALARM/SWITCH
I	INDICATOR/INDICATING
L	LOW ALARM/SWITCH
O	RESTRICTION ORIFICE, OPEN
P	TEST POINT
Q	QUANTITY OR TOTALIZER
R	RECORDER
S	SWITCH (ANY DIRECTION)
T	TRANSMITTER
V	VALVE
W	WELL OR PROBE
X	UNCLASSIFIED
Y	SOLENOID, RELAY, OR COMPUTING DEVICE

INSTRUMENT TAGGING



INSTRUMENT COMPONENT TYPES ARE CONSTRUCTED BY COMBINING A LETTER IDENTIFYING THE VARIABLE MEASURED ("FIRST LETTER") WITH ANY NUMBER OF SUBSEQUENT LETTERS CORRESPONDING TO THE FUNCTIONS PERFORMED BY THE DEVICE.

EXAMPLES:

- PIT: PRESSURE INDICATING TRANSMITTER
- ZC: POSITION CONTROLLER
- TE: TEMPERATURE ELEMENT (E.G. RTD OR THERMOCOUPLE)

1.8 Interfaces with Standard Design

This section addresses interface requirements between the NuScale Power Plant US460 standard design and the site-specific design. Section 1.2 identifies the structures, systems, and components that are included in the US460 standard design approval application. Figure 1.2-1 provides a representation of the overall facility and general boundaries between the US460 standard design and site-specific design.

Design assumptions related to site-specific design that are the responsibility of an applicant that references the NuScale Power Plant US460 standard design are 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 standard approved design, is identified throughout the Final Safety Analysis Report as COL information items. The COL items are not intended to duplicate all regulatory requirements that a licensee is required to comply with during construction and operation. Table 1.8-1 lists the COL information items, includes the COL information item text and identifies the section where the information item is located. The applicant addresses each COL information item in the section where it is located.

1.8.2 Departures

COL Item 1.8-1: An applicant that references the NuScale Power Plant US460 standard design will provide a list of departures from the approved design.

Table 1.8-1: Combined License Information Items

Item No.	Description of COL Information Item	Section
COL Item 1.1-1:	An applicant that references the NuScale Power Plant US460 standard design will identify the site-specific plant location.	1.1
COL Item 1.1-2:	An applicant that references the NuScale Power Plant US460 standard design will provide the schedules for completion of construction and commercial operation of each power module.	1.1
COL Item 1.4-1:	An applicant that references the NuScale Power Plant US460 standard design will identify the prime agents or contractors for the construction and operation of the nuclear power plant.	1.4
COL Item 1.7-1:	An applicant that references the NuScale Power Plant US460 standard design will list site-specific electrical and instrumentation and control drawings, diagrams, and legends, as applicable.	1.7
COL Item 1.7-2:	An applicant that references the NuScale Power Plant US460 standard design will list additional site-specific piping and instrumentation diagrams and legends as applicable.	1.7
COL Item 1.8-1:	An applicant that references the NuScale Power Plant US460 standard design will provide a list of departures from the approved design.	1.8
COL Item 1.9-1:	An applicant that references the NuScale Power Plant US460 standard design will review and address the conformance with regulatory criteria in effect six months before the submittal date of the application for the site-specific portions and operational aspects of the facility design.	1.9
COL Item 1.10-1:	An applicant that references the NuScale Power Plant US460 standard design will evaluate the potential hazards resulting from construction activities of the new NuScale facility to an operating nuclear power plant on a co-located site per 10 CFR 52.79(a)(31).	1.10
COL Item 2.0-1:	An applicant that references the NuScale Power Plant US460 standard design 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 applicant will demonstrate the acceptability of the site-specific values in the appropriate sections of its license application.	2.0
COL Item 2.1-1:	An applicant that references the NuScale Power Plant US460 standard design will describe the site geographic and demographic characteristics.	2.1
COL Item 2.2-1:	An applicant that references the NuScale Power Plant US460 standard design will describe nearby industrial, transportation, and military facilities. The 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:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design will investigate and describe the site-specific hydrologic characteristics for Section 2.4.1 through Section 2.4.14, except Section 2.4.8, Section 2.4.10, and Section 2.4.11.	2.4
COL Item 2.5-1:	An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific geology, seismology, and geotechnical characteristics for Section 2.5.1 through Section 2.5.5.	2.5
COL Item 3.3-1:	An applicant that references the NuScale Power Plant US460 standard design will confirm that nearby structures exposed to severe and extreme (tornado and hurricane) wind loads will not collapse and adversely affect the Seismic Category I portions of the Reactor Building or of the Control Building.	3.3
COL Item 3.4-1:	An applicant that references the NuScale Power Plant US460 standard design will confirm the final location of structures, systems, and components subject to flood protection. The final routing of piping, and site-specific tanks or water source tanks are placed in locations that would not cause flooding to the Reactor Building or Control Building.	3.4

Table 1.8-1: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 3.4-2:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design will determine the extent of waterproofing and damp proofing needed for the underground portion of the Reactor Building based on site-specific conditions. Additionally, the 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 applicant will describe how continued protection will be ensured.	3.4
COL Item 3.4-5:	An applicant that references the NuScale Power Plant US460 standard design 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 3.5-1:	An applicant that references the NuScale Power Plant US460 standard design will demonstrate the site-specific turbine missile parameters are bounded by the standard design 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; and location of the turbines with respect to the Reactor Building and Control Building.	3.5
COL Item 3.5-2:	An applicant that references the NuScale Power Plant US460 standard design will confirm the design-basis automobile missile parameters for the reference plant of velocity and maximum altitude of impact will not be exceeded as a result of extreme wind conditions that may occur in the vicinity of the site.	3.5
COL Item 3.5-3:	An applicant that references the NuScale Power Plant US460 standard design will evaluate site-specific hazards due to external events, such as turbine failures that can occur at nearby or co-located facilities, which may produce more energetic missiles than the design-basis missiles defined in Section 3.5.1.	3.5
COL Item 3.6-1:	An applicant that references the NuScale Power Plant US460 standard design 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). This analysis includes an evaluation of multi-module impacts in common pipe galleries, and evaluations regarding subcompartment pressurization. The as-built Pipe Rupture Hazards Analysis (PRHA) will show that the analysis of RXB piping bounds the possible effects of ruptures for the routings of lines outside of the RXB, or will perform the PRHA of the high- and moderate-energy lines outside the buildings.	3.6
COL Item 3.7-1:	An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific safe shutdown earthquake.	3.7
COL Item 3.7-2:	An applicant that references the NuScale Power Plant US460 standard design will provide site-specific time histories. In addition to the above criteria for cross correlation coefficients, time step and earthquake duration, strong motion durations, comparison to response spectra and power spectra density, the applicant will also confirm that site-specific ratios V/A and AD/V^2 (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

Table 1.8-1: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 3.7-3:	An applicant that references the NuScale Power Plant US460 standard design will include an analysis of the 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 the performance-based response spectra for the vertical direction.	3.7
COL Item 3.7-4:	An applicant that references the NuScale Power Plant US460 standard design will: • develop a site-specific strain-compatible soil profile. • confirm that the criterion for the minimum required response spectrum is satisfied. • determine whether the seismic site characteristics fall within the seismic design parameters such as soil layering assumptions used in the standard design, range of soil parameters, shear wave velocity values, and minimum soil bearing capacity.	3.7
COL Item 3.7-5:	An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific analysis that assesses the effects of soil separation. The applicant will confirm that the in-structure response spectra in the soil separation cases are bounded by the in-structure response spectra described in Section 3.7.2.	3.7
COL Item 3.7-6:	An applicant that references the NuScale Power Plant US460 standard design 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 3.7-7:	An applicant that references the NuScale Power Plant US460 standard design will confirm that nearby structures exposed to a site-specific safe shutdown earthquake will not collapse and adversely affect Seismic Category I portions of the Reactor Building and Control Building.	3.7
COL Item 3.7-8:	An applicant that references the NuScale Power Plant US460 standard design will demonstrate that the site-specific seismic demand is bounded by the Final Safety Analysis Report capacity for an empty dry dock condition.	3.7
COL Item 3.7-9:	An applicant that references the NuScale Power Plant US460 standard design will perform a soil-structure interaction analysis of the Reactor Building and the Control Building using the NuScale ANSYS models for those structures. The 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 3.7-10:	An applicant that references the NuScale Power Plant US460 standard design will determine the means and methods of lifting the bioshield. An applicant will demonstrate that bioshield components and connections can withstand the bioshield loads and appropriate load factors.	3.7

Table 1.8-1: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 3.7-11:	An applicant that references the NuScale Power Plant US460 standard design will prepare site-specific procedures for seismic instrumentation maintenance and post-earthquake activities. Administrative procedures define the maintenance and repair of the seismic instrumentation to keep the maximum number of instruments in-service during plant operations and shutdown. The procedures for post-earthquake activities must provide sufficient information to determine if the level of earthquake ground motion requiring shutdown has been exceeded and appropriate corrective actions to be taken if needed. Guidance for procedure development is in Regulatory Guide 1.12, “Nuclear Power Plant Instrumentation for Earthquakes,” Regulatory Guide 1.166, “Pre-Earthquake Planning, Shutdown, and Restart of a Nuclear Power Plant Following an Earthquake,” and EPRI Report 3002005284, “Guidelines for Nuclear Plant Response to an Earthquake” (Reference 3.7.4-1).	3.7
COL Item 3.8-1:	An applicant that references the NuScale Power Plant US460 standard design will provide the design of the reactor flange tool.	3.8
COL Item 3.8-2:	An applicant that references the NuScale Power Plant US460 standard design 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 the Effectiveness of Maintenance at Nuclear Power Plants." Monitoring is to include below grade walls, groundwater chemistry if needed, base settlements, and differential displacements.	3.8
COL Item 3.8-3:	An applicant that references the NuScale Power Plant US460 standard design will identify local stiff and soft spots in the foundation soil and address these in the design, as necessary.	3.8
COL Item 3.9-1:	An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific seismic analysis in accordance with Section 3.7.2. 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 certified seismic design response spectra, 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
COL Item 3.9-2:	An applicant that references the NuScale Power Plant US460 standard design will complete an assessment of piping systems inside the Reactor Building to determine the portions of piping to be tested for vibration, thermal expansion, and dynamic effects. 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 applicant may select the portions of piping in the 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 3.9-3:	An applicant that references the NuScale Power Plant US460 standard design will verify that evaluations are performed during 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 in “NuScale Comprehensive Vibration Assessment Program Analysis Technical Report,” TR-121353 is acceptable for this purpose. The applicant will update Section 3.9.2.1.1.1 to describe the results of this evaluation.	3.9

Table 1.8-1: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 3.9-4:	An applicant that references the NuScale Power Plant US460 standard design will provide applicable test procedures before the start of testing and will submit test and inspection results from the Comprehensive Vibration Assessment Program for the NuScale Power Module in accordance with Regulatory Guide 1.20.	3.9
COL Item 3.9-5:	An applicant that references the NuScale Power Plant US460 standard design 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 3.9-6:	An applicant that references the NuScale Power Plant US460 standard design will develop a Reactor Vessel Internals Reliability Program to address industry identified aging degradation mechanism issues.	3.9
COL Item 3.9-7:	An applicant that references the NuScale Power Plant US460 standard design 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 3.9-8:	An applicant that references the NuScale Power Plant US460 standard design will establish Preservice and Inservice Testing Programs. These programs are to be consistent with the requirements in the latest edition and addenda of the American Society of Mechanical Engineers (ASME) Operation and Maintenance (OM) Code incorporated by reference in 10 CFR 50.55a.	3.9
COL Item 3.9-9:	An applicant that references the NuScale Power Plant US460 standard design 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 3.9-10:	An applicant that references the NuScale Power Plant US460 standard design 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 3.11-1:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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 3.11-3:	An applicant that references the NuScale Power Plant US460 standard design will implement an Equipment Qualification Operational Program that incorporates the aspects in Section 3.11.5 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 3.12-1:	An applicant that references the NuScale Power Plant US460 standard design may use a piping analysis program other than the programs listed in Section 3.12.4; however, the applicant will implement a benchmark program using the models for the NuScale Power Plant US460 standard design.	3.12
COL Item 3.12-2:	An applicant that references the NuScale Power Plant US460 standard design will confirm that the site-specific seismic response is within the parameters specified in Section 3.7. An applicant may perform a site-specific piping stress analysis in accordance with the methodologies described in this section, as appropriate.	3.12

Table 1.8-1: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 3.13-1:	An applicant that references the NuScale Power Plant US460 standard design will provide an inservice inspection program for American Society of Mechanical Engineers Class 1, 2, and 3 threaded fasteners. The program will identify the applicable edition and addenda of American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI and ensure compliance with 10 CFR 50.55a.	3.13
COL Item 4.2-1:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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 design contains adequate overpressure protection features, including low temperature overpressure protection features.	5.2
COL Item 5.2-2:	An applicant that references the NuScale Power Plant US460 standard design 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 5.2-3:	An applicant that references the NuScale Power Plant US460 standard design 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 5.2-4:	An applicant that references the NuScale Power Plant US460 standard design 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 the American Society of Mechanical Engineers Operations and Maintenance Code, and will establish implementation milestones. If applicable, an applicant that references the NuScale Power Plant US460 standard design will identify the implementation milestone for the augmented inservice inspection program. The 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 5.2-5:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design will develop operating procedures to ensure that transients will not be more severe than those for which the reactor design adequacy had been demonstrated. These procedures will be based on material properties of the as-built reactor vessels.	5.3

Table 1.8-1: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 5.4-1:	An applicant that references the NuScale Power Plant US460 standard design 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 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:	An applicant that references the NuScale Power Plant US460 standard design 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 6.3-1:	An applicant that references the NuScale Power Plant US460 standard design 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 prohibit the introduction of coating materials into containment. • An inspection program to confirm containment vessel cleanliness before closing for normal power operation.	6.3
COL Item 6.4-1:	An applicant that references the NuScale Power Plant US460 standard design will comply with Regulatory Guide 1.78 Revision 2, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."	6.4
COL Item 6.6-1:	An applicant that references the NuScale Power Plant US460 standard design will develop Preservice Inspection and Inservice Inspection Program plans in accordance with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, and will establish the implementation milestones for the program. The applicant will identify the applicable edition of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code used in the program plan consistent with the requirements of 10 CFR 50.55a. The 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
COL Item 7.0-1:	An applicant that references the NuScale Power Plant US460 standard design will demonstrate 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:	An applicant that references the NuScale Power Plant US460 standard design will implement the life cycle processes for the operation phase for the instrumentation and controls systems, as defined in IEEE Std 1074-2006 and IEEE Std 1012-2004.	7.2
COL Item 7.2-2:	An applicant that references the NuScale Power Plant US460 standard design will implement the life cycle processes for the maintenance phase for the instrumentation and controls systems, as defined in IEEE Std 1074-2006 and IEEE Std 1012-2004.	7.2

Table 1.8-1: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 7.2-3:	An applicant that references the NuScale Power Plant US460 standard design will implement the life cycle processes for the retirement phase for the instrumentation and controls systems, as defined in Institute of IEEE Std 1074-2006 and IEEE Std 1012-2004. The Digital I&C Software Configuration Management Plan provides guidance for the retirement and removal of a software product from use.	7.2
COL Item 9.1-1:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design will provide the design of the spent fuel pool storage racks, including the structural dynamic and stress analyses, thermal hydraulic cooling analyses, criticality safety analysis, and material compatibility evaluation.	9.1
COL Item 9.1-3:	An applicant that references the NuScale Power Plant US460 standard design will provide the periodic testing plan for fuel handling equipment.	9.1
COL Item 9.1-4:	An applicant that references the NuScale Power Plant US460 standard design will describe the process for handling and receipt of critical loads including NPMs.	9.1
COL Item 9.1-5:	An applicant that references the NuScale Power Plant US460 standard design will provide a description of the program governing heavy loads handling. The program should address • operating and maintenance procedures. • inspection and test plans. • personnel qualification and operator training. • detailed description of the safe load paths for movement of heavy loads.	9.1
COL Item 9.3-1:	An applicant that references the NuScale Power Plant US460 standard design will submit a leakage control program for systems outside containment that contain (or might contain) accident source term radioactive materials following an accident. 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 9.5-1:	An applicant that references the NuScale Power Plant US460 standard design 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 10.3-1:	An applicant that references the NuScale Power Plant US460 standard design will provide a site-specific Secondary Water 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 97-06 at the time of the application.	10.3
COL Item 10.3-2:	An applicant that references the NuScale Power Plant US460 standard design 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 application.	10.3
COL Item 10.4-1:	An applicant that references the NuScale Power Plant US460 standard design will provide a secondary water chemistry analysis. This analysis must 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 Control Program described in Section 10.3.5, and it is compatible with the chemicals used.	10.4
COL Item 11.2-1:	An applicant that references the NuScale Power Plant US460 standard design 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

Table 1.8-1: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 11.2-2:	An applicant that references the NuScale Power Plant US460 standard design 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 11.2-3:	An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific evaluation using the site-specific source term and dilution flow for liquid effluent releases, and confirm that the discharge concentrations do not exceed the limits specified by 10 CFR 20, Appendix B, Table 2.	11.2
COL Item 11.2-4:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific cost-benefit analysis.	11.3
COL Item 11.3-2:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design will perform an analysis in accordance with Branch Technical Position 11-5 using the site-specific parameters.	11.3
COL Item 12.1-1:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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 12.3-6:	An applicant that references the NuScale Power Plant US460 standard design will develop the processes and programs associated with Objectives 5 and 6, to work in conjunction with design features, necessary to demonstrate compliance with 10 CFR 20.1406, and the guidance of Regulatory Guide 4.21.	12.3
COL Item 12.4-1:	An applicant that references the NuScale Power Plant US460 standard design will estimate doses to construction personnel from a co-located existing operating nuclear power plant that is not a NuScale Power Plant.	12.4

Table 1.8-1: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 12.5-1:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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 on-site operating organization, and (2) individuals holding management and supervisory positions in organizational units providing technical support to the on-site operating organization.	13.1
COL Item 13.1-2:	An applicant that references the NuScale Power Plant US460 standard design will provide a description of the proposed structure, functions, and responsibilities of the on-site 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:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design will provide a description of the Emergency Response facilities for management of overall licensee Emergency Response. The facility will meet the requirements of 10 CFR 52.79.	13.3
COL Item 13.3-2:	An applicant that references the NuScale Power Plant US460 standard design will provide a comprehensive Emergency Plan in accordance with 10 CFR 50 and 10 CFR 52.79(a)(21).	13.3
COL Item 13.4-1:	An applicant that references the NuScale Power Plant US460 standard design will provide site-specific information, including implementation milestones, for Operational Programs: • Inservice Inspection Programs (Section 5.2, Section 5.4, and Section 6.6) • Inservice Testing Programs (Section 3.9 and Section 5.2) • Environmental Qualification Program (Section 3.11) • Preservice Testing Program (Section 3.9.6 and Section 5.4) • Preservice Testing Program (Section 3.9.6 and Section 5.2) • Containment Leakage Rate Testing Program (Section 6.2) • Fire Protection Program (Section 9.5.1) • Process and Effluent Monitoring and Sampling Program (Section 11.5) • Radiation Protection Program (Section 12.5) • Non-Licensed Plant Staff Training Program (Section 13.2) • Reactor Operator Training Program (Section 13.2) • Reactor Operator Requalification Program (Section 13.2) • Emergency Planning (Section 13.3) • Process Control Program (Section 11.4) • Security (Section 13.6) • Quality Assurance Program (Section 17.5) • Maintenance Rule (Section 17.6) • Initial Test Program (Section 14.2)	13.4

Table 1.8-1: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 13.5-1:	An applicant that references the NuScale Power Plant US460 standard design 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, "Quality Assurance Program Requirements (Operation)," Revision 3.	13.5
COL Item 13.5-2:	An applicant that references the NuScale Power Plant US460 standard design 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 applicant will identify the group within the operating organization responsible for maintaining these procedures.	13.5
COL Item 13.5-3:	An applicant that references the NuScale Power Plant US460 standard design will describe the process to manage the development, review and approval of 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 applicant will describe the classification system for these procedures, and the general format and content of the different classifications.	13.5
COL Item 13.5-4:	An applicant that references the NuScale Power Plant US460 standard design 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 applicant will identify the group within the operating organization responsible for maintaining these procedures.	13.5
COL Item 13.5-5:	An applicant that references the NuScale Power Plant US460 standard design will provide a plan for the development, implementation, and control of emergency operating procedures, including preliminary schedules for preparation and target dates for completion. Additionally, the applicant will identify the group within the operating organization responsible for maintaining these procedures.	13.5
COL Item 13.5-6:	An applicant that references the NuScale Power Plant US460 standard design 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 will 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 13.5-7:	An applicant that references the NuScale Power Plant US460 standard design 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 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:	An applicant that references the NuScale Power Plant US460 standard design 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 • elements of the physical security system not located within the nuclear island and structures	13.6

Table 1.8-1: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 13.6-2:	An applicant that references the NuScale Power Plant US460 standard design will be responsible for the requirements described in Table 5-1 of TR-118318, “NuScale Design of Physical Security Systems” (Reference 13.6-1).	13.6
COL Item 13.6-3:	An applicant that references the NuScale Power Plant US460 standard design will provide a secondary alarm station that is equal and redundant to the central alarm station.	13.6
COL Item 13.6-4:	An applicant that references the NuScale Power Plant US460 standard design will provide a description of the Access Authorization Program.	13.6
COL Item 13.6-5:	An applicant that references the NuScale Power Plant US460 standard design will provide a Cyber Security Plan.	13.6
COL Item 14.2-1:	An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific organizations that manage, supervise, or execute the Initial Test Program, including the associated training requirements.	14.2
COL Item 14.2-2:	An applicant that references the NuScale Power Plant US460 standard design will develop the Startup Administration Manual that will contain the administrative procedures and requirements that control the activities associated with the Initial Test Program. The applicant will provide a milestone for completing the Startup Administrative Manual and making it available for Nuclear Regulatory Commission inspection.	14.2
COL Item 14.2-3:	An applicant that references the NuScale Power Plant US460 standard design will identify the specific operator training to be conducted during low-power testing related to the resolution of Three Mile Island Action Plan Item I.G.1, as described in NUREG-0660, NUREG-0694, and NUREG-0737.	14.2
COL Item 14.2-4:	An applicant that references the NuScale Power Plant US460 standard design will provide a schedule for the Initial Test Program.	14.2
COL Item 14.2-5:	An applicant that references the NuScale Power Plant US460 standard design will provide a test abstract for the potable water system pre-operational testing.	14.2
COL Item 14.2-6:	An applicant that references the NuScale Power Plant US460 standard design will provide a test abstract for the seismic monitoring system pre-operational testing.	14.2
COL Item 14.3-1:	An applicant that references the NuScale Power Plant US460 standard design will provide the site-specific selection methodology and inspections, tests, analyses, and acceptance criteria for emergency planning.	14.3
COL Item 14.3-2:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design, 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

Table 1.8-1: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 17.4-1:	An applicant that references the NuScale Power Plant US460 standard design will describe the Reliability Assurance Program conducted during the operations phases of the plant's 60-year design life.	17.4
COL Item 17.4-2:	An applicant that references the NuScale Power Plant US460 standard design will identify site-specific structures, systems, and components within the scope of the Reliability Assurance Program.	17.4
COL Item 17.4-3:	An applicant that references the NuScale Power Plant US460 standard design 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 18.5-1:	An applicant that references the NuScale Power Plant US460 standard design will address the staffing and qualifications of non-licensed operators.	18.5
COL Item 18.12-1:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design will identify and describe specific risk-informed applications being implemented during the COL application phase.	19.1
COL Item 19.1-3:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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 standard design.	19.1
COL Item 19.1-8:	An applicant that references the NuScale Power Plant US460 standard design will confirm the validity of the “key assumptions” and data used in the standard design approval application PRA and modify, as necessary, for applicability to the as-built, as-operated PRA.	19.1
COL Item 19.2-1:	An applicant that references the NuScale Power Plant US460 standard design 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:	An applicant that references the NuScale Power Plant US460 standard design 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

Table 1.8-1: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 19.2-3:	An applicant that references the NuScale Power Plant US460 standard design will evaluate severe accident mitigation design alternatives screened as not required for the standard design.	19.2
COL Item 19.3-1:	An applicant that references the NuScale Power Plant US460 standard design will identify site-specific Regulatory Treatment of Nonsafety Systems structures, systems, and components and applicable process controls.	19.3

1.9 Conformance with Regulatory Criteria

This section provides a guide to conformance with regulatory criteria in effect 6 months before the anticipated docket date as listed below.

Table 1.9-1, Conformance Status Legend

Table 1.9-2, Conformance with Regulatory Guides

Table 1.9-3, Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard

Table 1.9-4, Conformance with Interim Staff Guidance

Table 1.9-5, Conformance with Three Mile Island 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 (Commission papers (SECYs) and associated Staff Requirements Memoranda)

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: An applicant that references the NuScale Power Plant US460 standard design will review and address the conformance with regulatory criteria in effect six months before the submittal date of the application for the site-specific portions and operational aspects of the facility design.

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 (ESP) 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 - SSC that are 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 an applicant referencing the NuScale Power US460 standard design approval (SDA) or licensee. • 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 TMI action items may be identified and justified as "not technically relevant" to the design consistent with 10 CFR 52.137(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 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	Title	Rev.	Conformance Status	Comments	Section
1.6	Safety Guide 6 - Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	0	Not Applicable	The onsite electrical AC power systems do not contain Class 1E distribution systems.	Not Applicable
1.7	Control of Combustible Gas Concentrations in Containment	3	Partially Conforms	The design complies with the intent of RG 1.7 regulatory positions that address atmosphere mixing, hydrogen gas production, and containment structural integrity. However, the design deviates from the positions on hydrogen and oxygen monitors. The design includes a passive autocatalytic recombiner (PAR) that is sized to limit oxygen concentrations to a level that does not support combustion (less than four percent), this results in an inert containment atmosphere. The NuScale design supports an exemption to 10 CFR 50.44(c)(4).	6.2.5
1.8	Qualification and Training of Personnel for Nuclear Power Plants	4	Not Applicable	This guidance governs site-specific programmatic and operational activities that are the responsibility of the applicant or licensee.	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 instrument lines penetrate the NuScale Power Module (NPM) containment.	Not Applicable
1.12	Nuclear Power Plant Instrumentation for Earthquakes	3	Partially Conforms	Selection of specific equipment is the responsibility of the applicant or licensee. In addition, seismic instrumentation cannot be installed inside the containment, so Section 3.7.4 indicates seismic instrumentation is installed in the Reactor Building (RXB).	3.7.4 12.3

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.13	Spent Fuel Storage Facility Design Basis	2	Partially Conforms	The design 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, Seismic Category I makeup line. The design deviates from the guidance of Regulatory Position C.9, Pool Cooling, in its makeup system design. The size and reliability of the makeup source (the water inventory of the UHS) ensures greater than 30 days of makeup for the SFP following a loss of normal cooling and makeup from the PCWS, without operator action. Over-pressurization vents in the pool hall provide protection for safety-related components from high temperatures and moisture levels.	9.1.2 9.1.3 9.2.5 3.5.2
1.14	Reactor Coolant Pump Flywheel Integrity	1	Not Applicable	This guidance is applicable 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	4	Conforms	The first operational NPM is classified as a prototype in accordance with RG 1.20. Thus, the portions of this RG that 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 and the non-prototype portions of the RG apply. The identification of departures from RG 1.20 is the responsibility of the applicant or licensee.	3.9 5.4 14.2
1.21	Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste	3	Partially Conforms	Site-specific, programmatic, and operational aspects are the responsibility of the applicant or licensee.	3.1 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 governs site-specific activites that are the responsibility of the applicant or licensee.	Not Applicable

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure	0	Not Applicable	This guidance governs site-specific activites that are the responsibility of the applicant or licensee.	Not Applicable
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	6	Conforms	The quality group classification from RG 1.26 applicable to a specific component is described throughout the FSAR.	3.2 3.12 5.2 5.4 6.2 6.6 9.1 9.2 9.3 10.3 10.4 17.5
1.27	Ultimate Heat Sink for Nuclear Power Plants	3	Not Applicable	This guidance does not apply to plants that use a passive cooling system to transfer heat to the UHS. The NuScale design uses a passive cooling system.	Not Applicable
1.28	Quality Assurance Program Criteria (Design and Construction)	5	Partially Conforms	NuScale conforms with RG 1.28 Rev 4, but follows guidance in RG 1.28 Rev 5 in the following instances: • Compliance with 2011 Nuclear Information and Records Management Association technical guides for electronic records • Use of NEI 14-05A Rev 1 as a basis for laboratory calibration requirements • Qualifications for lead auditors based on audit participation Conformance with Rev 4 is maintained to align with SRP 17.5. Site-specific, programmatic, and operational aspects are the responsibility of the applicant or licensee.	6.3 7.0 7.2 17.5
1.29	Seismic Design Classification for Nuclear Power Plants	6	Conforms	The seismic classification from RG 1.29 applicable to a specific component is described throughout the FSAR.	3.2

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.30	Safety Guide 30 - Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	0	Not Applicable	This guidance endorses IEEE Std. 336-1971 for the installation, inspection, and testing of instrumentation and electric equipment. The 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	Not Applicable	This guidance is not applicable to the offsite and onsite AC and DC power systems.	Not Applicable
1.33	Quality Assurance Program Requirements (Operation)	3	Not Applicable	This guidance is the responsibility of the applicant or licensee.	Not Applicable
1.34	Control of Electroslag Weld Properties	1	Not Applicable	Electroslag welding is in accordance with RG 1.43.	Not Applicable
1.35.1	Determining Prestressing Forces for Inspection of Prestressed Concrete Containments	0	Not Applicable	The NuScale design uses a steel containment vessel (CNV) (i.e., does not use concrete in containment vessel design).	Not Applicable
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel	1	Not Applicable	The NuScale design does not use nonmetallic thermal insulation on reactor coolant pressure boundary (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	0	Not Applicable	This guidance is applicable to on-site electric power systems designed in accordance with RGs 1.6 and 1.32.	Not Applicable
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components	1	Conforms	None.	5.2 5.3 6.1

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.44	Control of the Processing and Use of Stainless Steel	1	Partially Conforms	This guidance is applicable except for its specification of applying RG 1.37 for cleaning and flushing of finished surfaces. RG 1.37 was withdrawn by the NRC.	4.5 5.2 5.3 6.1
1.45	Guidance on Monitoring and Responding to Reactor Coolant System Leakage	1	Conforms	None.	3.15.2 9.3 11.5 14.2
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
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 (DBA). 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 an NPP, 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.	5.4 6.2 6.3 7.1 15.1 15.2 15.5

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.54	Service Level I, II, III, and In-Scope License Renewal Protective Coatings Applied to Nuclear Power Plants	3	Partially Conforms	This guidance is applicable except for operational aspects (e.g., maintenance of safety-related coatings) that are the responsibility of the applicant or licensee.	11.2
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components	2	Partially Conforms	This guidance is applicable except for reference to 10 CFR 50.34(f)(3)(v), because per 10 CFR 50.34(f) and 10 CFR 52.137(a)(8), an applicant does not have to show compliance with 10 CFR 50.34(f)(3)(v). Use of typical reactor pressure vessel (RPV) load combinations for Class 1 vessels is more applicable to the CNV 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. The load combination of Position C.I.B.3(c)(iv) is not applicable because the design ensures an inert containment atmosphere; the load combination of Position C.I.B.3(c)(v) is not applicable because post-accident inerting does not add pressure to containment.	3.8.2 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 this guidance, an alternative damping value for the NPM substructure was determined.	3.7.1 3.7.2 3.8.2 3.12
1.62	Manual Initiation of Protective Actions	1	Conforms	None.	7.2

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.63	Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants	3	Partially Conforms	The portion of this 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.	3.11 6.2 8.1 8.3
1.65	Materials and Inspections for Reactor Vessel Closure Studs	1	Partially Conforms	This guidance is applicable except that inservice inspection is the responsibility of the applicant or licensee. The RPV bolting material is described in Table 5.2-3. 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 structures, systems, and components design features not in the NuScale design. Site-specific program implementation activities are the responsibility of the applicant or licensee.	5.4 8.2 8.3 14.2 14.3
1.68.1	Initial Test Program of Condensate and Feedwater Systems for Light-Water Reactors	2	Partially Conforms	This guidance 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 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 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 applicant or licensee.	3.8.4 12.3 14.2

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)	3	Not Applicable	SRP and NuScale Design Specific Review Standards (DSRS) are used.	Not Applicable
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 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	This 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	Partially Conforms	This guidance endorses IEEE Std 384-1992; however, some systems are designed per IEEE Std 384-2008, which was endorsed by RG 1.97, Revision 3.	7.1 7.2 9.5
1.76	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants	1	Conforms	Bounding or greater than design parameters postulated by RG 1.76 are used. Confirming the characteristics is the responsibility of the applicant or licensee.	2.3 3.3 3.5 3.8.4
1.78	Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	2	Partially Conforms	Aspects of this guidance 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 redundant atmospheric dispersion factors) or specify operational, programmatic emergency planning activities. These aspects are the responsibility of the applicant or licensee.	6.4 9.4

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors	2	Partially Conforms	The intent of this guidance is applicable to the NuScale design, but the literal language refers to SSC design features not in the NuScale design. For example, the emergency core cooling system (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 is performed on the ECCS in a manner that satisfies the intent of this guidance. Preoperational test implementation activities are the responsibility of the applicant or licensee.	6.3 14.2
1.79.1	Initial Test Program of Emergency Core Cooling Systems for New Boiling-Water Reactors	0	Not Applicable	This guidance is applicable to BWRs only.	Not Applicable
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants	1	Not Applicable	This guidance is applicable to emergency and shutdown electric systems, which are not included in the design.	Not Applicable

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.82	Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident	5	Partially Conforms	The 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 design complies with the guidance with respect to debris generation, debris transport, coating debris, latent debris, downstream, and chemical effects. The 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 design does not require operator action to mitigate debris accumulation. The design does not comply with regulatory position C1.1 with the exception that the design 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).	6.2 6.3

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

RG	Title	Rev.	Conformance Status	Comments	Section
				Positions C1.1.3 and C1.1.4 are not applicable because the 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 design does not comply with regulatory position C.1.2 (alternative water sources for inoperable strainers). The 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 design The design does not comply with regulatory positions C1.3.7 (upstream effects) or C.1.3.9 (strainer structural integrity). The design does not comply with regulatory position C1.3.12 (prototypical head loss testing). The 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 design does not comply with regulatory position C.3.	
1.84	Design, Fabrication, and Materials Code Case Acceptability, ASME Section III	39	Conforms	None.	3.12 3.13 5.2 6.1
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 guidance 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

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.89	Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants	1	Partially Conforms	This guidance 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); and (2) reference to RG 1.4 for source term, because the source term provisions of RG 1.4 are superseded by RG 1.183 for new reactors.	3.11 Appendix 3C
1.90	Inservice Inspection of Pre-stressed Concrete Containment Structures with Grouted Tendons	2	Not Applicable	This guidance is applicable to LWR designs that incorporate a pre-stressed concrete containment structure with grouted tendons. The CNV is steel (i.e., does not use concrete or grouted tendons in its design).	Not Applicable
1.91	Evaluations of Explosions Postulated to Occur at Nearby Facilities and on Transportation Routes Near Nuclear Power Plants	3	Not Applicable	This guidance governs the performance of site-specific evaluations and is the responsibility of the applicant or licensee.	Not Applicable
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis	3	Conforms	None.	3.7.2 3.7.3 3.8.4 3.9 3.10 3.12
1.93	Availability of Electric Power Sources	1	Not Applicable	This guidance 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 guidance is applicable only to BWR designs.	Not Applicable
1.97	Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants	5	Partially Conforms	The design satisfies power supply requirements in Section 6.6 of IEEE Std 497-2016 for Type B and C variables with augmented DC power rather than with Class 1E. Type F variables are also powered from the EDAS with the UHS having additional independent battery backup. The portions of RG 1.97 dealing with 10 CFR 50.34(f)(2)(xix) are addressed in Section 19.2.	3.11 5.4 7.1 7.2 11.5 12.3 19.2

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

RG	Title	Rev.	Conformance Status	Comments	Section
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 guidance is applicable only to BWR designs.	Not Applicable
1.99	Radiation Embrittlement of Reactor Vessel Materials	2	Partially Conforms	The design uses austenitic stainless steel in the reactor pressure vessel beltline. Austenitic stainless steel is highly ductile and less susceptible to the effects of neutron embrittlement than ferritic steel. This RG does not apply to austenitic stainless steel. TR-130721, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel."	5.3
1.100	Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants	4	Partially Conforms	This guidance 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 applicant. RG 1.100 endorses ASME QME-1 2017. 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 3.11 5.2 Appendix 3C
1.101	Emergency Response Planning and Preparedness for Nuclear Power Reactors	6	Not Applicable	This guidance is the responsibility of the 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	Not Applicable	The design assumes the NPP is located above the probable maximum flood height (including wind induced wave run-up).	2.4 3.4 3.8.4

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.105	Setpoints for Safety-Related Instrumentation	4	Conforms	Chapter 15 analyses use the safety-related setpoints described in Chapter 7.	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 guidance governs the application of thermal overload protection devices to ensure 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 guidance is applicable only to LWR designs that use a prestressed concrete containment structure. The CNV 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 guidance is applicable except for specification of site-specific information (e.g., meteorological data), which is the responsibility of the applicant or licensee.	11.2 11.3
1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors	1	Partially Conforms	This guidance is applicable except for aspects related to performance of a site-specific cost-benefit analysis that is the responsibility of the 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	Not Applicable	This guidance applies to applications for an operating license or construction permit.	Not Applicable
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Nuclear Power Reactors	1	Partially Conforms	This guidance is applicable except for specification of site-specific information (e.g., meteorological data) that is the responsibility of the applicant or licensee.	2.3 11.2 11.3

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I	1	Not Applicable	This guidance governs analysis of 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 pool surge control subsystem 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	This guidance is applicable except for site-specific guidance that is the responsibility of the applicant or licensee. Consistent with the discussion in RG 1.114, Section B.1, the ability of the 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 applicants.	18.5
1.115	Protection Against Turbine Missiles	2	Conforms	None.	3.5
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 RXB or the Seismic Category I portion of the Control Building is the responsibility of the applicant or licensee.	3.5 9.1.2
1.118	Periodic Testing of Electric Power and Protection Systems	3	Partially Conforms	This guidance is applicable except for site-specific guidance that is the responsibility of the applicant or licensee.	7.2 14.2
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes	0	Conforms	None.	5.4
1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components	1	Conforms	An applicant that references the NuScale Power Plant US460 standard design will confirm that the site-specific seismic demands of the standard design for critical SSC are bounded by the corresponding approved design seismic demands.	3.7.2
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Supports	3	Conforms	None.	3.9

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants	2	Not Applicable	The design does not require hydraulic structures.	Not Applicable
1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification	2	Conforms	None.	4.2
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants	2	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 applicant or licensee.	Not Applicable
1.128	Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	2	Not Applicable	The EDAS uses valve-regulated lead-acid batteries; thus IEEE Std 1187-2013 is applied.	8.3
1.129	Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	3	Not Applicable	The EDAS uses valve-regulated lead-acid batteries. The design applies IEEE Std 1188-2005 with the 2014 amendment.	8.3
1.130	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Supports	3	Conforms	None.	3.9
1.132	Geologic and Geotechnical Site Characterization Investigations for Nuclear Power Plants	3	Not Applicable	This guidance governs site investigations performed as part of site selection that are the responsibility of the 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

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.134	Medical Assessment of Licensed Operators or Applicants for Operator Licenses at Nuclear Power Plants	4	Not Applicable	This guidance governs site-specific operational program activities that are the responsibility of the applicant or licensee.	Not Applicable
1.136	Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments	4	Not Applicable	This guidance is applicable only to LWR designs that use concrete containments. The design uses a steel CNV.	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	3	Not Applicable	This guidance is related to site-specific laboratory investigation activities that are the responsibility of the 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	3	Partially Conforms	Design-related aspects of this guidance are applicable. The design deviates from the guidance of RG 1.140 by using the 2019 edition of ASME AG-1. An evaluation of the differences between the 2019 edition and the 2009 edition, which is endorsed by RG 1.140, shows that using the 2019 edition does not reduce the technical and safety margin requirements of system components. Aspects related to construction, testing, and repairs are the responsibility of the applicant or licensee.	9.4 11.3 12.3 14.2
1.141	Containment Isolation Provisions for Fluid Systems	1	Conforms	The 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 containment vessel isolation valves outside containment that serve non-ESF process systems.	6.2

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)	3	Partially Conforms	The intent of this guidance is applicable to concrete structures. The language endorses ACI 349-13 with exceptions. Regulatory Position 5.1: The design considers the load factors in ACI 349-13 instead of the revised values in RG 1.142 because the values in ACI 349-13 more closely align with recent design codes for developing load combinations. Regulatory Position 5.7: The design methodology applies the Φ =0.75 for ductile or non-ductile (brittle) limit state. The reason is that walls are designed to remain elastic and not incur significant ductility. Thus, there is no need to delay non-ductile failure modes and same value strength reduction factor (Φ) is applied for shear critical members.	3.5 3.8.4
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 guidance related to steam generator blowdown systems are not applicable to the design. Radioactive waste management system design criteria specified in this guidance are applicable. Construction, installation, and testing criteria are the responsibility of the applicant or licensee.	3.2 3.3 3.5 3.7.2 9.2.6 11.2 11.3 11.4
1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants	1	Not Applicable	This guidance does not include modeling building wake effects. For the short distances that may be used for the exclusion area boundary and the low population zone, RG 1.194 is used to determine representative atmospheric dispersion factors.	Not Applicable
1.147	Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1	20	Partially Conforms	This guidance is applicable except for performance of inservice inspections per the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, which is the responsibility of the applicant or licensee.	5.2 6.6

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

RG	Title	Rev.	Conformance Status	Comments	Section
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 applicant or licensee.	Not Applicable
1.151	Instrument Sensing Lines	2	Partially Conforms	This guidance governs design and installation of safety-related instrument sensing lines in nuclear power plants. The aspects of this guidance regarding installation criteria are the responsibility of the applicant or licensee.	3.2 7.2
1.152	Criteria for Use of Computers in Safety Systems of Nuclear Power Plants	3	Partially Conforms	The instrumentation and controls (I&C) development lifecycle differs from the conceptual waterfall lifecycle in RG 1.152. 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
1.153	Criteria for Safety Systems	1	Conforms	This guidance is applicable to safety-related I&C.	3.11 7.1 7.2
1.155	Station Blackout	0	Partially Conforms	The design conforms to the aspects of the guidance as it pertains to passive plant designs; however, the guidance is not used to demonstrate compliance with 10 CFR 50.63.	6.2 8.1
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	0	Not Applicable	Best estimate calculations are not used.	Not Applicable
1.158	Qualification of Safety-Related Vented Lead-Acid Lead Storage Batteries for Nuclear Power Plants	1	Not Applicable	This guidance is not applicable because the augmented DC power system batteries are non-Class 1E. However, Guidance in IEEE Standard 535-2013 (endorsed by RG 1.158) is used as supplemental guidance to IEEE Standard 323-2003 to address aging of valve regulated lead acid batteries.	3.11
1.159	Assuring the Availability of Funds for Decommissioning Nuclear Reactors	2	Not Applicable	Decommissioning funding activities are the responsibility of the licensee.	Not Applicable

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	4	Not Applicable	Monitoring the effectiveness of maintenance activities is the responsibility of the licensee.	Not Applicable
1.161	Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb	0	Not Applicable	The reactor pressure vessel is made of austenitic stainless steel, which does not undergo impact testing per ASME BPVC Section III, NB-2311, and thus cannot have a Charpy upper-shelf energy. The design supports an exemption from 10 CFR 50.60, so this guidance is not applicable.	Not Applicable
1.162	Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels	0	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 applicant or licensee.	Not Applicable
1.163	Performance-Based Containment Leak-Test Program	0	Partially Conforms	The design supports an exemption from 10 CFR 50 Appendix A, GDC 52 and thus 10 CFR 50 Appendix J Type A tests. The CNV design allows testing and inspection, other than as anticipated by GDC 52. 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, RG 1.163, and NEI 94-01. The design accommodates the 10 CFR 50, Appendix J, test method frequencies of Option A or Option B. This RG is the responsibility of an applicant or licensee that seeks to implement Option B.	6.2
1.164	Dedication of Commercial-Grade Items for Use in Nuclear Power Plants	0	Not Applicable	The guidance is applicable to the design but is the responsibility of an applicant referencing the NuScale US460 standard design or licensee.	17.5
1.166	Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post Earthquake Actions	1	Not Applicable	This guidance governs programmatic activities (earthquake planning and post-earthquake actions) that are the responsibility of the applicant or licensee.	Not Applicable

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

RG	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 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 I&C development lifecycle.	7.2
1.169	Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	Partially Conforms	For this guidance, the requirements of IEEE 828-2005 are tailored to the 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 I&C development lifecycle.	7.2
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 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 I&C development lifecycle. Exceptions are taken to some of the administrative mandatory requirements in the standard that conflict with established Engineering or quality documentation requirements.	7.2
1.171	Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	Partially Conforms	Exceptions are taken to some of the administrative mandatory requirements in IEEE 1008-1987 that conflict with established Engineering or quality documentation requirements.	7.2
1.172	Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	Partially Conforms	Exceptions are taken to some of the administrative mandatory requirements in IEEE 830-1998 standard that conflict with established Engineering or quality documentation requirements.	7.2

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

RG	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 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 I&C development lifecycle. Exceptions are taken to some of the administrative mandatory requirements in the standard that conflict with established Engineering or quality documentation requirements.	7.2
1.174	An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis	3	Not Applicable	This guidance is applicable to licensees seeking changes in licensing basis, which is the responsibility of the licensee.	Not Applicable
1.175	Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing	1	Not Applicable	This guidance is applicable to licensees seeking change to licensing basis using a risk-informed approach, which is the responsibility of the applicant or licensee.	Not Applicable
1.177	Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications	2	Not Applicable	This guidance applies to existing licensees seeking NRC approval of changes to their plant-specific technical specifications. NuScale considered this guidance, as appropriate, in risk-informed technical specification development.	16.1
1.178	Plant-Specific, Risk-Informed Decisionmaking for Inservice Inspections of Piping	2	Not Applicable	This guidance 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 applicant or licensee.	Not Applicable
1.179	Standard Format and Content of License Termination Plans for Nuclear Power Reactors	2	Not Applicable	This guidance governs site-specific decommissioning and license termination planning and implementation activities that are the responsibility of the applicant or licensee.	Not Applicable
1.180	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems	2	Partially Conforms	Aspects of this guidance related to the design of SSC to address effects of electromagnetic and radio-frequency interference are applicable. Aspects of this guidance related to the design of site-specific SSC and installation and testing practices for addressing these effects on safety-related I&C systems are the responsibility of the applicant or licensee.	3.11 Appendix 3C 7.2 9.5

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.181	Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)	0	Not Applicable	This guidance governs site-specific reporting activities that are the responsibility of the 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 design, the safety analysis shows that core damage does not occur during a design-basis event. Thus, the RG 1.183 guidance is partially applicable to the dose consequence analysis. The basis and justification for departures from the RG 1.183 guidance for the limiting dose consequence analysis are provided in topical report TR-0915-17565-P-A, Rev. 4. The alternative source term non-loss-of-coolant accident (LOCA) or transient-specific guidance of RG 1.183 for Chapter 15 events is used.	3.1 3.11 6.4 12.2 15.0.3 15.10
1.184	Decommissioning of Nuclear Power Reactors	1	Not Applicable	This guidance governs site-specific decommissioning planning and implementation activities that are the responsibility of the applicant or licensee.	Not Applicable
1.185	Standard Format and Content for Post-Shutdown Decommissioning Activities Report	1	Not Applicable	This guidance governs site-specific decommissioning planning activities that are the responsibility of the applicant or licensee.	Not Applicable
1.186	Guidance and Examples for Identifying 10 CFR 50.2 Design Bases	0	Not Applicable	This guidance endorses NEI 97-04 Appendix B, which is the responsibility of the applicant or licensee.	Not Applicable
1.187	Guidance for Implementation of 10 CFR 50.59, “Changes, Tests, and Experiments”	3	Not Applicable	This guidance implements change process requirements that are the responsibility of the applicant or licensee.	Not Applicable
1.188	Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses	2	Not Applicable	This guidance is applicable to operating reactor licensees seeking to renew their operating licenses.	Not Applicable
1.189	Fire Protection for Nuclear Power Plants	4	Partially Conforms	This guidance 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 applicant or licensee.	3.1 3.2 9.4 9.5.1 Appendix 9A 19.1

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.190	Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence	0	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-118976-P, "Fluence Calculation Methodology and Results."	5.3
1.191	Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown	1	Not Applicable	This guidance 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	4	Not Applicable	This guidance governs implementation of inservice testing that is the responsibility of the applicant or licensee.	Not Applicable
1.193	ASME Code Cases Not Approved for Use	7	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	0	Conforms	None.	15.0.3
1.195	Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors	0	Not Applicable	This guidance 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
1.196	Control Room Habitability at Light-Water Nuclear Power Reactors	1	Partially Conforms	Aspects of this guidance related to control room habitability design within the scope of the standard plant design are applicable. References to ESF ventilation systems are not applicable to the NuScale design. The control room habitability system (CRHS) 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 applicant or licensee.	3.8.4 6.4 18.7
1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors	0	Not Applicable	This guidance governs inleakage testing activities that are the responsibility of the applicant or licensee.	Not Applicable

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.198	Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites	0	Not Applicable	This guidance governs evaluations that are the responsibility of the applicant or licensee.	Not Applicable
1.199	Anchoring Components and Structural Supports in Concrete	1	Partially Conforms	The intent of this guidance is applicable, but the specific language endorses Appendix D of ACI 349-2013, and loads and combinations provided in RG 1.142 Positions 5 and 6. The design considers the load factors in ACI 349-13 for safety-related concrete structures, which deviate from RG 1.142 and more closely align with recent design codes that also include load combinations.	3.8.4
1.200	Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities	3	Conforms	As referenced in SRP 19.0 with regard to PRA acceptability.	17.4 19.1
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. A risk-informed, performance-based approach to safety classification that blends the strengths of deterministic engineering judgment and probabilistic methods is used. Specifically, the 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. 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 is applied.	3.2 17.4

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.202	Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors	0	Not Applicable	This guidance implements regulatory requirements for decommissioning cost estimates that are applicable only to licensees.	Not Applicable
1.203	Transient and Accident Analysis Methods	0	Conforms	None.	6.2 6.3 15.0.2
1.204	Guidelines for Lightning Protection of Nuclear Power Plants	0	Conforms	None.	3.8.4 7.0 7.2
1.205	Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants	2	Not Applicable	This guidance 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 such a program would be the responsibility of applicants or licensees that reference the design and that elect to implement the provisions of 10 CFR 50.48(c).	Not Applicable
1.206	Applications for Nuclear Power Plants	1	Partially Conforms	The SDA application was developed in consideration of the portions of this guidance that are relevant. Other aspects of this guidance relevant to the license application are the responsibility of the applicant.	Not applicable
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	1	Conforms	None.	3.8.2 3.9 3.12
1.208	A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion	0	Not Applicable	This guidance is for development of site-specific ground motion response spectra and is the responsibility of the applicant or licensee.	Not Applicable
1.209	Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants	0	Conforms	None.	3.11 7.2

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.210	Qualification of Safety-Related Battery Chargers and Inverters for Nuclear Power Plants	0	Not Applicable	The design does not use safety-related battery chargers or inverters.	Not Applicable
1.211	Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants	0	Conforms	None.	3.11
1.212	Sizing of Large Lead-Acid Storage Batteries	1	Not Applicable	This guidance is written in the context of a safety-related standby battery system and endorses IEEE Std. 485-2010. This is not applicable to the nonsafety-related normal DC power system and EDAS.	Not Applicable
1.213	Qualification of Safety-Related Motor Control Centers for Nuclear Power Plants	0	Not Applicable	The 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 applicants and licensees.	Not Applicable
1.216	Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design-Basis Pressure	0	Conforms	None.	3.8.2 6.2
1.217	Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts	0	Conforms	None.	19.5
1.218	Condition-Monitoring Techniques for Electric Cables Used in Nuclear Power Plants	0	Not Applicable	The applicant 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 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	1	Not Applicable	This guidance is applicable to operating reactor licensees, including applicants.	Not Applicable

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.221	Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants	0	Conforms	None.	2.0 3.3 3.5 3.8.4
1.226	Flexible Mitigation Strategies for Beyond-Design-Basis Events	0	Not Applicable	This guidance describes acceptable methods of complying with the requirements of 10 CFR 50.155, which is applicable to applicants and licensees.	Not Applicable
1.227	Wide-Range Spent Fuel Pool Level Instrumentation	0	Not Applicable	This guidance interprets implementation of 10 CFR 50.155, which is the responsibility of the applicant or licensee.	Not Applicable
1.230	Regulatory Guidance on the Alternate Pressurized Thermal Shock Rule	0	Not Applicable	This guidance applies to each holder of an operating license for a pressurized-water nuclear power reactor whose construction permit was issued before February 3, 2010, and whose RPV was designed and fabricated to the requirements of the ASME Boiler and Pressure Vessel Code, 1998 edition or earlier.	Not Applicable
1.231	Acceptance of Commercial-Grade Design and Analysis Computer Programs Used in Safety-Related Applications for Nuclear Power Plants	0	Conforms	None.	3.5
1.232	Developing Principal Design Criteria for Non-Light Water Reactors	0	Not Applicable	This guidance is applicable to non-LWRs. The NuScale design is an LWR.	Not Applicable
1.233	Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors	0	Not Applicable	This guidance is applicable to non-LWRs. The NuScale design is an LWR.	Not Applicable
1.234	Evaluating Deviations and Reporting Defects and Noncompliance under 10 CFR Part 21	0	Conforms	This guidance provides licensees and applicants with an acceptable method of evaluating and reporting defects under 10 CFR Part 21.	17.5

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.236	Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents	0	Conforms	NuScale Topical Report TR-0716-50353-P Rev. 2, states the methodology for evaluating rod ejection accidents uses the acceptance criteria and guidance of RG 1.236. This topical report methodology is used to evaluate the NPM rod ejection accident.	15.4.8
1.237	Guidance for Changes During Construction for New Nuclear Power Plants Being Constructed Under a Combined License Referencing a Certified Design Under 10 CFR Part 52	0	Not Applicable	This guidance governs site-specific activites that are the responsibility of the applicant or licensee referencing a certified design.	Not Applicable
1.239	Licensee Actions to Address Nonconservative Technical Specifications	0	Not Applicable	This guidance implements process requirements that are the responsibility of the applicant or licensee.	Not Applicable
1.240	Fresh and Spent Fuel Pool Criticality Analyses	0	Not Applicable	Spent fuel rack design and safety analysis is the responsibility of the applicant.	9.1.1 9.1.2
1.243	Safety-Related Steel Structures and Steel-Plate Composite Walls for Other than Reactor Vessels and Containments	0	Partially Conforms	This guidance is applicable to load and resistance factor design of safety-related steel structures and steel-plate composite walls. This guidance endorses ANSI.AISC N690-18 with modifications. As it relates to design, fabrication and erection of steel-plate composite walls, the NuScale design conforms with the load combinations for L and Ro provided in RG 1.243 with the exception of the live load factor in load combination NB2-4 and the accident pressure load factor in load combination NB2-8. A live load factor of 0.8 is used for load combination NB2-4, as defined in AISC N690-18, because it aligns with the principle used in recent design codes for developing load combinations. The load factor for the accident pressure in load combination NB2-8 is kept at 1.2 as accident pressures for NPP are calculated considering an extreme range of conditions, which results in low uncertainty in the calculated pressure values. The load combination NB2-5 is not applicable as NuScale defines operating basis earthquake (OBE) as one-third of safe shutdown earthquake (SSE).	3.8

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

RG	Title	Rev.	Conformance Status	Comments	Section
1.244	Control of Heavy Loads at Nuclear Facilities	0	Partially Conforms	This guidance endorses, with clarifications, the following ASME standards for control of heavy loads: • NML-1-2019, "Rules for the Movement of Loads Using Overhead Handling Equipment in Nuclear Facilities" • NOG-1-2020, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)" with exceptions listed in Section 9.1.5 • Chapters 1-3 of BTH-1-2017, "Design of Below-the-Hook Lifting Devices"	9.1.5
1.245	Preparing Probabilistic Fracture Mechanics Submittals	0	Not Applicable	The design does not use probabilistic fracture mechanics.	Not Applicable
1.247	Trial - Acceptability of Probabilistic Risk Assessment Results for Advanced Non-Light Water Reactor Risk-Informed Activities	Trial	Not Applicable	This guidance is applicable to non-LWRs. The NuScale design is an LWR.	Not Applicable
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 applicant or licensee.	Not Applicable
4.2	Preparation of Environmental Reports for Nuclear Power Stations	3	Not Applicable	This guidance governs site-specific environmental evaluation activities that are the responsibility of a license or construction permit applicant.	Not Applicable
4.2S1	Supplement 1 to RG 4.2, Preparation of Supplemental Environmental Reports for Applications to Renew Nuclear Power Plant Operating Licenses	1	Not Applicable	This guidance is applicable to licensees seeking renewal of their operating license.	Not Applicable
4.7	General Site Suitability Criteria for Nuclear Power Stations	3	Not Applicable	This guidance governs site-specific evaluation activities that are the responsibility of the 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

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

RG	Title	Rev.	Conformance Status	Comments	Section
4.13	Environmental Dosimetry - Performance Specification, Testing, and Data Analysis.	2	Not Applicable	This guidance applies to licensees and describes an approach to meet regulatory requirements for performing surveys and evaluations of public dose in the unrestricted area and the controlled area of a licensed facility from direct radiation using environmental dosimetry.	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	This guidance is applicable to 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
4.18	Standard Format and Content of Environmental Reports for Near-Surface Disposal of Radioactive Waste	0	Not Applicable	This guidance is applicable only to near-surface disposal of radioactive waste, which is not within the scope of the design.	Not Applicable
4.19	Guidance for Selecting Sites for Near-Surface Disposal of Low-Level Radioactive Waste	0	Not Applicable	This guidance is applicable only to near-surface disposal of radioactive waste, which is not within the scope of the 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

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

RG	Title	Rev.	Conformance Status	Comments	Section
4.21	Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning	0	Partially Conforms	This guidance is applicable except for site-specific, operational aspects that are the responsibility of the applicant or licensee.	9.1 9.4 10.4 11.2 11.3 12.3 12.5
4.22	Decommissioning Planning During Operations	0	Not Applicable	This guidance is applicable to operating reactor licensees.	Not Applicable
4.24	Aquatic Environmental Studies for Nuclear Power Stations	0	Not Applicable	This guidance governs site-specific environmental evaluation activities for freshwater, estuarine, and marine environments that are the responsibility of a license or construction permit applicant.	Not Applicable
4.25	Assessment of Abnormal Radionuclide Discharges in Ground Water to the Unrestricted Area at Nuclear Power Plant Sites	0	Not Applicable	This guidance governs assessments that are the responsibility of the applicant or licensee.	Not Applicable
4.26	Volcanic Hazards Assessment for Proposed Nuclear Power Reactor Sites	0	Not Applicable	This guidance governs assessments that are the responsibility of the applicant.	Not Applicable
5.4	Standard Analytical Methods for the Measurement of Uranium Tetrafluoride (UF4) and Uranium Hexafluoride (UF6)	0	Not Applicable	This guidance is applicable to 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	0	Not Applicable	This guidance is applicable to licensees of fuel fabrication facilities.	Not Applicable
5.7	Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas	1	Partially Conforms	This guidance is applicable except for site-specific, programmatic aspects that are the responsibility of the applicant or licensee.	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	This guidance is applicable to 10 CFR Part 70 facilities.	Not Applicable

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

RG	Title	Rev.	Conformance Status	Comments	Section
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 design.	Not Applicable
5.11	Nondestructive Assay of Special Nuclear Material Contained in Scrap and Waste	1	Not Applicable	This guidance 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 Control of: Facilities, Radioactive Materials, Classified Information, Classified Matter, and Safeguards Information	1	Partially Conforms	This guidance is applicable except for site-specific, programmatic aspects that are the responsibility of the applicant or licensee.	13.6 (via Security Technical Report)
5.20	Training, Equipping, and Qualifying of Guards and Watchmen	0	Not Applicable	This guidance governs site-specific activities that are the responsibility of the applicant or licensee.	Not Applicable
5.21	Nondestructive Uranium-235 Enrichment Assay by Gamma Ray Spectrometry	1	Not Applicable	This guidance is applicable to 10 CFR Part 70 facilities.	Not Applicable
5.23	In Situ Assay of Plutonium Residual Holdup	1	Not Applicable	This guidance is applicable to 10 CFR Part 70 facilities.	Not Applicable
5.25	Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Equipment for Wet Process Operations	0	Not Applicable	This guidance is applicable to 10 CFR Part 70 facilities.	Not Applicable
5.26	Selection of Material Balance Areas and Item Control Areas	1	Not Applicable	This guidance is applicable to 10 CFR Part 70 facilities.	Not Applicable
5.27	Special Nuclear Material Doorway Monitors	1	Not Applicable	This guidance is applicable to an applicant.	Not Applicable
5.29	Nuclear Material Control systems for Nuclear Power Plants	2	Not Applicable	This guidance is not applicable to the design but may be used by an 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	This guidance is applicable to an applicant.	Not Applicable

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

RG	Title	Rev.	Conformance Status	Comments	Section
5.34	Nondestructive Assay for Plutonium in Scrap Material by Spontaneous Fission Detection	1	Not Applicable	This guidance is applicable to 10 CFR Part 70 processing.	Not Applicable
5.37	In Situ Assay of Enriched Uranium Residual Holdup	1	Not Applicable	This guidance is applicable to 10 CFR Part 70 processing.	Not Applicable
5.38	Nondestructive Assay of High-Enrichment Uranium Fuel Plates by Gamma Ray Spectrometry	1	Not Applicable	This guidance is applicable to 10 CFR Part 70 processing.	Not Applicable
5.39	General Methods for the Analysis of Uranyl Nitrate Solutions for Assay, Isotopic Distribution, and Impurity Determinations	0	Not Applicable	This guidance is applicable to an applicant for a special nuclear material.	Not Applicable
5.41	Shipping, Receiving, and Internal Transfer of Special Nuclear Material At Fuel Cycle Facilities	0	Not Applicable	This guidance is applicable to 10 CFR Part 70 facilities.	Not Applicable
5.42	Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Equipment for Dry Process Operations	0	Not Applicable	This guidance is applicable to 10 CFR Part 70 facilities.	Not Applicable
5.44	Perimeter Intrusion Alarm Systems	3	Partially Conforms	This guidance is applicable except for site-specific, programmatic aspects that are the responsibility of the applicant or licensee.	13.6 (via Security Technical Report)
5.48	Design Considerations-Systems for Measuring the Mass of Liquids	0	Not Applicable	This guidance governs site-specific considerations that are the responsibility of the applicant or licensee.	Not Applicable
5.51	Management Review of Nuclear Material Control and Accounting Systems	1	Not Applicable	This guidance 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	This guidance is not applicable to nuclear power plants.	Not Applicable

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

RG	Title	Rev.	Conformance Status	Comments	Section
5.53	Qualification, Calibration, and Error Estimation Methods for Nondestructive Assay	1	Not Applicable	This guidance 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 applicant or licensee.	Not Applicable
5.55	Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities	0	Not Applicable	This guidance is applicable to fuel cycle facilities.	Not Applicable
5.58	Considerations for Establishing Traceability of Special Nuclear Material Accounting Measure-ments	1	Not Applicable	This guidance governs site-specific security activites concerning SNM and is the responsibility of an 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	This guidance governs site-specific security activites concerning SNM and is the responsibility of an applicant or licensee.	Not Applicable
5.60	Standard Format and Content of a Licensee Physical Protection Plan for Strategic Special Nuclear Material in Transit	0	Not Applicable	This guidance governs site-specific security activites concerning SNM and is the responsibility of an applicant or licensee.	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 an applicant or licensee.	Not Applicable
5.63	Physical Protection for Transient Shipments	0	Not Applicable	This guidance governs site-specific security activites and is the responsibility of an applicant or licensee.	Not Applicable
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 appli-cant or licensee.	Not Applicable
5.69	Guidance for the Application of Radiological Sabotage Design-Basis Threat in the Design, Development and Implementa-tion of a Physical Security Pro-gram that Meets 10 CFR 73.55 Requirements (SGI)	1	Not Applicable	This guidance governs site-specific security activites concerning SNM and is the responsibility of an applicant or licensee.	Not Applicable

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

RG	Title	Rev.	Conformance Status	Comments	Section
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)	1	Not Applicable	This guidance governs site-specific security activites and is the responsibility of an applicant or licensee.	Not Applicable
5.71	Cyber Security Programs for Nuclear Facilities	0	Partially Conforms	This guidance is applicable except for site-specific operational and programmatic activities (e.g., development and implementation of operational cyber security plans) that are the responsibility of the applicant or licensee.	13.6 (via Security Technical Report)
5.73	Fatigue Management for Nuclear Power Plant Personnel	0	Not Applicable	This guidance is not applicable to the design but may be used by an applicant or licensee to meet the fatigue management requirements of 10 CFR 26 Subpart I.	Not Applicable
5.74	Managing the Safety/Security Interface	1	Not Applicable	This guidance governs site-specific activities that are the responsibility of an applicant or licensee.	Not Applicable
5.75	Training and Qualification of Security Personnel at Nuclear Power Reactor Facilities	1	Not Applicable	This guidance governs site-specific activities that are the responsibility of an applicant or licensee.	Not Applicable
5.76	Physical Protection Programs at Nuclear Power Reactors (SGI)	1	Not Applicable	This guidance governs site-specific physical protection program activities that are the responsibility of the applicant or licensee.	Not Applicable
5.77	Insider Mitigation Program (SGI)	0	Not Applicable	This guidance governs site-specific activities that are the responsibility of an applicant or licensee.	Not Applicable
5.78	Physical Protection of Mixed Oxide Fuels in Nuclear Power Plants (SGI)	0	Not Applicable	The design does not use mixed oxide fuels.	Not Applicable
5.79	Protection of Safeguard Information	0	Conforms	Safeguards Information is protected against unauthorized disclosure in accordance with RG 5.79. Site-specific guidance is the responsibility of the applicant or licensee.	Not Applicable
5.80	Pressure-Sensitive and Tamper-Indicating Device Seals for Material Control and Accounting of Special Nuclear Material	0	Not Applicable	This guidance is not applicable to the NuScale design.	Not Applicable

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

RG	Title	Rev.	Conformance Status	Comments	Section
5.81	Target Set Identification and Development for Nuclear Power Reactors (OUO-SRI)	1	Not Applicable	This guidance governs site-specific activities that are the responsibility of an applicant or licensee.	Not Applicable
5.83	Cyber Security Event Notifications	0	Not Applicable	This guidance governs site-specific activities that are the responsibility of an applicant or licensee.	Not Applicable
5.84	Fitness-For-Duty for New Nuclear Power Plant Construction Sites	0	Not Applicable	This guidance governs site-specific activities that are the responsibility of an applicant or licensee.	Not Applicable
5.88	Physical Inventories and Material Balances at Fuel Cycle Facilities	0	Not Applicable	This guidance is applicable to 10 CFR Part 70 facilities.	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 applicant or licensee.	Not Applicable
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 applicant or licensee.	Not Applicable
8.7	Instructions for Recording and Reporting Occupational Radiation Dose Data	4	Not Applicable	This guidance governs site-specific programmatic activities related to recording and reporting dose data that are the responsibility of the 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 an applicant. However, the NuScale considered this guidance to be applicable to the extent necessary to provide reasonable assurance that an applicant referencing the NuScale US460 standard 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.	9.3 9.4 10.4 11.2 11.4 11.5 12.1 12.3 12.5
8.9	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program	1	Not Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the applicant.	Not Applicable
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable	2	Not Applicable	This guidance governs site-specific activities that are the responsibility of an applicant or licensee.	Not Applicable

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

RG	Title	Rev.	Conformance Status	Comments	Section
8.11	Applications of Bioassay for Uranium	1	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 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 applicant.	Not Applicable
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 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 applicant referencing the design. Construction activities dose assessments are the responsibility of the 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 applicant or licensee.	Not Applicable
8.21	Health Physics Surveys for Byproduct Material at NRC Licensed Processing and Manufacturing Plants	1	Not Applicable	This guidance is applicable to processing and manufacturing plants.	Not Applicable
8.22	Bioassay at Uranium Mills	2	Not Applicable	This guidance is applicable to uranium mills.	Not Applicable
8.24	Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication	2	Not Applicable	This guidance governs activities applicable 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 applicant or licensee.	Not Applicable
8.26	Applications of Bioassay for Fission and Activation Products	0	Not Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the applicant or licensee.	Not Applicable

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

RG	Title	Rev.	Conformance Status	Comments	Section
8.27	Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants	0	Not Applicable	This guidance governs site-specific operational training programs, plans, and procedures that are the responsibility of the applicant or licensee.	Not Applicable
8.28	Audible-Alarm Dosimeters	0	Not Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the 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 applicant or licensee.	Not Applicable
8.30	Health Physics Surveys in Uranium Recovery Facilities	1	Not Applicable	This guidance governs activities applicable 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 to uranium recovery facilities.	Not Applicable
8.32	Criteria for Establishing a Tritium Bioassay Program	0	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	0	Not Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the 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 applicant or licensee.	Not Applicable
8.36	Radiation Dose to the Embryo/Fetus	0	Not Applicable	This guidance governs site-specific, programmatic activities, procedures, equipment, and methods that are the responsibility of the applicant or licensee.	Not Applicable
8.37	ALARA Levels for Effluents from Materials Facilities	0	Not Applicable	This guidance governs activities applicable to materials facilities.	Not Applicable

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

RG	Title	Rev.	Conformance Status	Comments	Section
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 applicant. However, this guidance was considered to be applicable to the extent necessary to provide reasonable assurance that the applicant or licensee referencing the approved design can meet these requirements.	12.1 12.3 12.5 14.2
8.39	Release of Patients Administered Radioactive Materials	1	Not Applicable	This guidance governs activities applicable to facilities that administer radio-pharmaceuticals.	Not Applicable
8.40	Methods for Measuring Effective Dose Equivalent from External Exposure	0	Not Applicable	This guidance governs dosimetry methods for determining effective dose equivalent for external radiation exposures. These methods are the responsibility of the applicant or licensee.	Not Applicable

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

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	Not Applicable	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, Rev 1: 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, Rev 1: Site Characteristics and Site Parameters	II.2	COL Application Referencing an Early Site Permit but not a Certified Design	Not Applicable	This acceptance criterion is for COL applicants referencing an ESP.	2.0
SRP 2.0, Rev 1: Site Characteristics and Site Parameters	II.3	COL Application Referencing a Certified Design but not an Early Site Permit	Not Applicable	This acceptance criterion is for COL applicants that reference a certified design.	Not Applicable
SRP 2.0, Rev 1: 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 that are referencing both an ESP and a certified design.	Not Applicable
SRP 2.0, Rev 1: 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 to COL applicants that do not reference either an ESP or a certified design.	Not Applicable
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
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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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 2.3.2, Rev 3: Local Meteorology	All	Various	Not Applicable	Site-specific.	Not Applicable
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
SRP 2.3.5, Rev 3: Long-Term Atmospheric Dispersion Estimates for Routine Releases	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 2.4.1, Rev 3: Hydrologic Description	All	Various	Not Applicable	Site-specific.	2.4
SRP 2.4.2, Rev 4: Floods	All	Various	Not Applicable	Site-specific.	2.4
SRP 2.4.3, Rev 4: Probable Maximum Flood (PMF) on Streams and Rivers	All	Various	Not Applicable	Site-specific.	2.4
SRP 2.4.4, Rev 3: Potential Dam Failures	All	Various	Not Applicable	Site-specific.	2.4
SRP 2.4.5, Rev 3: Probable Maximum Surge and Seiche Flooding	All	Various	Not Applicable	Site-specific.	2.4
SRP 2.4.6, Rev 3: Probable Maximum Tsunami Hazards	All	Various	Not Applicable	Site-specific.	2.4
SRP 2.4.7, Rev 3: Ice Effects	All	Various	Not Applicable	Site-specific.	2.4
SRP 2.4.8, Rev 3: Cooling Water Canals and Reservoirs	All	Various	Not Applicable	The NuScale US460 standard design does not rely on safety-related cooling water or canals.	2.4

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 2.4.9, Rev 3: Channel Diversions	All	Various	Not Applicable	Site-specific.	2.4
SRP 2.4.10, Rev 3: Flooding Protection Requirements	All	Various	Not Applicable	There are no flood protection requirements.	2.4
SRP 2.4.11, Rev 3: Low Water Considerations	All	Various	Not Applicable	Low water considerations are not applicable to the NuScale US460 standard design.	2.4
SRP 2.4.12, Rev 3: Groundwater	All	Various	Not Applicable	Site-specific.	2.4
SRP 2.4.13, Rev 3: Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	All	Various	Not Applicable	Site-specific.	2.4
SRP 2.4.14, Rev 3: Technical Specifications and Emergency Operation Requirements	All	Various	Not Applicable	The NuScale US460 standard design does not rely on protective measure to minimize the impact of adverse hydrology-related events.	2.4
SRP 2.5.1, Rev 5: Geologic Characterization Information	All	Regional and Site Geology	Not Applicable	Site-specific.	2.5
SRP 2.5.2, Rev 5: Vibratory Ground Motion	All	Various	Not Applicable	Site-specific.	2.5
SRP 2.5.3, Rev 6: Surface Deformation	All	Various	Not Applicable	Site-specific.	2.5
SRP 2.5.4, Rev 5: Stability of Subsurface Materials and Foundations	All	Various	Not Applicable	Site-specific.	2.5
SRP 2.5.5, Rev 5: Stability of Slopes	All	Various	Not Applicable	Site-specific.	Not Applicable
SRP 3.2.1, Rev 3: 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.2 of RG 1.29 are designated Seismic Category II rather than Seismic Category I.	3.2.1
SRP 3.2.2, Rev 3: System Quality Group Classification	II.1	Quality Group Classification to Meet GDC 1 and 10 CFR 50.55a	Conforms	None.	3.2.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 3.2.2, Rev 3: System Quality Group Classification	Table 3.2.2-1	Summary of Construction Codes and Standards for Components of Water-Cooled Nuclear Power Plants by NRC Quality Classification System	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 3.2-2
SRP 3.2.2, Rev 3: 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 design does not include emergency diesel generators, ESF rooms, or pressurizer power operated relief valves.	Table 3.2-2
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
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	All	Various	Conforms	None.	3.4.1

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 3.4.2, Rev. 3: Analysis Procedures	All	Various	Conforms	None.	3.4.2
SRP 3.5.1.1, Rev. 3: Internally-Generated Missiles (Outside Containment)	All	Various	Conforms	None.	3.5.1
SRP 3.5.1.2, Rev. 3: Internally Generated Missiles (Inside Containment)	All	Various	Conforms	None.	3.5.1
DSRS 3.5.1.3, Rev. 0: Turbine Missiles	All except II.6	Various	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 SSC 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
SRP 3.5.1.4, Rev. 4: Missiles Generated by Extreme Winds	All	Various	Conforms	None.	3.5.1
SRP 3.5.1.5, Rev 4: Site Proximity Missiles (Except Aircraft)	All	Various	Not Applicable	The design assumes no proximity missiles.	Not Applicable
SRP 3.5.1.6, Rev 4: Aircraft Hazards	All	Various	Not Applicable	The design assumes no aircraft hazard missiles.	Not Applicable
SRP 3.5.2, Rev 3: Structures, Systems, and Components to be Protected From Externally-Generated Missiles	II	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	Partially Conforms	A finite element analysis for predicting penetration distance of turbine missiles in concrete is used, rather than the modified National Defense Research Council formula specified in Section II.1.A.	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 and Design Specific Review Standard (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	Partially Conforms	“Design of Composite SC Walls to Prevent Perforation from Missile Impact,“ Bruhl et al., 2015, is used for determining residual velocity.	3.5
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	All	Various	Conforms	None.	3.6.1
SRP 3.6.2, Rev 3: Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	All	Various	Conforms	None.	3.6.2
SRP 3.6.3, Rev 1: Leak-Before-Break Evaluation Procedures	All	Various	Not Applicable	None.	3.6.3
DSRS 3.7.1, Rev. 0: Seismic Design Parameters	All	Various	Conforms	None.	3.7.1
DSRS 3.7.2, Rev 0: Seismic System Analysis	All except II.12	Various	Conforms	TR-121515-P, US460 NuScale Power Module Seismic Analysis	3.7.1 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
DSRS 3.7.3, Rev. 0: Seismic Subsystem Analysis	All except II.13	Various	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 design does not use dams.	Not Applicable
SRP 3.7.4, Rev 3: Seismic Instrumentation	II.1	Comparison with RG 1.12	Partially Conforms	Locations are identified in conformance with RG 1.12; however, seismic instrumentation cannot be placed inside containment.	3.7.4

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 3.7.4, Rev 3: Seismic Instrumentation	II.2	Comparison with RG 1.166	Not Applicable	RG 1.166 in Table 1.9-2.	Not Applicable
SRP 3.7.4, Rev 3: 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 design does not have a concrete containment.	Not Applicable
DSRS 3.8.2, Rev. 0: Steel Containment	All	Various	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 containment does not have internal structures.	Not Applicable
DSRS 3.8.4, Rev. 0, Other Seismic Category I Structures	All except II.8	Various	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	All	Various	Conforms	None.	3.8.5
SRP 3.9.1, Rev 4: Special Topics for Mechanical Components	All except II.3	Various	Conforms	None.	3.9.1
SRP 3.9.1, Rev 4: 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
SRP 3.9.2, Rev 4: 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 an applicant referencing the NuScale US460 standard design.	3.9.2
SRP 3.9.2, Rev 4: Dynamic Testing and Analysis of Systems, Structures, and Components	All except II.1	Various	Conforms	None.	3.9.2
SRP 3.9.3, Rev 3: ASME Code Class 1, 2, and 3 Components and Component Supports, and Core Support Structures	II.1	Loading Combinations, System Operating Transients, and Stress Limits	Conforms	None.	3.9.3

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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 RPV, with a surge line connecting the two. In the NuScale design, the pressurizer is integral (i.e., is located within) to the RPV: 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 4: Control Rod Drive Systems	All	Various	Conforms	None.	3.9.4
SRP 3.9.5, Rev 4: Reactor Pressure Vessel Internals	All except II.5	Various	Conforms	None.	3.9.3 3.9.5
SRP 3.9.5, Rev 4: 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 different LWR and SSC conditions not relevant to the 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 CNV 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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 3.9.6, Rev 4: 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 design.	3.9.6
SRP 3.9.6, Rev 4: Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	II.2	Inservice Testing Program	Conforms	None.	3.9.6
SRP 3.9.6, Rev 4: Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	II.3	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 design. The 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 4: Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	II.4	Inservice Testing Program for Valves	Partially Conforms	Section 3.9.6.3.2 describes valve testing and Section 3.9.6.5 describes augmented valves testing program.	3.9.6
SRP 3.9.6, Rev 4: Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	II.5	Inservice Testing Program for Dynamic Restraints	Not Applicable	The NPP 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 4: Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	II.6	Relief Requests and Proposed Alternatives	Conforms	Section 3.9.6.5 contains relief requests and alternative authorizations to the code.	3.9.6

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 3.9.6, Rev 4: Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	II.7	ITAAC	Conforms	None.	3.9.6
SRP 3.9.6, Rev 4: Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	II.8	COL Action Items	Conforms	None.	3.9.6
SRP 3.9.6, Rev 4: Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	II.9	Operational Programs	Not Applicable	This acceptance criterion is related to operational activities, including implementation of preservice testing, inservice testing and inspection, and motor-operated valve testing programs, that are the responsibility of an applicant referencing the approved design.	Not Applicable
SRP 3.9.6, Rev 4: Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	II.10	Tier2* Information	Conforms	None.	3.9.6
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 applicants that reference the NuScale US460 standard 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 applicants that reference the NuScale US460 standard design, and that elect to implement such a program.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 3.10, Rev 4: 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 4: Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	II.2	Testing of Instrumentation Described in RG 1.97	Partially Conforms	RG 1.97 in Table 1.9-2	3.10 3.11
SRP 3.10, Rev 4: 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 4: Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	II.4	Records	Conforms	A Records program is required.	3.10
SRP 3.10, Rev 4: 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 4: Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	II.6	Documentation of Qualification Program	Conforms	None.	3.10
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	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. RG 1.63 entry in Table 1.9-2 provides the other aspects of RG 1.63.	3.11

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.4	Application of RG 1.73 for Environmental Design and Qualification of Class 1E Electric Valve Operators	Conforms	None.	3.11
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	RG 1.89 in Table 1.9-2.	3.11
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	RG 1.97 in Table 1.9-2.	3.11
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
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
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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.12	Application of RG 1.158 for Environmental Qualification of Class 1E Lead Storage Batteries	Not Applicable	RG 1.158 in Table 1.9-2. Lead acid storage batteries requiring qualification, use IEEE Std. 535-2013 instead.	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	RG 1.180 in Table 1.9-2.	3.11
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	RG 1.183 in Table 1.9-2.	3.11
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	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 environmental qualification because it is associated with an external natural event.	Not Applicable
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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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 applicant.	Not Applicable
DSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.23	Operational Program Implementation	Not Applicable	This is an 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
DSRS 3.11, Rev. 0: Environmental Qualification of Mechanical and Electrical Equipment	II.25	Design and Procurement Specifications	Not Applicable	This is an 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
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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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
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
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
SRP 3.13, Rev. 0: Threaded Fasteners - ASME Code Class 1, 2, and 3	II.2	Preservice and Inservice Inspection Requirements	Conforms	None.	3.13
BTP 3-1, Rev 2: Classification of Main Steam Components Other Than the Reactor Coolant Pressure Boundary for BWR Plants	All	Classification Quality Groups	Not Applicable	This guidance is applicable only to BWR plants.	Not Applicable
BTP 3-2, Rev 2: Classification of BWR/6 Main Steam and Feedwater Components Other Than the Reactor Coolant Pressure Boundary	All	Classification Quality Groups	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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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 3: Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment	B.A	High-Energy Fluid System Piping	Partially Conforms	The guidance for break exclusion is extended both inside and outside the CNV.	3.6 15.1 15.2 15.5 15.6
BTP 3-4, Rev 3: 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 3: 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
SRP 4.2, Rev 3: Fuel System Design	II.1	Design Bases	Conforms	None.	4.2
SRP 4.2, Rev 3: Fuel System Design	II.1.A	Fuel System Damage	Conforms	None.	4.2
SRP 4.2, Rev 3: Fuel System Design	II.1.B	Fuel Rod Failure	Conforms	None.	4.2
SRP 4.2, Rev 3: Fuel System Design	II.1.C	Fuel Coolability	Conforms	None.	4.2
SRP 4.2, Rev 3: Fuel System Design	II.2	Description and Design Drawings	Conforms	None.	4.2
SRP 4.2, Rev 3: Fuel System Design	II.3	Design Evaluation	Conforms	None.	4.2
SRP 4.2, Rev 3: Fuel System Design	II.4	Testing, Inspection, and Surveillance Plans	Conforms	None.	4.2
SRP 4.2, Rev 3: Fuel System Design	App A	Evaluation of Fuel Assembly Structural Response to Externally Applied Forces	Conforms	None.	4.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 4.2, Rev 3: Fuel System Design	App B	Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents	Not Applicable	SRP 4.2, Rev 3, Appendix B has been superseded by RG 1.236, Rev 0.	4.2 15.0.0
SRP 4.3, Rev 3: Nuclear Design	II.1	Design Limits for Power Densities and Power Distributions	Conforms	None.	4.3
SRP 4.3, Rev 3: Nuclear Design	II.2	Reactivity Coefficients	Conforms	None.	4.3
SRP 4.3, Rev 3: Nuclear Design	II.3	Control Rod Patterns and Reactivity Worth	Conforms	None.	4.3
SRP 4.3, Rev 3: Nuclear Design	II.4	Analytical Methods and Data	Conforms	None.	4.3
DSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.1	Fuel Design Limits, Core Design, and Thermal Margin	Conforms	None.	4.4
DSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.2	Subchannel Hydraulic Analysis Codes	Conforms	None.	4.4
DSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.3	Core Oscillations and Thermal-Hydraulic Instabilities	Conforms	None.	4.4 15.9
DSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.4	RPV Fluid Flow Calculations	Conforms	None.	4.4
DSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.5	Technical Specifications	Conforms	None.	4.4 16.1
DSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.6	Preoperational and Initial Test Programs	Conforms	None.	4.4
DSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.7	Loose Parts Monitoring System	Not Applicable	Low flow in primary systems precludes damage from loose parts and the need for loose parts monitoring system.	4.4
DSRS 4.4, Rev 0: Thermal and Hydraulic Design	II.8	Critical Heat Flux Calculations and Process Monitoring	Conforms	None.	4.4
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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.10	Core Stability Performance During Anticipated Transient without Scram Event	Not Applicable	Diverse reactor trip system (RTS) signals prevent an anticipated transient without scram (ATWS) from occurring. This prevents flow instabilities from occurring, so this acceptance criteria is not applicable based on the ATWS approach. The design supports an exemption from 10 CFR 50.62.	15.9
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
SRP 4.5.1, Rev 3: Control Rod Drive Structural Materials	II.2	Austenitic Stainless Steel Components	Conforms	The NuScale Quality Assurance Program Description (QAPD) is based on ANSI/ ASME NQA-1-2008 with NQA-1a-2009 addenda, as endorsed by RG 1.28, Rev. 4.	4.5
SRP 4.5.1, Rev 3: Control Rod Drive Structural Materials	II.3	Other Materials	Conforms	None.	4.5
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
SRP 4.5.2, Rev 3: Reactor Internal and Core Support Structure Materials	All	Various	Conforms	None.	4.5
SRP 4.6, Rev 2: Functional Design of Control Rod Drive System	II.1	Environmental and Dynamic Effects - GDC 4	Conforms	None.	4.6
SRP 4.6, Rev 2: Functional Design of Control Rod Drive System	II.2	Failure Modes and Effects - GDC 23	Conforms	None.	4.6
SRP 4.6, Rev 2: Functional Design of Control Rod Drive System	II.3	Single Malfunction - GDC 25	Conforms	None.	4.6

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.4	Operational Control and Reliability - GDC 26	Conforms	None.	4.6
SRP 4.6, Rev 2: Functional Design of Control Rod Drive System	II.5	Combined Capability - GDC 27	Conforms	None.	3.1 4.2 4.3 4.6
SRP 4.6, Rev 2: Functional Design of Control Rod Drive System	II.6	Reactivity Limits - GDC 28	Conforms	None.	4.6
SRP 4.6, Rev 2: Functional Design of Control Rod Drive System	II.7	Protection Against Anticipated Operational Occurrences (AOOs) - GDC 29	Conforms	None.	4.6
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	Various	Not Applicable	This BTP is applicable only to PWR designs that use the CAOC operating scheme. The CAOC operating scheme is not used.	Not Applicable
SRP 5.2.1.1, Rev 4: 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	RG 1.26 in Table 1.9-2.	5.2
SRP 5.2.1.2, Rev 4: Applicable Code Cases	II.1	Use of RG 1.84 to meet GDC 1 and 10 CFR 50.55a	Conforms	RG 1.84 in Table 1.9-2.	5.2
SRP 5.2.1.2, Rev 4: Applicable Code Cases	II.2	Use of RG 1.147 to meet GDC 1 and 10 CFR 50.55a	Partially Conforms	RG 1.147 in Table 1.9-2.	5.2
SRP 5.2.1.2, Rev 4: Applicable Code Cases	II.3	Use of RG 1.192 to meet GDC 1 and 10 CFR 50.55a	Partially Conforms	RG 1.192 in Table 1.9-2.	5.2
SRP 5.2.2, Rev 3: Overpressure Protection	II.1	Material Specifications	Conforms	None.	5.2
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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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 reactor protection system initiates the reactor trip. The design does not have a secondary safety-grade RTS.	5.2
SRP 5.2.2, Rev 3: Overpressure Protection	II.4	Design Requirements for PWRs Operating at Low Temperature (Startup, Shutdown)	Conforms	None.	5.2
SRP 5.2.2, Rev 3: Overpressure Protection	II.5	Testing and Inspections	Conforms	None.	5.2
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 partially applicable.	5.2
SRP 5.2.2, Rev 3: Overpressure Protection	II.7	TMI Action Plan Requirements	Conforms	None.	5.2
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
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
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.3.A	Fracture Toughness - 10 CFR 50, Appendix G	Partially Conforms	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix G, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel”	5.2
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.3.B	Control of Ferritic Steel Welding	Conforms	None.	5.2
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.3.C	Non-Destructive Examination of Ferritic Steel Tubular Products	Conforms	None.	5.2

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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
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. The NuScale design is based on NQA-1-2015.	5.2
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 RCPB components. Nonmetallic thermal insulation is not used on RCPB 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
SRP 5.2.3, Rev 3: Reactor Coolant Pressure Boundary Materials	II.4.E	Non-Destructive Examination of Austenitic Stainless Steel Tubular Products	Conforms	None.	5.2
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 applicant.	Not Applicable
DSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.1 - II.10	Various	Conforms	None.	5.2
DSRS 5.2.4, Rev 0: Reactor Coolant Pressure Boundary Inservice Inspection and Testing	II.11	Other Inspection Programs	Partially Conforms	A brief description of the boric acid control program is provided.	5.2
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 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 to applicants.	5.2

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.14	Risk Informed ISI Program	Not Applicable	This acceptance criterion governs plant-specific programmatic information that is the responsibility of the applicant.	Not Applicable
DSRS 5.2.5, Rev 0: Reactor Coolant Pressure Boundary Leakage Detection	All	Various	Conforms	None.	5.2
DSRS 5.3.1, Rev 0: Reactor Vessel Materials	II.1 - II.4	Various	Conforms	None.	5.3
DSRS 5.3.1, Rev 0: Reactor Vessel Materials	II.5	Fracture Toughness	Partially Conforms	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix H, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel.”	5.3
DSRS 5.3.1, Rev 0: Reactor Vessel Materials	II.6	Material Surveillance	Not Applicable	The design supports an exemption from 10 CFR 50.61 and from 10 CFR 50.60 and thus from 10 CFR 50, Appendix G, due to austenitic stainless steel in the reactor pressure vessel beltline.	5.3
DSRS 5.3.1, Rev 0: Reactor Vessel Materials	II.7	Reactor Vessel Fasteners	Conforms	None.	5.3
DSRS 5.3.1, Rev 0: Reactor Vessel Materials	II.8	10 CFR 52.47(b)(1) compliance	Partially Conforms	10 CFR 52.47(b)(1) applies to design certification applications. The SDA includes proposed ITAAC.	Not Applicable
DSRS 5.3.2, Rev 0: Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	II.1 - II.3	Various	Partially Conforms	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix G, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel."	5.3.1

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.1.B	Pressure-Temperature Requirements	Partially Conforms	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix G, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel."	5.3
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	Not Applicable	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix G, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel."	5.3
DSRS 5.3.2, Rev 0: Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	II.2.B	Upper-Shelf Energy Requirements	Not Applicable	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix G, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel."	5.3
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	Not Applicable	The design supports an exemption from 10 CFR 50.61 due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel."	5.3
DSRS 5.3.2, Rev 0: Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	II.3.B	Pressurized Thermal Shock Requirements	Not Applicable	The design supports an exemption from 10 CFR 50.61 due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel."	5.3
DSRS 5.3.2, Rev 0: Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	II.4	10 CFR 52.47(b)(1) compliance	Partially Conforms	10 CFR 52.47(b)(1) applies to design certification applications. The SDA includes proposed ITAAC.	Not Applicable

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.1	Design	Conforms	None.	5.3
DSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.2	Materials of Construction	Conforms	None.	5.3
DSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.3	Fabrication Methods	Conforms	None.	5.3
DSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.4	Inspection Requirements	Conforms	None.	5.3
DSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.5	Shipment and Installation	Conforms	None.	5.3
DSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.6	Operating Conditions	Conforms	None.	5.3
DSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.7	Inservice Surveillance	Partially Conforms	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix H, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," provides justification.	5.3
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 applicant.	Not Applicable
DSRS 5.3.3, Rev 0: Reactor Vessel Integrity	II.9	10 CFR 52.47(b)(1) compliance	Partially Conforms	10 CFR 52.47(b)(1) applies to design certification applications. The SDA includes proposed ITAAC.	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 design does not require or include reactor coolant pumps. The NuScale design uses passive natural circulation of the primary coolant.	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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 5.4.2.1, Rev 0: Steam Generator Materials	II.2	Steam Generator Design	Conforms	None.	5.4
DSRS 5.4.2.1, Rev 0: Steam Generator Materials	II.3	Fabrication and Processing of Ferritic Materials	Conforms	None.	5.4
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
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
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
DSRS 5.4.2.1, Rev 0: Steam Generator Materials	II.7	10 CFR 52.47(b)(1) compliance	Not Applicable	10 CFR 52.137 Contents of applications; technical information (under Subpart E - Standard Design Approvals) is substituted for 10 CFR 52.47 Contents of applications; technical information (under Subpart B - Standard Design Certifications). Paragraph (b)(1) is not part of 10 CFR 52.137.	Not Applicable
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 applicants referencing a NuScale US460 standard design.	5.4
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 partially applicable.	5.4
DSRS 5.4.2.2, Rev 0: Steam Generator Program	II.4	Steam Generator Tube Repair Criteria	Conforms	None.	5.4
DSRS 5.4.2.2, Rev 0: Steam Generator Program	II.5	Steam Generator Tube Repair Methods	Conforms	None.	5.4
DSRS 5.4.2.2, Rev 0: Steam Generator Program	II.6	Steam Generator Tube Preservice Inspection	Conforms	None.	5.4

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.7	Periodic Tube Inspection and Testing in Certified Design Technical Specifications	Partially Conforms	None	5.4
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 an applicant referencing a NuScale US460 standard design.	5.4
DSRS 5.4.2.2, Rev 0: Steam Generator Program	II.9	ITAAC	Partially Conforms	A portion of this acceptance criterion is applicable to applicants.	5.4
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
DSRS 5.4.7, Rev 0: Decay Heat Removal (DHR) System Responsibilities	II.4	GDC 5	Conforms	None.	5.4
DSRS 5.4.7, Rev 0: Decay Heat Removal (DHR) System Responsibilities	II.5	GDC 14	Not Applicable	The decay heat removal system (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	Partially Conforms	The 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
DSRS 5.4.7, Rev 0: Decay Heat Removal (DHR) System Responsibilities	II.7	GDC 34	Partially Conforms	The 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 PDC in lieu of this GDC.	5.4

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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 are 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
DSRS 5.4.7, Rev 0: Decay Heat Removal (DHR) System Responsibilities	II.9	DHRS Interface with other systems	Conforms	None.	5.4
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 RCS overpressure protection system is routed directly to the CNV.	Not Applicable

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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	Not Applicable	Because of the integral RCS configuration, non-condensable gases accumulating in the pressurizer space do not interfere with core cooling during or after DBAs. The design supports an exemption from the requirements of 10 CFR 50.46a related to RCS high point venting, as well as the substantively similar requirements of 10 CFR 50.34(f)(2)(vi).	Not Applicable
SRP 5.4.13, Rev. 0: 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 10.3
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 an applicant referencing the NuScale US460 standard design.	5.4 10.3
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 an applicant referencing the NuScale US460 standard design.	5.4 10.3
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 an applicant referencing the NuScale US460 standard 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	Partially Conforms	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix G, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel."	5.2

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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. The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix G, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel."	5.2
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
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
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 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
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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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 system should not depend on the availability of offsite power to perform its function - applies to the design.	5.2
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
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 design, the low temperature overpressure protection system is not connected to a low-pressure system. However, the intent of this guidance - to ensure that the low temperature overpressure protection system is not inadvertently isolated from the primary system - is applicable.	5.2
BTP 5-3, Rev 3: Fracture Toughness Requirements	1	Preservice Fracture Toughness Test Requirements	Partially Conforms	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix G, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," provides justification.	5.3

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
BTP 5-3, Rev 3: Fracture Toughness Requirements	1.1	Determination of RT_{NDT} for Vessel Materials	Not Applicable	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix G, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," provides justification.	5.3
BTP 5-3, Rev 3: Fracture Toughness Requirements	1.2	Estimation of Charpy V-Notch Upper Shelf Energies	Not Applicable	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix G, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," provides justification.	5.3
BTP 5-3, Rev 3: Fracture Toughness Requirements	1.3	Reporting Requirements	Partially Conforms	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix G, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," provides justification.	5.3
BTP 5-3, Rev 3: Fracture Toughness Requirements	2	Operating Limitations for Fracture Toughness	Partially Conforms	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix G, due to austenitic stainless steel in the reactor pressure vessel beltline. The Pressure-Temperature Operating Limitations are calculated but are not in accordance with 10 CFR 50, Appendix G. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," provides justification.	5.3

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
BTP 5-3, Rev 3: Fracture Toughness Requirements	2.1	Pressure-Temperature Operating Limitations	Partially Conforms	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix G, due to austenitic stainless steel in the reactor pressure vessel beltline. The Pressure-Temperature Operating Limitations are calculated but are not in accordance with 10 CFR 50, Appendix G. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," provides justification.	5.3
BTP 5-3, Rev 3: Fracture Toughness Requirements	2.2	Recommended Bases for Operating Limitations	Partially Conforms	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix H, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," provides justification.	5.3
BTP 5-3, Rev 3: Fracture Toughness Requirements	2.3	Reporting Requirements	Conforms	None.	5.3
BTP 5-3, Rev 3: Fracture Toughness Requirements	3	Inservice Surveillance of Fracture Toughness	Not Applicable	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix H, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," provides justification.	5.3
BTP 5-3, Rev 3: Fracture Toughness Requirements	3.1	Surveillance Program Requirements	Not Applicable	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix H, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," provides justification.	5.3

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
BTP 5-3, Rev 3: Fracture Toughness Requirements	3.2	SAR Criteria	Not Applicable	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix H, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," provides justification.	5.3
BTP 5-3, Rev 3: Fracture Toughness Requirements	3.3	Surveillance Test Procedures	Not Applicable	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix H, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," provides justification.	5.3
BTP 5-3, Rev 3: Fracture Toughness Requirements	3.4	Reporting Criteria	Not Applicable	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix H, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," provides justification.	Not Applicable
BTP 5-3, Rev 3: Fracture Toughness Requirements	3.5	Technical Specification Changes	Not Applicable	The design supports an exemption from 10 CFR 50.60 and thus from 10 CFR 50, Appendix H, due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," provides justification.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
BTP 5-3, Rev 3: Fracture Toughness Requirements	4.1	Pressurized Thermal Shock Requirements	Partially Conforms	The design supports an exemption from 10 CFR 50.61 due to austenitic stainless steel in the reactor pressure vessel beltline. TR-130721, Revision 0, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," provides justification.	5.3
BTP 5-3, Rev 3	4.2	Alternative Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events	Not Applicable	This acceptance criterion is only applicable to construction permits for PWRs issued prior to February 3, 2010.	Not Applicable
DSRS BTP 5-4, Rev 0: Design Requirements of the Decay Heat Removal (DHR) System Responsibilities	B.1	Functional Requirements	Conforms	None.	5.4
DSRS BTP 5-4, Rev 0: Design Requirements of the Decay Heat Removal (DHR) System Responsibilities	B.2	Pressure Relief Requirements	Conforms	None.	5.4
DSRS BTP 5-4, Rev 0: Design Requirements of the Decay Heat Removal (DHR) System Responsibilities	B.3	Test Requirements	Conforms	None.	5.4
DSRS BTP 5-4, Rev 0: Design Requirements of the Decay Heat Removal (DHR) System Responsibilities	B.4	Operational Procedures	Partially Conforms	The procedures governed by this acceptance criterion are site-specific and are the responsibility of the applicant.	5.4
DSRS BTP 5-4, Rev 0: Design Requirements of the Decay Heat Removal (DHR) System Responsibilities	B.5	Implementation	Conforms	None.	5.4
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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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 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 was 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 NPM design does not use protective coatings inside the CNV.	Not Applicable
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
DSRS 6.2.1.1.A, Rev 0: Containment	II.2	Reducing Containment Pressure Following Postulated Design Basis Accident	Conforms	The peak post-accident pressure is reduced to less than 50 percent of the peak calculated pressure for the design-basis LOCA within 24 hours of the postulated accident.	6.2.1
DSRS 6.2.1.1.A, Rev 0: Containment	II.3	Containment Heat Removal Capability and Design Margin - LOCA Assumptions	Conforms	Postulated single failures are considered by the containment response analysis methodology for postulated secondary system pipe ruptures.	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	Design specific analyses demonstrate the capability of the containment to withstand the maximum expected external pressure.	6.2.1

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.6	Containment Monitoring Instrumentation	Conforms	Instrumentation is provided to monitor containment parameters for normal operation, anticipated operational occurrences and accidents.	6.2.1 7.1
DSRS 6.2.1.1.A, Rev 0: Containment	II.7	Design of Containment Internal Structures and System Components	Conforms	The design satisfies GDC 4. The containment is designed to accommodate the environmental conditions resulting from external events.	6.2.1
DSRS 6.2.1.1.A, Rev 0: Containment	II.8	Evaluation of Accident Involving Generated Hydrogen	Not Applicable	The containment atmosphere is inert, therefore analysis of the effects of hydrogen combustion on containment integrity is not necessary.	6.2.5
DSRS 6.2.1.1.A, Rev 0: Containment	II.9	Evaluation of an Accident on other Modules	Conforms	None.	3.1.1 9.2.5
SRP 6.2.1.1.B, Draft Rev 3: Ice Condenser Containments	All	Various	Not Applicable	The 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 CNV 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
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	Partially Conforms	The energy from metal-water reactions is not included. The design supports an exemption from selected portions of 10 CFR 50, Appendix K.	6.2.1

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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 to 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 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
DSRS 6.2.2, Rev 0: Containment Heat Removal Systems	II.2	GDC 38, Containment Heat Removal	Partially Conforms	The 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 PDC 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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.4	GDC 40, Testing of Containment Heat Removal System	Not Applicable	The design supports an exemption from GDC 40.	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 CNV 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
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 containment isolation valves.	Not Applicable

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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
DSRS 6.2.4, Rev 0: Containment Isolation System	II.13	Radiation Monitors for Initiation of Containment Isolation on Open Paths to the Environs	Partially Conforms	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 containment evacuation system to the environs, consistent with the intent of this acceptance criterion. The 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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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
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 containment atmosphere is inert, therefore the design safely accommodates hydrogen generated by an equivalent of a 100% fuel clad-coolant reaction without limiting containment hydrogen concentration to less than 10% by volume.	6.2.5
DSRS 6.2.5, Rev 0: Combustible Gas Control in Containment	II.2	Equipment Survivability and Containment Structural Integrity	Partially Conforms	The design satisfies 10 CFR 50.44(c)(3) by maintaining an inert atmosphere, therefore the environmental conditions created by hydrogen combustion are not considered.	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	Partially Conforms	The 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 PDC in lieu of this GDC.	6.2.5
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	Partially Conforms	The design includes a PAR. The test and inspection of containment components are addressed in FSAR Section 6.2.5.	6.2.5

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.6	Containment Structural Integrity Analysis	Partially Conforms	The design includes a PAR that maintains an inert containment atmosphere and precludes hydrogen combustion. A beyond-design-basis containment structural evaluation considers an amount of hydrogen exceeding that generated by 100 percent fuel clad-coolant reaction; the containment remains below design pressure.	6.2.5
DSRS 6.2.6, Rev 0: Containment Leakage Testing	All	Various	Partially Conforms	The 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	Partially Conforms	None.	6.3.2
DSRS 6.3, Rev 0: Emergency Core Cooling System	II.4	Combined Reactivity Control System Capability and Actuation Provisions	Conforms	The ECCS supplemental boron feature supplies sufficient boron addition after ECCS actuation to ensure subcriticality.	6.3.1
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 15.0.5

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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 DBA. The CRHS neither relies on nor uses emergency filtration to protect operators during accident conditions. Rather, clean air is provided using compressed air tanks.	Not Applicable
SRP 6.4, Rev 3: Control Room Habitability System	II.5	Relative Location of Source and Control Room	Not Applicable	This guidance is applicable to reactor designs that rely on the control room emergency ventilation system for control room habitability during a DBA. The CRHS 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	Radiation Hazards	Partially Conforms	Acceptance criterion 6.A is applicable only to currently operating reactors. Acceptance criterion 6.B is applicable and requires the applicant to meet the requirements of GDC 19. The plant design supports an exemption from GDC 19; however, PDC 19 includes the control room dose requirements of this acceptance criterion.	6.4

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.7	Toxic Gas Hazards	Not Applicable	Compliance with RG 1.78 is the responsibility of the applicant.	Not Applicable
SRP 6.5.1, Rev 4: ESF Atmosphere Cleanup Systems	II	Design, Testing, and Maintenance of ESF Atmosphere Cleanup System Air Filtration and Adsorption Units	Not Applicable	The NPP design does not use ESF filter systems or ESF ventilation systems to mitigate the consequences of a DBA. In the NPP design there is a nonsafety-related RXB 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 CNV design does not incorporate a spray system.	Not Applicable
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 CNV 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 NPM, wherein the compact CNV is submerged in the reactor pool. Therefore, the aspects of this guidance related to these systems are not applicable. This guidance is applicable to the review of certain containment parameters and design features, such as design leakage rate and systems leakage before containment isolation.	6.5.3

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.2	Secondary Containment	Not Applicable	This acceptance criterion is applicable only to LWRs that incorporate both a primary and secondary containment. The CNV 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 NPP design is the CNV in conjunction with the containment isolation valves and passive containment isolation barriers.	Not Applicable
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 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 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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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
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
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	The ASME BPVC Case N-849, “In Situ VT-3 Examination of Removable Core Support Structure Without Removal” meets the conditions in NRC Regulatory Guide 1.147.	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 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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
BTP 6-1, Rev. 0: pH for Emergency Coolant Water for Pressurized Water Reactors	B.1	Minimum pH for Emergency Coolant Water	Conforms	This acceptance criterion is applicable but language in BTP 6-1, which would be applied by Acceptance Criterion B.1, refers to SSC that are not in the design. Specifically, the design does not use a containment spray system or a sump. However, during ECCS operation, ECCS water does collect inside the CNV for recirculation back to the reactor core, and thus the intent of this acceptance criterion is applicable.	6.1.1
BTP 6-1, Rev. 0: 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 design. Specifically, the design does not use a containment spray system or a sump. However, during ECCS operation, ECCS water does collect inside the CNV for recirculation back to the reactor core, and thus the pH guideline contained in this acceptance criterion is applicable.	6.2.2
BTP 6-1, Rev. 0: pH for Emergency Coolant Water for Pressurized Water Reactors	B.3	Hydrogen Generation from Aluminum Corrosion	Conforms	This acceptance criterion is applicable but language in BTP 6-1, which would be applied by Acceptance Criterion B.3, refers to SSC that are not in the design. Specifically, the design does not use a containment spray system or a sump. However, during ECCS operation, ECCS water does collect inside the CNV for recirculation back to the reactor core, and thus the intent of this acceptance criterion is applicable.	6.3.2
BTP 6-2, Rev 3: Minimum Containment Pressure Model for PWR ECCS Performance Evaluation	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 design, a LOCA does not result in core uncovery.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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 CNV design does not include a secondary containment.	Not Applicable
BTP 6-4, Rev 3: Containment Purging During Normal Plant Operations	All (B.1 thru B.5)	Various	Not Applicable	This guidance pertains to containment purge systems used to vent containment directly to the environs. While the CNV 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 (DSRS 6.2.4, AC II.13). The CNV 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 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. 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
DSRS Appendix 7.0-B, Rev 0: I&C - System Architecture	-	I&C - System Architecture	Conforms	None.	7.1 7.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS Appendix 7.0-C, Rev 0: I&C - Simplicity	-	I&C - Simplicity	Conforms	None.	7.1
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
DSRS 7.1.2, Rev 0: Independence	II.1	Ensure compliance to current version of RG 1.75	Conforms	None.	7.1
DSRS 7.1.2, Rev 0: Independence	II.2	Ensure compliance to current version of RG 1.152	Conforms	None.	7.1
DSRS 7.1.3, Rev 0: Redundancy	All	Conformance with RG 1.53	Conforms	None.	7.1
DSRS 7.1.4, Rev 0: Predictability and Repeatability	All	Predictability and Repeatability	Conforms	None.	7.1
DSRS 7.1.5, Rev 0: Diversity and Defense in Depth	II.1	Methods for performing diversity and defense in depth (D3) analyses of reactor protection systems	Conforms	The D3 assessment of the I&C design is consistent with the guidelines in NUREG/CR-6303.	7.1
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
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 7.1.1. The I&C systems are designed to the Quality Assurance Program described in Section 17.5.	7.1
DSRS 7.1.5, Rev 0: Diversity and Defense in Depth	II.4	Conformance to RG 1.53	Conforms	None.	7.1
DSRS 7.1.5, Rev 0: Diversity and Defense in Depth	II.5	Conformance to RG 1.62	Conforms	RG 1.62 in Table 1.9-2.	7.2
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
DSRS 7.2.1 Rev. 0: Quality	II.1	Conformance to RG 1.28	Conforms	RG 1.28 in Table 1.9-2.	7.2
DSRS 7.2.1 Rev. 0: Quality	II.2	Conformance to RG 1.152	Conforms	RG 1.152 in Table 1.9-2.	7.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 7.2.1 Rev. 0: Quality	II.3	Conformance to RG 1.168	Partially Conforms	RG 1.168 in Table 1.9-2.	7.2
DSRS 7.2.1 Rev. 0: Quality	II.4	Conformance to RG 1.169	Partially Conforms	RG 1.169 in Table 1.9-2.	7.2
DSRS 7.2.1 Rev. 0: Quality	II.5	Conformance to RG 1.170	Partially Conforms	RG 1.170 in Table 1.9-2.	7.2
DSRS 7.2.1 Rev. 0: Quality	II.6	Conformance to RG 1.171	Partially Conforms	RG 1.171 in Table 1.9-2.	7.2
DSRS 7.2.1 Rev. 0: Quality	II.7	Conformance to RG 1.172	Partially Conforms	RG 1.172 in Table 1.9-2.	7.2
DSRS 7.2.1 Rev. 0: Quality	II.8	Conformance to RG 1.173	Partially Conforms	RG 1.173 in Table 1.9-2.	7.2
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
DSRS 7.2.2, Rev 0: Equipment Qualification	II.2	Conformance to RG 1.209	Conforms	None.	7.2
DSRS 7.2.2, Rev 0: Equipment Qualification	II.3	Conformance to RG 1.151	Partially Conforms	RG 1.151 in Table 1.9-2.	7.2
DSRS 7.2.2, Rev 0: Equipment Qualification	II.4	Conformance to RG 1.180	Partially Conforms	RG 1.180 in Table 1.9-2.	7.2
DSRS 7.2.2, Rev 0: Equipment Qualification	II.5	Conformance to RG 1.204	Partially Conforms	RG 1.204 in Table 1.9-2.	7.2
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
DSRS 7.2.4, Rev 0: Operating and Maintenance Bypasses	II.1	Conformance to RG 1.47	Conforms	None.	7.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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
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
DSRS 7.2.7, Rev 0: Setpoints	II.1	Conformance to RG 1.105	Conforms	RG 1.105 in Table 1.9-2.	7.2
DSRS 7.2.7, Rev 0: Setpoints	II.2	NRC Regulatory Issue Summary (RIS) 2006-17	Not Applicable	The setpoint methodology conforms to RG 1.105.	7.2
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-122844, Instrument Setpoint Methodology Technical Report (Reference 7.2-25).	7.2
DSRS 7.2.8, Rev 0: Auxiliary Features	All	Various	Conforms	There are no specific DSRS acceptance criteria in this section.	7.2
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
DSRS 7.2.9, Rev 0: Control of Access, Identification, and Repair	II.2	Conformance to RG 1.75	Conforms	None.	7.2
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
DSRS 7.2.11, Rev 0: Multi-Unit Stations	II.1	Conformance to RG 1.53	Conforms	None.	7.1
DSRS 7.2.12, Rev 0: Automatic and Manual Controls	II.1	Conformance to RG 1.62	Conforms	RG 1.62 in Table 1.9-2.	7.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 7.2.13, Rev 0: Displays and Monitoring	II.1	Conformance to RG 1.97	Partially Conforms	RG 1.97 in Table 1.9-2.	7.2
DSRS 7.2.13, Rev 0: Displays and Monitoring	II.2	Conformance to RG 1.47	Conforms	None.	7.2
DSRS 7.2.13, Rev 0: Displays and Monitoring	II.3	SECY-93-087	Conforms	The main control room and alternate operator workstations are designed to maintain alarm system reliability in accordance with item II.T of SECY-93-087.	7.2
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
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
DSRS 7.2.15, Rev 0: Capability for Test and Calibration	II.2	Conformance to RG 1.118	Partially Conforms	RG 1.118 in Table 1.9-2.	7.2
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 NRC requirements, guidance, and Commission policy documents, and industry codes and standards 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 partially relevant to the design.	8.1 8.2 8.3 8.4
DSRS 8.2, Rev 0: Offsite Power System	II.1	Compliance with GDC 5	Not Applicable	Not applicable to the offsite power system as described in Section 8.2.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 8.2, Rev 0: Offsite Power System	II.2	Compliance with GDC 17	Not Applicable	The design supports an exemption from GDC 17 that includes the associated requirements for the offsite power system.	8.2
DSRS 8.2, Rev 0: Offsite Power System	II.3	Compliance with GDC 18	Not Applicable	The design supports an exemption from GDC 18 that includes the associated requirements for the offsite power system.	8.2
DSRS 8.2, Rev 0: Offsite Power System	II.4	Compliance with GDC 33	Not Applicable	The design supports an exemption from GDC 33.	8.2
DSRS 8.2, Rev 0: Offsite Power System	II.4	Compliance with GDCs 34, 35, 38, 41, and 44	Partially Conforms	NuScale complies with a set of principal design criteria in lieu of these GDC.	8.2
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 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 an applicant referencing the US460 standard 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.	8.1 8.3
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.	8.1 8.3
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 8.3
DSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.4	Compliance with GDC 17	Not Applicable	The design supports an exemption from GDC 17 that includes the associated requirements for the onsite AC power system.	8.3
DSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.5	Compliance with GDC 18	Not Applicable	The design supports an exemption from GDC 18 that includes the associated requirements for the onsite AC power system.	8.1 8.3

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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 (No Number)	Compliance with GDC 33	Not Applicable	The design supports an exemption from GDC 33.	8.1 8.3
DSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II (No Number)	Compliance with GDCs 34, 35, 38, 41, and 44	Partially Conforms	NuScale complies with a set of principal design criteria in lieu of these GDC.	8.1 8.3
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 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 10 CFR 50.65 maintenance rule program, including the identification of SSC that require assessment per 10 CFR 50.65(a)(4), is the responsibility of an applicant referencing the NuScale US460 standard design.	Not Applicable
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)	Partially Conforms	10 CFR 52.47(b)(1) applies to design certification applications. The SDA application includes proposed ITAAC.	8.1 8.3 14.2
DSRS 8.3.1, Rev 0: AC Power Systems (Onsite)	II.10	Compliance with 10 CFR 52.80(a)	Not Applicable	This acceptance criterion is the responsibility of the applicant or licensee.	Not Applicable
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 2	Conforms	None.	8.3
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 4	Conforms	None.	8.3
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 5	Conforms	None.	8.3
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 17	Not Applicable	The design supports an exemption from GDC 17 that includes the associated requirements for the onsite DC power systems.	8.3

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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 (No Number)	Compliance with GDC 18	Not Applicable	The design supports an exemption from GDC 18 that includes the associated requirements for the onsite DC power systems.	8.3
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 33	Not Applicable	The design supports an exemption from GDC 33.	8.3
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II (No Number)	Compliance with GDC 34, 35, 38, 41, and 44	Partially Conforms	Nuscale complies with a set of PDC in lieu of these GDC.	8.3
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 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	Not Applicable	None.	8.2.2
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.2	Conformance with RG 1.75	Not Applicable	None.	8.3
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.3	Conformance with RG 1.81	Not Applicable	Compliance with GDC 5 is shown without use of the RG.	8.3
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.4	Conformance with RG 1.118	Not Applicable	None.	Not Applicable
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.5	Conformance with RG 1.153	Not Applicable	None.	Not Applicable
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.6	Conformance with RG 1.53	Not Applicable	None.	Not Applicable
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.7	Conformance with RG 1.63	Partially Conforms	RG 1.63 in Table 1.9-2.	8.1 8.3
DSRS 8.3.2, Rev 0: DC Power Systems (Onsite)	II.8	Conformance with RG 1.160	Not Applicable	Development of the 10 CFR 50.65 maintenance rule program including the identification of SSC that require assessment per 10 CFR 50.65(a)(4) is the responsibility of an applicant referencing the NuScale US460 standard design.	8.3
DSRS 8.4, Rev 0: Station Blackout	II.1	Compliance with 10 CFR 50.63 and the guidelines of RG 1.155	Partially Conforms	Section 8.4.	8.4

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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 a station blackout (SBO) event. As described in Section 19.3, a regulatory treatment of nonsafety systems (RTNSS) process has been implemented. Consequently, the Alternate AC Power Source is not applicable to the 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 (EDAS) is described in Section 8.3.	8.3 8.4
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 license 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 design does not use safety injection tanks (or equivalent) in response to a DBA. Design and operation of the ECCS also do not involve motor-operated valves.	Not Applicable
SRP BTP 8-2, Rev 3: Use of Diesel-Generator Sets for Peaking	B.	Use of Onsite Emergency Power Diesel-Generator Sets for Purposes Other Than Supplying Standby Power is Prohibited	Not Applicable	The design does not rely on AC power sources for the performance of safety-related functions; therefore, the guidance of BTP 8-2 need not be applied.	8.1
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 an applicant.	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 an applicant.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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 ESFs and are not relied on to support ESFs.	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 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 NPP does not require or include safety-related emergency diesel generators.	Not Applicable
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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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
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	The criticality analysis for new and spent fuel storage is the responsibility of an applicant that references the NuScale US460 standard design.	9.1.1 9.1.2
DSRS 9.1.2, Rev 0: New and Spent Fuel Storage	II.1	Specific Criteria to Meet GDC 2	Conforms	None.	9.1.2
DSRS 9.1.2, Rev 0: New and Spent Fuel Storage	II.2	Specific Criteria to Meet GDC 4	Conforms	None.	9.1.2
DSRS 9.1.2, Rev 0: New and Spent Fuel Storage	II.3	Specific Criteria to Meet GDC 5	Conforms	None.	9.1.2
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 (RG 1.52).	9.1.2
DSRS 9.1.2, Rev 0: New and Spent Fuel Storage	II.5	Specific Criteria to Meet GDC 63	Conforms	None.	9.1.2
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
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 and Design Specific Review Standard (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 large inventory of water within the Seismic Category I structures forming the UHS is the source for spent fuel cooling and shielding for accident conditions. A UHS makeup supply line is designed to Quality Group D and Seismic Category I requirements. (2) An ESF ventilation system is not required (RG 1.52).	9.1.3 9.2.5
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 (RG 1.52).	9.1.3 9.2.5
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 9.2.5
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 9.2.5
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
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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.	9.1.3
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 to 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 and Design Specific Review Standard (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 site cooling water system (SCWS) does not provide essential cooling to safety-related SSC and is not safety-related or augmented quality. The applicability of GDC 2 to the SCWS reviewed under this acceptance criterion is limited to aspects ensuring a failure of the nonsafety-related SCWS does not result in an adverse effect on a Seismic Category I SSC. For the 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.2.	9.2.7 (Used for SCWS)
SRP 9.2.1, Rev 5: Station Service Water System	II.2	Environmental and Dynamic Effects (GDC 4)	Partially Conforms	The SCWS 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 SCWS 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 SCWS)
SRP 9.2.1, Rev 5: Station Service Water System	II.3	Sharing of Structures, Systems, and Components (GDC 5)	Conforms	The SCWS does not provide essential cooling to safety-related SSC and is not safety-related or risk-significant. The design and layout of these systems satisfy GDC 5. Specifically, sharing of the SCWS among modules has no reasonable likelihood of adversely affecting essential SSC and associated safety functions.	9.2.7 (Used for SCWS)
SRP 9.2.1, Rev 5: Station Service Water System	II.4	Cooling Water System (GDC 44)	Not Applicable	The SCWS does not serve a safety-related cooling or accident mitigation function.	Not Applicable

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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 SCWS 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 SCWS 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	Conforms	This criterion is based on RG 1.29. The RCCWS is not classified as Seismic Category I. The CHWS complies with Staff Regulatory Guidance C.2 in that the SSC whose failure could adversely affect Seismic Category I SSC 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	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	Conforms	None.	9.2.2
SRP 9.2.2, Rev 4: Reactor Auxiliary Cooling Water System	II.4	Cooling Water System	Not Applicable	The RCCWS is a nonsafety-related system, and does not provide cooling to safety-related or risk-significant components that require the cooling to perform their safety function.	Not Applicable
SRP 9.2.2, Rev 4: Reactor Auxiliary Cooling Water System	II.5	Cooling Water System Inspection	Not Applicable	Addressed in comment above for Acceptance Criterion II.4.	Not Applicable
SRP 9.2.2, Rev 4: Reactor Auxiliary Cooling Water System	II.6	Cooling Water System Testing	Not Applicable	Addressed in comment above for Acceptance Criterion II.4.	Not Applicable
SRP 9.2.4, Rev 3: Potable and Sanitary Water Systems	II.1	Control of Releases of Radioactive Materials to the PWSW	Conforms	None.	9.2.4

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 9.2.5, Rev 3: Ultimate Heat Sink	II.1	Protection Against Natural Phenomena	Partially Conforms	RG 1.27 is not applicable to the design. Compliance with GDC 2 is demonstrated by adherence to RG 1.13, Regulatory Positions C.1 and C.2. The UHS provides both spent fuel cooling and containment heat removal, and is protected from natural phenomena and site-related events by the RXB structure and with a Seismic Category I emergency makeup line.	9.2.5 9.2.7
SRP 9.2.5, Rev 3: Ultimate Heat Sink	II.2	Sharing of Structures, Systems, and Components	Conforms	None.	9.2.5 9.2.7
SRP 9.2.5, Rev 3: Ultimate Heat Sink	II.3	Cooling Water System	Partially Conforms	The plant design supports an exemption from the power provisions of GDC 44. As described in Section 3.1.4, the design complies with a principal design criterion in lieu of this GDC. The aspects of this acceptance criterion that relate to the use of fiberglass piping (RG 1.72) do not apply to the UHS, because the 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 and Design Specific Review Standard (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 condensate storage facility is neither safety-related nor risk-significant. The condensate storage facility and components are located outside the RXB. 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.2 of RG 1.29, no portion of the condensate storage facility 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
SRP 9.2.6, Rev 3: Condensate Storage Facilities	II.2	Sharing of Structures, Systems, and Components	Conforms	None.	9.2.6
SRP 9.2.6, Rev 3: Condensate Storage Facilities	II.3	Condensate Storage Facility	Not Applicable	This acceptance criterion applies to condensate storage facilities that provide decay heat removal or essential cooling for safety-related equipment, neither of which is the case in the plant design.	Not Applicable
SRP 9.2.6, Rev 3: Condensate Storage Facilities	II.4	Condensate Storage Facility Inspection	Not Applicable	Addressed in comment above for Acceptance Criterion II.3.	Not Applicable
SRP 9.2.6, Rev 3: Condensate Storage Facilities	II.5	Condensate Storage Facility Testing	Not Applicable	Addressed in comment above for Acceptance Criterion II.3.	Not Applicable
SRP 9.2.6, Rev 3: Condensate Storage Facilities	II.6	Control of Radioactive Releases to the Environment	Conforms	None.	9.2.6
SRP 9.2.6, Rev 3: Condensate Storage Facilities	II.7	Loss of All Alternating Current Power	Conforms	The condensate storage facility meets the intent of this acceptance criterion by being neither safety-related nor risk-significant. The NPP passive design means that the condensate storage facility is not required to cool safety-related or risk-significant components.	9.2.6

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.1	Quality Standards and Records	Not Applicable	The chilled water system (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.2 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
SRP 9.2.7, Rev 0: Chilled Water System	II.4	Sharing of Structures, Systems, and Components	Not Applicable	The CHWS is a nonsafety-related 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 CHWS is a nonsafety-related 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 CHWS is a nonsafety-related 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 CHWS is a nonsafety-related 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 liquid radioactive waste system and gaseous radioactive waste system 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	Compressed air systems are nonsafety-related, 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	Compressed air systems are nonsafety-related, 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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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 an 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 NPP design. The 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 an SBO in the NuScale design.	9.3.1 8.4
SRP 9.3.2, Rev 3: Process and Post-Accident Sampling Systems	II.1	Sampling Capability	Partially Conforms	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 drain system penetrate the containment barrier.	9.3.3

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.1	CVCS Functional Performance during Adverse Environmental Phenomena; Pumping Capacity; and defense-in-depth RCS makeup	Partially Conforms	The CVCS is a nonsafety-related system; however, the CVCS includes safety-related demineralized water isolation valves whose safety function is to preclude inadvertent boron dilution of the RCS.	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 RCS.	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
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 design. The 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 CVCS design, which includes appropriate venting and draining capability.	9.3.4

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.8	ITAAC	Conforms	None.	9.3.4
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 9.3.7
DSRS 9.3.6, Rev 0: Containment Evacuation and Flooding Systems	II.2	GDC 60	Conforms	None.	9.3.6 9.3.7
DSRS 9.3.6, Rev 0: Containment Evacuation and Flooding Systems	II.3	TMI 10 CFR 50.34(f)	Not Applicable	The design supports an exemption from 10 CFR 50.34(f)(2)(xiv)(E)	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 normal control room ventilation system is part of normal plant operations. Up to six 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	Partially Conforms	This acceptance criterion is applicable except for aspects related to ESF atmosphere cleanup systems. The CRHS 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 and Design Specific Review Standard (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 an 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 NPP design. The 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 the safety benefit a typical large LWR gains by having an alternate AC power source (e.g., gas turbine generator) for SBO. Moreover, and specific to this SRP Section 9.4.1 acceptance criterion, the CRHS (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 (RG 1.52).	9.4.2

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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 Radioactive Waste Building HVAC system does not filter exhaust. Exhaust is filtered by the Reactor Building HVAC 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 Turbine Building HVAC system is not relied on to control airborne radioactivity concentrations in the Turbine Building and gaseous effluents during normal operations (including AOOs) and after accidents that result in a radioactive material release. Furthermore, there are no requirements for Turbine Building HVAC system 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 and Design Specific Review Standard (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 (DBA) environment. The design does not rely on ESF ventilation systems to mitigate the consequences of a DBA. 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 applicants that reference the 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 the current year subtier documents are used.	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 and Design Specific Review Standard (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 applicants under 10 CFR 52. Applicants referencing NuScale US460 standard design are responsible for implementing this guidance. Notwithstanding the above, NuScale, as an applicant for a standard design approval, considers this guidance to be applicable to the design approval application to the extent necessary to ensure the 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. Because of the modular nature and small size of the NPM, it is not feasible in all instances to provide installed passive separation of redundant trains. When train separation is not feasible, fire protection for redundant shutdown systems is employed to ensure, to the extent practicable, that one shutdown division remains free of fire damage.	9.5.1 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 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 applicants that reference the 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 applicant.	9.5.2

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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 plant design includes provisions for an onsite Technical Support Center as specified by 10 CFR 50.34(f)(2)(xxv) and this acceptance criterion. Communication systems serving this facility in support of emergency response are the responsibility of the applicant.	9.5.2
DSRS 9.5.2, Rev 0: Communication Systems	II.3	Emergency Facilities and Equipment for Meeting 10 CFR 50.47(b)(6) and 10 CFR 50.47(b)(8)	Partially Conforms	Plant design includes provisions for communications that are maintained in the emergency facilities and main control room to support emergency response. Communication systems and equipment serving these facilities in support of emergency response are the responsibility of the 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 plant design are applicable. 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 applicant.	Ch 14
DSRS 9.5.2, Rev 0: Communication Systems	II.6	ITAAC for a COL applicant	Not Applicable	Preparation of site-specific ITAAC is the responsibility of the applicant.	Not Applicable
DSRS 9.5.2, Rev 0: Communication Systems	II.7	Compliance with GDC 1	Conforms	None.	9.5.2
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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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	Partially Conforms	The plant design supports an exemption from GDC 19. As described in Section 3.1.2, the design complies with a NuScale-specific 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 main 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 the 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 plant design does not reprocess spent fuel or use or transport special nuclear material.	Not Applicable
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 applicant. Aspects of this acceptance criterion related to the physical design of the power reactor and communication systems are within the scope of the design.	13.6 (via Security Technical Report)

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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 applicant. Aspects of this acceptance criterion that are related to the physical design of the power reactor and communication systems within the scope of the design.	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 applicant and are addressed as part of the facility physical security plan.	13.6 (via Security Technical Report)
SRP 9.5.3, Rev 3: Lighting Systems	II.1	Degree of similarity with previously reviewed plants	Conforms	None.	9.5.3
SRP 9.5.3, Rev 3: Lighting Systems	II.2	Adequate normal lighting	Conforms	None.	9.5.3
SRP 9.5.3, Rev 3: Lighting Systems	II.3	Adequate emergency lighting	Conforms	None.	9.5.3
SRP 9.5.3, Rev 3: Lighting Systems	II.4	Conformance to NUREG-0700	Partially conforms	SRP 9.5.3, Rev 3 references NUREG-0700, Rev 2. The control room lighting design conforms to NUREG-0700, Rev 3, which includes specific guidance for the lighting of a computer-based control room.	9.5.3
SRP 9.5.4, Rev 3: Emergency Diesel Engine Fuel Oil Storage and Transfer System	All (II.1 thru II.4)	Compliance with GDC 2, GDC 4, GDC 5, and GDC 17	Not Applicable	The 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 NPP safety functions.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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 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 NPP 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 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 NPP 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 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 NPP 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 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 NPP safety functions.	Not Applicable
SRP 10.2, Rev 3: Turbine Generator	II.1	Protect important to safety SSC from the effects of turbine missiles with a turbine overspeed protection system (GDC 4)	Not Applicable	The design relies on the use of barriers for the protection of essential SSC from the effects of turbine missiles.	Not Applicable
SRP 10.2, Rev 3: Turbine Generator	II.2	Inservice Inspection covering valves essential for overspeed protection.	Not Applicable	The design relies on the use of barriers for the protection of essential SSC 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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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 SSC 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 SSC 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	Preservice 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 SSC 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 SSC 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	Partially Conforms	Per DSRS 10.2.3, Section I and DSRS 3.5.1.3, Section I.1, plants that use barriers to protect essential SSC specified in RG 1.115 do not have to rely on the turbine missile generation probabilities, including turbine rotor integrity. However, major system components are accessible for inspection.	10.2

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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	10 CFR 52.47(b)(1) applies to design certification applications. The SDA application includes proposed ITAAC. Per DSRS 10.2.3, Section I and DSRS 3.5.1.3, Section I.1, plants that use barriers to protect essential SSC 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 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.2 and C.3.	10.3
DSRS 10.3, Rev 0: Main Steam Supply System	II.2	Protection of important to safety SSC from the effects of turbine missiles (GDC 4)	Conforms	The MSS is not safety-related or risk-significant. Thus, the applicability of GDC 4 to the 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
DSRS 10.3, Rev 0: Main Steam Supply System	II.3	Shared important to safety SSC perform required safety functions (GDC 5)	Conforms	None.	10.3
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 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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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
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 design contains no Class 2 or 3 components.	Not Applicable
SRP 10.3.6, Rev 3: Steam and Feedwater System Materials	II.2	Fracture Toughness of Class 2 and 3 Components	Not Applicable	The 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
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
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
SRP 10.4.4, Rev 3: Turbine Bypass System	II.1	Piping Failures (GDC 4)	Conforms	None.	10.4
SRP 10.4.4, Rev 3: Turbine Bypass System	II.2	Residual Heat Removal (GDC 34)	Partially Conforms	The 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 PDC in lieu of this GDC.	10.4
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 important to safety SSC (GDC 4)	Not Applicable	None.	Not Applicable
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 RCPB (GDC 14)	Not Applicable	BWR only.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 10.4.6, Rev 3: Condensate Cleanup System	II.2	Maintain indirect cycle PWR water quality to avoid corrosion-induced failure of the RCPB (GDC 14)	Conforms	In the SG design, the primary water is outside the SG tubes, the secondary water is inside the tubes, and there is no SG blowdown so the secondary chemistry requirements for the design differ from those outlined in the referenced EPRI report.	10.4
DSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.1	Seismic Events (GDC 2)	Conforms	None.	10.4
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 NPP design.	10.4
DSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.3	Sharing of Structures, Systems, and Components (GDC 5)	Conforms	None.	10.4
DSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.4	Heat Removal Capability (GDC 44)	Not Applicable	The FWS is not a system used to transfer heat to a UHS.	Not Applicable
DSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.5	Inspection (GDC 45)	Not Applicable	The FWS is not a system used to transfer heat to a UHS.	Not Applicable
DSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.6	Testing (GDC 46)	Not Applicable	The FWS is not a system used to transfer heat to a UHS.	Not Applicable
DSRS 10.4.7, Rev 0: Condensate and Feedwater System	II.7	Flow Accelerated Corrosion	Conforms	None.	10.3 10.4
SRP 10.4.8, Rev 3: Steam Generator Blowdown System	All	Various	Not Applicable	The SG design does not use a blowdown system.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 10.4.9, Rev 3: Auxiliary Feedwater System (PWR)	All	Various	Not Applicable	The design neither requires nor uses an auxiliary feedwater system. The 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
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 system (AFWS) pumps powered by electrical and steam sources. The DHRS fulfills a similar function as the AFWS at a large PWR. The 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 design does not use a top-feed SG 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 design does not use a preheat SG 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 SG design. The NPP design does not involve an AFWS as would be found at a typical large LWR, but does include the DHRS that fulfills a similar function as a typical AFWS. However, the SG 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	Partially Conforms	The tests and test procedures are the responsibility of the applicant.	5.4

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
DSRS 11.1, Rev 0: Coolant Source Terms	II.1	RG 1.110	Partially Conforms	RG 1.110 in Table 1.9-2.	11.1 11.2 11.3
DSRS 11.1, Rev 0: Coolant Source Terms	II.2	RG 1.112	Partially Conforms	RG 1.112 in Table 1.9-2.	11.1 11.2 11.3
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 11.2 11.3
DSRS 11.1, Rev 0: Coolant Source Terms	II.4	DC/COL-ISG-5	Not Applicable	The 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 design.	Not Applicable
DSRS 11.1, Rev 0: Coolant Source Terms	II.5	Normal operation and anticipated operational occurrence (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 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 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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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	RG 1.110 in Table 1.9-2.	11.1 11.2 11.3
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
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 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 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 the design is partially based on a failed fuel fraction much less than 0.25 percent, which is described in 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.14	calculational technique or any source term parameter	Conforms	None.	11.1

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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 applicant.	11.2
DSRS 11.2, Rev 0: Liquid Waste Management System	II.2	Design for Anticipated Processing Requirements	Conforms	None.	11.2
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
DSRS 11.2, Rev 0: Liquid Waste Management System	II.4	Provisions to Control Leakage and Facilitate Operation and Maintenance	Conforms	None.	11.2
DSRS 11.2, Rev 0: Liquid Waste Management System	II.5	Automatic control features	Conforms	None.	11.2 11.5 11.6
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 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 applicant.	11.3
DSRS 11.3, Rev 0: Gaseous Waste Management System	II.2	Design for Anticipated Processing Requirements	Conforms	None.	11.3
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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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 applicant.	11.3
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
DSRS 11.3, Rev 0: Gaseous Waste Management System	II.6	Automatic control features	Conforms	None.	11.3
DSRS 11.3, Rev 0: Gaseous Waste Management System	II.7	Design to Withstand Effects of Hydrogen Explosion	Conforms	None.	11.3
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
DSRS 11.3, Rev 0: Gaseous Waste Management System	II.9	Criteria for Early Site Permit Applications	Not Applicable	This acceptance criterion is applicable 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.
DSRS 11.4, Rev 0: Solid Waste Management System	II.1	Design Parameters Based on Expected Radionuclide Distributions and Concentrations	Conforms	None.	11.4
DSRS 11.4, Rev 0: Solid Waste Management System	II.2	Sizing of Processing Equipment	Conforms	None.	11.4
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 applicant.	11.4
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 applicant.	Not Applicable

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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 Offsite Dose Calculation Manual (ODCM) are the responsibility of the 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 applicant.	11.4
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 applicant.	11.4
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
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 applicant.	11.4 12.3
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 applicant.	11.4 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 NPP design has no long term storage facility for solid radioactive waste. This is an 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 applicant. This guidance is applicable to the extent necessary to ensure that the applicant referencing the approved design can satisfy the guidance.	11.4

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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 applicant. This guidance is applicable to the extent necessary to ensure that the applicant referencing the approved design can satisfy the guidance.	11.4
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 applicant. This guidance is applicable to the extent necessary to ensure that the applicant referencing the certified design can satisfy the guidance contained therein.	11.4
DSRS 11.4, Rev 0: Solid Waste Management System	II.15	All Effluent Releases Associated with Operation of the solid waste management system	Partially Conforms	This acceptance criterion is applicable except for site-specific, programmatic aspects that are the responsibility of the applicant.	11.4
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 applicant.	Not Applicable
DSRS 11.4, Rev 0: Solid Waste Management System	II.17	Automatic control features	Not Applicable	None.	Not Applicable
DSRS 11.4, Rev 0: Solid Waste Management System	II.18	Design of exhaust ventilation systems	Conforms	None.	11.4
DSRS 11.4, Rev 0: Solid Waste Management System	II.19	Seismic design of structures housing solid waste management system	Conforms	None.	11.4
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 (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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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 (RG 1.21, RG 1.33, RG 1.97, RG 4.15, RG 4.21 and BTP 7-10). Administrative and procedural controls are applicant responsibility.	9.3.2 11.5 12.3
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 (RG 1.21, RG 1.33, RG 1.97 and RG 4.15). Administrative and procedural controls are 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 (RG 1.97 and BTP 7-10). Administrative and procedural controls are 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 applicant's responsibility.	11.5
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 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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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 an applicant's responsibility.	9.3.2 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.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 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 applicant responsibilities.	Not Applicable

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.8	Compliance with 10 CFR 20.1406 via RG 4.21, NEI 97-06, 08-08A and 07-07.	Partially Conforms	RG 4.21 in Table 1.9-2.	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.9	Description of design features and instrumentation used in primary and secondary coolant system leakage detection	Conforms	None.	5.2 9.3.4 9.3.6 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.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 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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.1	Processing Requirements	Conforms	This guidance is applicable except for aspects related to PCP development and implementation that are applicable to applicants.	11.4
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 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
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 applicants.	11.4
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 applicants.	11.4
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
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
BTP 11-6, Rev 4: Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	B.1	Failure Mechanism and Radioactivity Releases	Not Applicable	This acceptance criteria is responsibility of the applicant.	Not Applicable

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.2	Mitigating Design Features	Not Applicable	This acceptance criteria is responsibility of the 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 applicant.	11.2
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 the 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 the 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 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 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 applicant referencing the design.	12.1
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
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 applicant.	Not Applicable

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.4	Radiation Protection Considerations	Not Applicable	Addressed in comment above for Acceptance Criterion II.3.	Not Applicable
DSRS 12.2, Rev 0: Radiation Sources	II.1	RG 1.183	Partially Conforms	RG 1.183 in Table 1.9-2.	12.2
DSRS 12.2, Rev 0: Radiation Sources	II.2	RG 1.7	Not Applicable	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	RG 1.112 in Table 1.9-2.	12.2
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	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	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
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.2	RG 1.52	Not Applicable	RG 1.52 in Table 1.9-2.	Not Applicable

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.3	RG 1.69	Partially Conforms	RG 1.69 in Table 1.9-2.	12.3
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.4	RG 1.97	Partially Conforms	RG 1.97 in Table 1.9-2.	7.2 12.3
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.5	RG 1.183	Partially Conforms	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	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	RG 8.8 in Table 1.9-2.	12.3
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.8	RG 8.10	Not Applicable	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	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	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	RG 8.38 in Table 1.9-2.	12.3
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.13	ANSI/ANS/HPSSC-6.8.1-1981	Conforms	None.	12.3
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.14	ANSI/HPS N13.1-2011	Conforms	None.	12.3

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.15	ANSI/ANS-6.4-2006	Conforms	None.	12.3
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.	12.3
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.17	RG 1.140	Partially Conforms	RG 1.140 in Table 1.9-2.	12.3
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.18	RG 1.89	Partially Conforms	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	RG 4.21 in Table 1.9-2.	12.3
DSRS 12.3-12.4, Radiation Protection Design Features	II.20	RG 1.45	Partially Conforms	RG 1.45 in Table 1.9-2.	5.2
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	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	This guidance is applicable except for aspects related to PCP development and implementation that are applicable to applicants.	11.4
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.24	RG 1.97	Partially Conforms	RG 1.97 in Table 1.9-2.	12.3
DSRS 12.3-12.4, Rev 0: Radiation Protection Design Features	II.25	RG 1.12	Partially Conforms	RG 1.12 in Table 1.9-2.	12.3

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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 applicant referencing the design.	Not Applicable
SRP 13.1.1, Rev 6: Management and Technical Support Organization	All	General and Specific Requirements	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.1.2 - 13.1.3, Rev 7: Operating Organization	All	Operating Organization	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.2.1, Rev 4: Reactor Operator Requalification Program; Reactor Operator Training	All	General and Specific Requirements	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.2.2, Rev 4: Non-Licensed Plant Staff Training	All	Various	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	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	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.2	Onsite and Offsite Emergency Response Plans	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.3	Emergency Classification and Action Level Scheme	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.4	Meteorological Criteria	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	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	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.7	Protective Action Recommendations	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.8	Alternatives to NUREG-0654/ FEMA-REP-1, Rev 1,	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 13.3, Rev 3: Emergency Planning	II.9	State, Tribal, and Local Government Planning and Preparedness	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.10	Emergency Planning Zones	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.11	Evacuation Time Estimates	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.12	Emergency Response Data System	Partially Conforms	The design includes an Emergency Response data system. Site-specific aspects are the responsibility of the applicant.	13.3
SRP 13.3, Rev 3: Emergency Planning	II.13	Acceptability of Emergency Plans	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.14	Offsite Emergency Planning When Local Governments Decline to Participate	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	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 SDAA.	Not Applicable

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.24	Generic Emergency Planning ITAAC	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.25	Design and Implementation of Emergency Response Facilities	Partially Conforms	The design includes a Technical Support Center. The Emergency Operations Facility is the responsibility of the applicant that references the 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 applicant that references the NuScale design.	13.3
SRP 13.3, Rev 3: Emergency Planning	II.27	Reactor Coolant System and Containment Sampling	Not Applicable	The 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 applicant.	9.3.2 11.5
SRP 13.3, Rev 3: Emergency Planning	II.29	NRC Notifications and Communications	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.30	Generic Communications and Commission Orders Pertaining to Emergency Planning	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.31	Operational Programs	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.4, Rev 3: Operational Programs	Not Applicable	Various	Not Applicable	There are no specific requirements for this SRP section.	Not Applicable
SRP 13.5.1.1, Rev 2: Administrative Procedures - General	All	Various	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.5.2.1, Rev 2: Operating and Emergency Operating Procedures	All	Various	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 13.6, Rev 3R, Physical Security	N/A	Introduction	Not Applicable	SRP Section 13.6 assigns a separate section of the SRP to each type of licensing action (i.e., SRP Sections 13.6.1 through 13.6.6).	Not Applicable
SRP 13.6.1, Rev 2: Physical Security - Combined License and Operating Reactors	All	Various	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.6.2, Rev 2: Physical Security - Review of Physical Security System Designs - Standard Design Certification and Operating Reactor Licensing Applications	All	Various	Conforms	Applicable for the physical security elements within the boundary of the NPP.	13.6 (via Security Technical Report)
SRP 13.6.3, Rev 2: Physical Security - Early Site Permit and Reactor Siting Criteria	All	Various	Not Applicable	Applicable for an ESP applicant.	Not Applicable
SRP 13.6.4, Rev 0: Access Authorization Operational Program	II	10 CFR 73.56	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.6.6, Rev 0: Cyber Security Plan	All	Various	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.7.1, Rev 0: Fitness for Duty - Operational Program	All	Various	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	Not Applicable
SRP 13.7.2, Rev 0: Fitness for Duty - Construction	All	Various	Not Applicable	This acceptance criteria is applicable to the applicant or licensee.	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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.3	Initial Test Program Administrative Procedures	Partially Conforms	Subheading DC Applicant, Items A through D, are applicable to the SDAA. Subheading COL/OL applicants, Items A through H, are applicable to an applicant.	14.2
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 SDAA. Subheading COL/OL applicants, Items A and B, are applicable to 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 SDAA. Subheading COL/OL applicants, Items A through C, are applicable to applicants.	14.2
SRP 14.2.1, Rev. 0: 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, Rev. 0: Inspections, Tests, Analyses, and Acceptance Criteria	II.1	Acceptability of the Scope of ITAAC	Conforms	None.	14.3
SRP 14.3, Rev. 0: 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, Rev 1: 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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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
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, Rev. 0: 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, Rev. 0: 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, Rev. 0: 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 applicant addresses Physical Security Hardware ITAAC outside of the nuclear island and structures.	13.6 14.3

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.1	Categorization of Transients and Accidents	Conforms	None.	15.0
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 infrequent event (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. Critical heat flux (CHF), not departure from nucleate boiling ratio (DNBR), is used to determine the thermal margin for the fuel cladding. The LOCA acceptance criteria are 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 applicants.	15.0
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. CHF is evaluated, which is more applicable to the design than DNBR.	15.0.2

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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 design uses a modified version of the alternative source term 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 design utilizes a modified version of the alternative source term methodology to evaluate radiological consequences of accidents.	Not Applicable
SRP 15.0.2, Rev 0: Review of Transient and Accident Analysis Methods	II.1	Evaluation Model	Partially Conforms	The design supports an exemption from selected portions of 10 CFR 50 Appendix K. ECCS evaluation models for LOCAs only address technically relevant features required by Appendix K.	15.0.2
SRP 15.0.2, Rev 0: Review of Transient and Accident Analysis Methods	II.2	Accident Scenario Identification Process	Conforms	None.	15.0.2
SRP 15.0.2, Rev 0: Review of Transient and Accident Analysis Methods	II.3	Code Assessment	Partially Conforms	The 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, Rev 0: 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, Rev 0: Review of Transient and Accident Analysis Methods	II.5	Quality Assurance Plan	Conforms	None.	15.0.2
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 15.10

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.2	Control Room Radiological Habitability	Conforms	None.	6.4 9.4 13.3 15.0.3 15.10
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 are 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 15.10
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
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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.2 Specific Criterion	Minimum Critical Heat Flux Ratio Remains Above 95/95 Limit	Conforms	NuScale determined that CHF more accurately describes plant phenomena than departure from nucleate boiling (DNB).	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	Events are classified by AOO, IE, accident, and special event, but conform with the 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
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

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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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 determined that CHF more accurately describes plant phenomena than DNB.	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
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 determined that CHF more accurately describes plant phenomena than DNB.	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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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.	15.0.3
SRP 15.1.5.A, Rev 2: Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR	II (No number)	First full paragraph and Items 1 and 2 on Page 15.1.5-11, Exposure Guidelines for Calculated Doses	Partially Conforms	The part of this guidance specifying the calculation of radiological consequences of a postulated main steam line break outside containment is applicable. 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

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

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

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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.D	Values of Parameters Used in Analytical Model - Instrumentation Setpoints for Mitigating System Actuation	Conforms	None.	15.2.1-15.2.5
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	II.1	Reactor Coolant System and Main Steam System Pressures	Conforms	None.	15.2.8

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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	II.2	Evaluation of Core Damage Potential	Conforms	For slower reactivity insertions, the analysis uses a heat generation rate limit to ensure fuel centerline melting limits are met.	15.2.8
DSRS 15.2.8, Rev 0: Feedwater System Pipe Break Inside and Outside Containment	II.3	Calculated Site Boundary Doses	Conforms	None.	15.2.8
DSRS 15.2.8, Rev 0: Feedwater System Pipe Break Inside and Outside Containment	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	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 3: Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	All	Various	Not Applicable	SRP Sections 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 is more appropriate terminology for analysis phenomena than 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, the analysis uses a heat generation rate limit to ensure fuel centerline melting limits are met.	15.4.1

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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 analysis phenomenon than DNBR.	15.4.2
SRP 15.4.2, Rev 3: Uncontrolled Control Rod Assembly Withdrawal at Power	II.1.B	Fuel Centerline Temperatures	Conforms	For slower reactivity insertions, the analysis 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 analysis 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.	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.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.	Not Applicable

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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.	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.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.	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.	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.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.	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	Critical heat flux is more appropriate terminology for analysis 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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 15.4.6, Rev 2: Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR)	II.4	Minimum Time Intervals for Required Operator Actions	Not Applicable	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
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, are applicable.	15.0.3

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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 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. However, per SRP Section 15.0.3, Section I, Areas of Review, Item 10.C under subheading Review Interfaces, the part of this acceptance criterion that specifies radiological acceptance criteria is superseded by SRP Section 15.0.3.	15.0.3
SRP 15.4.8.A, Rev 1: Radiological Consequences of a Control Rod Ejection Accident (PWR)	II (No number)	First full paragraph on page 15.4.8-6) - Technical Specification for Primary-to-Secondary Leak Rate	Partially Conforms	The part of this guidance related to the required technical specification for primary-to-secondary leak rate is applicable. 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	Various	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	Various	Not Applicable	This SRP section and its acceptance criteria are applicable only to BWRs.	Not Applicable

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.1	The frequency classification for this event is an AOO.	Conforms	None.	15.0 15.5.1
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
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
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
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
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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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 is bounded by an inadvertent ECCS valve actuation, analyzed in Section 15.6.6.	15.6.1 15.6.6
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. 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. 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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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)	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. 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
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. 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	Various	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	Partially Conforms	The 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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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
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. However, per SRP Section 15.0.3, Section I, Areas of Review, Item 10.H under subheading Review Interfaces, the part of this acceptance criterion that specifies radiological acceptance criteria is superseded by SRP Section 15.0.3. A portion of this guidance is applicable only to BWRs.	15.0.3 15.6.5

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
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. 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
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. 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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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. 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 design.	15.6.6
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 is 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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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 RXB. 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 design does not use a containment building. Rather, each NPM has its own compact steel CNV that 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 and Design Specific Review Standard (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 design does not rely on ESF ventilation systems to mitigate the consequences of a DBA. 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, AOOs, 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 design does not use a containment building. Rather, each NPM has its own compact steel CNV immediately surrounding the reactor vessel. The containment design provisions of this guidance for fuel handling operations inside containment are not relevant to the CNV design. However, the intent of this acceptance criterion is appropriate to apply to the RXB, 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 and Design Specific Review Standard (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 Reactor Building crane (RBC) is to move spent fuel casks in the RXB refueling area. The RBC system design conforms to ASME standards so that any credible failure of a single component does not result in the loss of capability to stop and hold a critical load. The use of the RBC precludes the need to perform load drop evaluations and as a result no accident analysis is performed to assess radiological consequences of a spent fuel cask drop accident or an NPM drop accident.	9.1.5 15.7.5
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 ASME standards so that any credible failure of a single component does not result in the loss of capability to stop and hold a critical load. The use of the RBC precludes the need to perform load drop evaluations and as a result no accident analysis is performed to assess radiological consequences of a spent fuel cask drop accident or an 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 ASME standards so that any credible failure of a single component does not result in the loss of capability to stop and hold a critical load. The use of the RBC precludes the need to perform load drop evaluations and as a result no accident analysis is performed to assess radiological consequences of a spent fuel cask drop accident or an NPM drop accident.	Not Applicable

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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 ASME standards so that any credible failure of a single component does not result in the loss of capability to stop and hold a critical load. The use of the RBC precludes the need to perform load drop evaluations and as a result no accident analysis is performed to assess radiological consequences of a spent fuel cask drop accident or an 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 ASME standards so that any credible failure of a single component does not result in the loss of capability to stop and hold a critical load. The use of the RBC precludes the need to perform load drop evaluations and as a result no accident analysis is performed to assess radiological consequences of a spent fuel cask drop accident or an 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 ASME standards so that any credible failure of a single component does not result in the loss of capability to stop and hold a critical load. The use of the RBC precludes the need to perform load drop evaluations and as a result no accident analysis is performed to assess radiological consequences of a spent fuel cask drop accident or an NPM drop accident.	15.7.5
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

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.2	Acceptance Criteria for Pressurized Water Reactors (PWRs)	Not Applicable	NuScale is characterized as an evolutionary plant (Addressed in the acceptance criteria in II.3).	Not Applicable
SRP 15.8, Rev 2: Anticipated Transients without Scram	II.3.A.i	Provide a diverse scram system	Partially Conforms	The 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 design relies on diversity within the reactor protection system 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	The design features required by 10 CFR 50.62(c)(1) either do not apply to the 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 reactor protection system 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	Not Applicable	None.	Not Applicable
DSRS 15.9.A, Rev 0: Thermal Hydraulic Stability Review Responsibilities	II.2	Meeting Requirements of GDC 12	Conforms	None.	4.4 15.9

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.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
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 design. Reactor trip signals prevent violation of CHF limits before unstable flow oscillations can develop. Protective action occurs before 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 specified acceptable fuel design limit (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 15.9
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	The RTS trips the reactor before conditions that could initiate flow instabilities. Stabilities are not detected and suppressed.	4.4 15.9
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	The RTS trips the reactor before 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 15.9 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 15.9

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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.11	Detect-and-suppress system extremely high probability of functioning in the event of an AOO.	Partially Conforms	The RTS is used instead of a detect-and-suppress system. Reactor trip occurs before conditions that could initiate instabilities.	4.4 15.9

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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 an applicant. For the latter, the application contains COL information items, as appropriate, that describe the required development of plant-specific technical specifications that is deferred to the applicant referencing the NuScale design. Notwithstanding the above and consistent with DSRS 16.0, the SDAA 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 and Design Specific Review Standard (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, 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 an SDAA.	16.1
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 an SDAA.	16.1
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 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 issued SRP Section 17.5 (based on NQA-1) for the review of QAPDs for new reactor applicants - including applicants for standard design approval - 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 and Design Specific Review Standard (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 issued SRP Section 17.5 for the review of QAPDs for new reactor applicants - including applicants for standard design approval - 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 SDA.	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 to applicants.	Not Applicable
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants	II.A	Organization	Partially Conforms	The onsite, offsite, operational, and maintenance organizational elements of Item II.A.3 are the responsibility of an applicant referencing the standard design.	17.5
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New License 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 standard design approval phase, and are to be addressed within the operational QA Program developed and maintained by an applicant referencing the standard design.	17.5
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants	II.C	Design Control and Verification	Conforms	None.	17.5

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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 License 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 License 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 License Applicants	II.F	Document Control	Partially Conforms	The site-specific and operational provisions of document control are the responsibility of an applicant referencing the standard design.	17.5
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New License 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 License Applicants	II.H	Identification and Control of Materials, Parts, and Components	Not Applicable	This acceptance criterion governs activities that are the responsibility of an applicant referencing the standard design.	Not Applicable
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants	II.I	Control of Special Processes	Not Applicable	This acceptance criterion governs activities that are the responsibility of an applicant referencing the standard design.	Not Applicable
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants	II.J	Inspection	Partially Conforms	Site-specific and operational activities. are the responsibility of an applicant referencing the standard design.	17.5

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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 License 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 License 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 License Applicants	II.M	Handling, Storage, and Shipping	Not Applicable	This acceptance criterion governs activities that are the responsibility of the an applicant referencing the standard design.	Not Applicable
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants	II.N	Inspection, Test, and Operating Status	Not Applicable	This acceptance criterion governs activities that are the responsibility of the an applicant referencing the standard design.	Not Applicable
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New License 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 License 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 License Applicants	II.Q	Quality Assurance Records	Conforms	None.	17.5

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard (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 License Applicants	II.R	Audits	Conforms	None.	17.5
SRP 17.5, Rev 1: Quality Assurance Program Description - Design Certification, Early Site Permit and New License 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 License 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 License 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 License 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 an applicant.	Not Applicable
SRP 18.0, Rev 3: Human Factors Engineering	II.A	Review of the HFE Aspects of a New Plant	Conforms	None.	18.1 - 18.12
SRP 18.0, Rev 3: 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 and Design Specific Review Standard (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 18.0, Rev 3: 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 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 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 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 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 updates are the responsibility of an applicant referencing the standard design.	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
SRP 19.2, Rev. 0: 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

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

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 19.3, Rev 0: Regulatory Treatment of Nonsafety 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	Not Applicable	The mitigation of beyond-design-basis events are the responsibility of an applicant referencing the standard design.	Not Applicable
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 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 (ISG)	-	This ISG section points to the guidance provided in ISG 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 design. The OBE is specified as one-third of the CSDRS thus does not require any analysis. There are COL items for the applicant to ensure the ground motion response spectra 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 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 applications was incorporated into SRP 19.0, Rev 3.	Not Applicable
COL/ESP-ISG-4: Definition of Construction and on Limited Work Authorizations	Not Applicable	Not Applicable	Not Applicable	Closed: This ISG has been incorporated into RG 1.206, Revision 1.	Not Applicable

Table 1.9-4: Conformance with Interim Staff Guidance (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 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 NPP 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	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 applicant referencing the 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.	12.3
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 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	Not Applicable	Not Applicable	Not Applicable	Closed: This ISG has been incorporated into RG 1.206, Revision 1.	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
DC/COL-ISG-11: Finalizing Licensing-basis Information	Not Applicable	Not Applicable	Not Applicable	Closed: This ISG has been incorporated into RG 1.206, Revision 1.	Not Applicable

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

ISG Section/ Title	AC	AC Title / Description	Conformance Status	Comments	Section
DC/COL-ISG-13: Assessing the Radiological Consequences of Accidental Releases of Radioactive Materials from Liquid Waste Tanks for Combined License Applications	1	Failure Mechanism and Radioactivity Releases	Partially Conforms	Site-specific aspects are the responsibility of the applicant.	11.2
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 applicant.	11.2
DC/COL-ISG-13	3	Radioactive Source Term (Including Attachment A)	Partially Conforms	Site-specific aspects are the responsibility of the applicant.	11.2
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 applicant referencing the approved 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 applicant referencing the approved design.	Not Applicable
DC/COL-ISG-13	6	SRP Dose Acceptance Criteria	Partially Conforms	Site-specific aspects are the responsibility of the applicant.	11.2
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 applicant.	11.2
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 (Continued)

ISG Section/ Title	AC	AC Title / Description	Conformance Status	Comments	Section
DC/COL-ISG-14: Assessing the Radiological Consequences of Accidental Releases of Radioactive Materials from Liquid Waste Tanks in Ground and Surface Waters for Combined License Applications	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 applicant referencing the approved design.	Not Applicable
ESP/DC/COL-ISG-15: Post-Combined License Commitments	Not Applicable	Not Applicable	Not Applicable	Closed: This ISG has been incorporated into RG 1.206, Revision 1.	Not Applicable
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 standard design approval 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 site-specific seismic design information submitted to support COL applications.	Not Applicable
DC/COL-ISG-19: Gas Accumulation Issues in Safety Related Systems	All	Various	Not Applicable	This guidance is applicable 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 ECCS and DHRS (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 (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 seismic margin analysis submitted to support applications was 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 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 design uses onsite backup diesel generators instead of gas turbine generators. However, regardless of the type of standby AC generation used in the 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 DBA.	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
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, licensees with proposed changes to their current licensing basis during construction.	Not Applicable

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

ISG Section/ Title	AC	AC Title / Description	Conformance Status	Comments	Section
COL/ESP-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
COL/ESP-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
COL-ISG-29: Environmental Considerations Associated with Micro-reactors	Not Applicable	Not Applicable	Not Applicable	This guidance is applicable to micro-reactors. The NPM is a small modular reactor, so this guidance is not applicable.	Not Applicable
Digital I&C-ISG-01: Cyber Security	Not Applicable	Not Applicable	Not Applicable	Closed: This guidance was incorporated into RG 5.71, “Cyber Security Programs for Nuclear Facilities,” and RG 1.152, Revision 3, “Criteria for Use of Computers in Safety Systems of Nuclear Power Plants.”	Not Applicable
Digital I&C-ISG-02: Diversity and Defense-in-Depth (D3)	Not Applicable	Not Applicable	Not Applicable	Closed: This guidance was incorporated into Branch Technical Position 7-19, Revision 6, “Guidance for Evaluation of Diversity and Defense-in-Depth on Digital Computer-Based Instrumentation and Control Systems.” Digital I&C-ISG-02 is not applicable. DSRS 7.1.5 in Table 1.9-3 provides information on the Diversity and Defense-in-Depth review.	Not Applicable

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

ISG Section/ Title	AC	AC Title / Description	Conformance Status	Comments	Section
Digital I&C-ISG-03: Review of New Reactor Digital Instrumentation and Control Probabilistic Risk Assessments	4	Staff Position	Not Applicable	Digital I&C-ISG-03 is not applicable. DSRS 7.0 in Table 1.9-3 provides an overview of the I&C review process.	Not Applicable
Digital I&C-ISG-04: Highly Integrated Control Rooms - Communications Issues (HICRc)	1	Interdivisional Communications - Staff Position (Pages 4 through 8)	Not Applicable	Digital I&C-ISG-04 is not applicable; 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
Digital I&C-ISG-04	2	Command Prioritization - Staff Position (Pages 8 through 10)	Not Applicable	Digital I&C-ISG-04 is not applicable; 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
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; 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
Digital I&C-ISG-04	3.1	Independence and Isolation (Pages 11 through 13)	Not Applicable	Digital I&C-ISG-04 is not applicable; 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
Digital I&C-ISG-04	3.2	Human Factors Considerations (Pages 13 through 15)	Not Applicable	Digital I&C-ISG-04 is not applicable; 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
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; 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
Digital I&C-ISG-05: Highly Integrated Control Rooms - Human Factors Issues (HICR-HF)	All	Various	Partially Conforms	Closed: This guidance was incorporated into NUREG-0800, Chapter 18, Revision 3, Human Factors Engineering. This position is applicable except for site-specific operational elements of subtier NUREG-0899 that are the responsibility of the applicant.	18.7

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

ISG Section/ Title	AC	AC Title / Description	Conformance Status	Comments	Section
Digital I&C-ISG-06, Rev 2: 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: Digital Instrumentation and Control Systems in Safety Applications at 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 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 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 2: Compliance with Order EA-12-049 Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events	Not Applicable	Not Applicable	Not Applicable	Closed: Per 10 CFR 50.155(h), Order EA-12-049 was withdrawn for each licensee or construction permit holder issued those Orders. RG 1.226, in part, makes the requirements of EA-12-049 generically applicable.	Not Applicable
JLD-ISG-12-02, Draft: Interim Staff Guidance for Compliance with Order EA-12-050 Concerning Reliable Hardened Vents	Not Applicable	Not Applicable	Not Applicable	This ISG has been withdrawn.	Not Applicable

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

ISG Section/ Title	AC	AC Title / Description	Conformance Status	Comments	Section
JLD-ISG-12-03, Draft: Compliance with Order EA-12-051 Reliable Spent Fuel Pool Instrumentation	Not Applicable	Not Applicable	Not Applicable	Closed: Per 10 CFR 50.155(h), Order EA-12-051 was withdrawn for each licensee or construction permit holder issued those Orders. RG 1.227, in part, makes the requirements of EA-12-049 generically applicable.	Not Applicable
JLD-ISG-12-04, Rev 0: Guidance on Performing a Seismic Margin Assessment in Response to the March 2012 Request for Information Letter	Not Applicable	Not Applicable	Not Applicable	This ISG has been withdrawn.	Not Applicable
JLD-ISG-12-05, Rev 0: Guidance for Performance of an Integrated Assessment for Flooding	All	Various	Not Applicable	This ISG is for response to the 10 CFR 50.54(f) March 2012 request for information letter.	Not Applicable
JLD-ISG-12-06, Rev 0: Guidance for Performing a Tsunami, Surge, or Seiche Hazard Assessment	All	Various	Not Applicable	This ISG is for response to the 10 CFR 50.54(f) March 2012 request for information letter.	Not Applicable
JLD-ISG-13-01, Rev 0: Guidance for 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 applicant.	Not Applicable
JLD-ISG-13-02, Draft: Compliance with 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

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

ISG Section/ Title	AC	AC Title / Description	Conformance Status	Comments	Section
JLD-ISG-15-01, 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
JLD-ISG-16-01, Rev 0: Guidance for Activities Related to Near-Term Task Force Recommendation 2.1, Flooding Hazard Reevaluation; Focused Evaluation and Integrated Assessment	Not Applicable	Not Applicable	Not Applicable	This ISG has been withdrawn.	Not Applicable
DSS-ISG-2010-01, Rev 0, Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools	Not Applicable	Not Applicable	Not Applicable	This guidance was incorporated into RG 1.240, “Fresh and Spent Fuel Pool Criticality Analysis.”	Not Applicable

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

ISG Section/ Title	AC	AC Title / Description	Conformance Status	Comments	Section
DSS-ISG-2016-01, Rev 1, Clarification of Licensee Actions in Receipt of Enforcement Discretion Per Enforcement Guidance Memorandum EGM 15-002, “Enforcement Discretion for Tornado-generated Missile Protection Noncompliance”	Not Applicable	Not Applicable	Not Applicable	This ISG expired with the close out of EGM 15-002.	Not Applicable
ATF-ISG-2020-01: Supplemental Guidance Regarding the Chromium-Coated Zirconium Alloy Fuel Cladding Accident Tolerant Fuel Concept	Not Applicable	Not Applicable	Not Applicable	This guidance is applicable to fuel designs with chromium-coated zirconium fuel rods.	Not Applicable

Table 1.9-5: Conformance with Three Mile Island 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 approval addresses reliability of core and containment heat removal systems, with an update required by an applicant to reflect site-specific conditions.	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 systems. The design does have an AFWS.	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 design does not require or include 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	The 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, and the issue contemplated by this requirement is related to PORVs. The design does not use PORVs.	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
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

Table 1.9-5: Conformance with Three Mile Island Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933) (Continued)

Item	Regulation Description / Title	Conformance Status	Comments	Section
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 52.137(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 standard design.	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 applicant referencing the approved 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 applicant referencing the approved 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 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 18.7

Table 1.9-5: Conformance with Three Mile Island 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
50.34(f)(2)(vi)	Provide the capability of high point venting of noncondensible gases from the RCS, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. (II.B.1)	Departure	The design supports an exemption from 10 CFR 50.34(f)(2)(vi).	5.4
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 design does not contain vital areas, as defined by NUREG-0737, Item II.B.2, other than the 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 RCS and containment that may contain accident source term radioactive materials (II.B.3)	Departure	The design does not rely on primary coolant or containment samples to assess the extent of potential core damage. The 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 RCS 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 52.137(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 standard design.	Not Applicable

Table 1.9-5: Conformance with Three Mile Island 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 RCS relief and safety valves and, for PWRs, PORV block valves (II.D.1)	Departure	This requirement is applicable except for aspects specifying PORV block valve testing and consideration of ATWS conditions in the testing program. The NuScale design does not use PORVs. The ATWS provision is not technically relevant to the 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 design supports an exemption from 10 CFR 50.62(c)(1) because the 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
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

Table 1.9-5: Conformance with Three Mile Island 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 AFWS initiation, and provide AFWS flow indication in the control room (II.E.1.2)	Not Applicable	The design does not have an AFWS. Also, while the DHRS performs some of the functions of an AFWS at a PWR, the DHRS is designed for design-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 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 RCS pressure and temperature. 10 CFR 50.34(f)(2)(xii) is not considered applicable to the DHRS. Because the language and intent of 10 CFR 50.34(f)(1)(ii) do not apply, the requirement is not applicable to the 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 design equivalent to hot standby condition as stated in 10 CFR 50.34(f)(2)(xiii) is hot shutdown condition. The design does not rely on pressurizer heaters to establish and maintain natural circulation in hot shutdown conditions and supports an exemption from 10 CFR 50.34(f)(2)(xiii).	5.4 8.3

Table 1.9-5: Conformance with Three Mile Island 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 setpoint 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 design conforms to 50.34(f)(2)(xiv)(A), (B), (C), and (D). The design supports an exemption from 10 CFR 50.34(f)(2)(xiv)(E) with respect to the containment evacuation system. 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 setpoint 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 prevents a release to the environs. For 50.34(f)(2)(xiv)(E), the design differs from that of a traditional LWR 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 containment evacuation system 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 containment evacuation system to the environs.	5.2 6.2.4 7.1 7.2 9.3.6 19.2

Table 1.9-5: Conformance with Three Mile Island 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)	Departure	The CNV 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 ECCS design does not include pumps and does not involve a typical pressurized water reactor 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 not technically relevant to the 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 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)	Departure	The design supports an exemption to 10 CFR 50.34(f)(2)(xvii)(C).	6.2.1 7.1 7.2 9.3.2 11.5 12.3
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 6.3 7.2
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 7.2 19.2

Table 1.9-5: Conformance with Three Mile Island Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933) (Continued)

Item	Regulation Description / Title	Conformance Status	Comments	Section
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 design supports an exemption from the portions of 10 CFR 50.34(f)(2)(xx) related to pressurizer level indicators.	5.4 7.2 8.1 8.3
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 integrated control system 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.	9.5.2 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 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 applicant or licensee.	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

Table 1.9-5: Conformance with Three Mile Island 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 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 applicant or licensee.	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 extent it is relevant to design activities in support of the SDA. 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 applicant or licensee.	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 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 CNV 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, the calculated peak containment pressure 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. (TR-0716-50424, Section 2.8).	6.2 19.2

Table 1.9-5: Conformance with Three Mile Island Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933) (Continued)

Item	Regulation Description / Title	Conformance Status	Comments	Section
50.34(f)(3)(v)	Preliminary Design Information - Containment Integrity (II.B.8)	Not Applicable	Pursuant to 10 CFR 52.137(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 standard design.	Not Applicable
50.34(f)(3)(vi)	For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations (II.E.4.1)	Not Applicable	The 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 to applicants and holders of reactor facility licenses.	Not Applicable
Issue 191	Assessment of Debris Accumulation on PWR Sump Performance	Conforms	Based upon the design and cleanliness requirements, there is minimal debris generation and accumulation expected, and it does not adversely impact the ability of ECCS to perform its required functions. Debris limits address generic safety issue GSI-191. The results of evaluation of the effects of fibrous, particulate, and chemical debris in the reactor coolant on the long-term cooling capability demonstrate that long-term core cooling is not adversely impacted. Evaluations assess the debris impact on ECCS components, the fuel, and the core.	6.3
Issue 193	Boiling Water Reactor Emergency Cooling Water System (ECCS) Suction Concerns	Not Applicable	This Issue is specific to BWRs.	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 instrument and control air system provides instrument air that meets the quality standards of ANSI/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 passive plant design are considered in the design of electrical systems.	8.1 8.3
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 EDAS.	8.1 8.3
Generic Letter 96-01	Testing of Safety-Related Logic Circuits	Conforms	None.	7.2
Generic Letter 2006-02	Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power	Not Applicable	The design does not rely on offsite power for safety-related or risk-significant functions. Grid stability studies are the responsibility of an applicant that references the NuScale US460 standard plant design.	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 applicant applying the guidance of RG 1.218.	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	Gas accumulation buildup does not impact ECCS under accident conditions. The DHRS does not interface with the RCS. It is connected to the secondary system.	5.4 Ch 6
Generic Letter 2016-01	Monitoring Of Neutron-Absorbing Materials In Spent Fuel Pools	Not Applicable	Not applicable to 10 CFR Part 52 applicants. NEI 16-03-A is used in lieu of the absorber monitoring program.	Not Applicable
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	The applicant is responsible for addressing mitigation of beyond-design-basis events in accordance with 10 CFR 50.155.	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 8.2.2.	8.2

Table 1.9-7: Conformance with Advanced and Evolutionary Light Water Reactor Design Issues (SECYs and Associated Staff Requirements Memoranda)

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. Table 1.9-8 contains 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 advanced light water reactor (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. Table 1.9-8 contains 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
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
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
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 Staff Requirements Memoranda) (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	Partially Conforms	Table 1.9-8 provides further information.	1.9
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 NPP.	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
SECY-21-0039	Elimination of the Shift Technical Advisor for the NuScale Design	Conforms	SECY-21-0039 pertains to staff evaluation and acceptance of elimination of the shift technical advisor for the NuScale design, presented in TR-0420-69456-P-A, Revision 1.	18.5

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 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 residual heat removal systems in PWR.	Not Applicable	Design does not use external loops and no drain down condition for refueling.	Not Applicable
I.D	Station Blackout: 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 interfacing systems loss-of-coolant accident.	Conforms	None.	9.3.4 19.2
I.G	Hydrogen Control: Position on acceptable requirements to measure and mitigate the effects of hydrogen produced due to a water reaction with zirconium fuel cladding.	Partially Conforms	The design includes a PAR that is sized to limit oxygen concentrations to a level that does not support combustion (less than four percent), this results in an inert containment atmosphere. The NuScale design supports an exemption to 10 CFR 50.44(c)(4).	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

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.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
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
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 one-third of the SSE it is decoupled from the design evaluation process.	3.7
I.N	Inservice 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	The standard design uses the latest endorsed codes and standards or others on case by case basis.	All FSAR Sections
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 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
II.D	Leak-Before-Break: Position on use of leak-before-break concept.	Not Applicable	The design uses break exclusion criteria.	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 270 mph found in RG 1.76 Revision 1 rather than the outdated 300 mph guidance found in SECY-93-087.	3.3

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

Issue	Description	Conformance Status	Comments	Section
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 (before rule change).	Partially Conforms	None.0	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.	Not Applicable	The design supports an exemption from 10 CFR 50.34(f)(2)(viii).	9.3.2
II.J	Level of Detail: Position on a design certification submittal with depth of detail similar to that in an FSAR.	Conforms	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
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
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

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.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
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
III.D	Safe Shutdown Requirements: Position on using non-safety grade active cooling systems to bring a reactor to cold shutdown because 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 CNV walls.	3.1.4 5.4 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

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.G	Simplification of Offsite Emergency Planning: Position on simplifying off-site emergency planning of passive designs because of 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 Sites With Multiple Nuclear Power Plants

COL Item 1.10-1: An applicant that references the NuScale Power Plant US460 standard design will evaluate the potential hazards resulting from construction activities of the new NuScale facility to an operating nuclear power plant on a co-located site per 10 CFR 52.79(a)(31).