August 19, 2024 Docket No. 052-050

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of Presentation Material Entitled "ACRS Subcommittee Meeting (Open Session) US460 Standard Design Approval Application Chapters 7, 9, 12, and 18," PM-172558, Revision 0

The purpose of this submittal is to provide presentation materials for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) NuScale Subcommittee Meeting on August 22, 2024. The materials support NuScale's presentation of the subject chapters of the US460 Standard Design Approval Application.

The enclosure to this letter is the nonproprietary presentation entitled "ACRS Subcommittee Meeting (Open Session) US460 Standard Design Approval Application Chapters 7, 9, 12, and 18," PM-172558, Revision 0.

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

If you have any questions, please contact Chelsea Lockwood at 541-452-7171 or at clockwood@nuscalepower.com.

Sincerely,

Thomas Griffith
Manager, Licensing
NuScale Power, LLC

Distribution: Mahmoud Jardaneh, Chief New Reactor Licensing Branch, NRC
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Enclosure 1: "ACRS Subcommittee Meeting (Open Session) US460 Standard Design Approval Application Chapters 7, 9, 12, and 18," PM-172558, Revision 0

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"ACRS Subcommittee Meeting (Open Session) US460 Standard Design Approval Application Chapters 7, 9, 12, and 18," PM-172558, Revision 0

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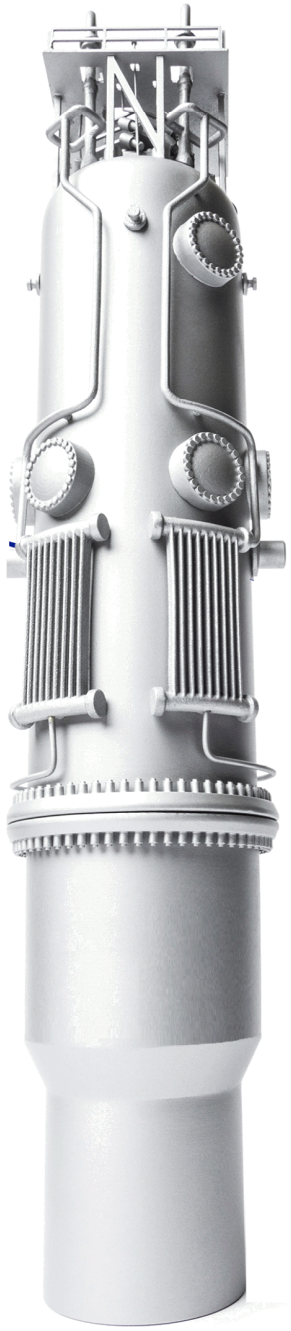


ACRS Subcommittee Meeting
(Open Session)

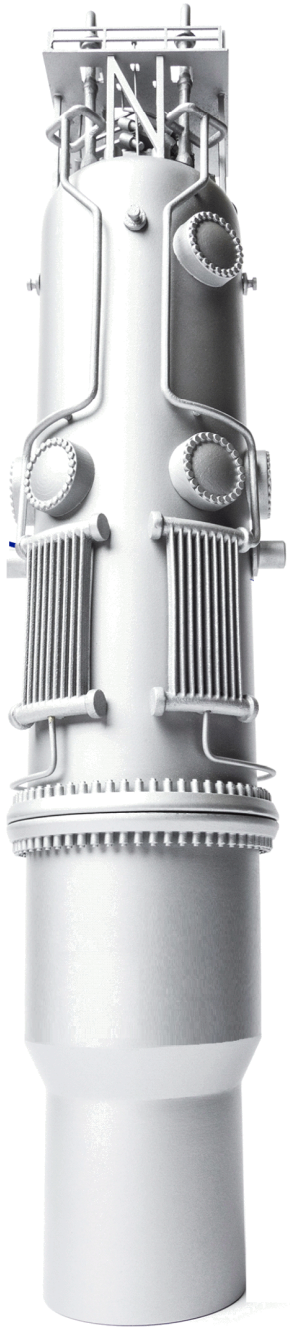
August 22, 2024

US460 Standard Design Approval Application

Chapters 7, 9, 12, and 18

Acknowledgement and Disclaimer

This material is based upon work supported by the Department of Energy under Award Number DE-NE0008928.



Chapter 7

Instrumentation and Controls

August 22, 2024

Presenter: Thomas Case

Instrumentation and Controls

- Section 7.0: Instrumentation and Controls – Introduction and Overview
- Section 7.1: Fundamental Design Principles
- Section 7.2: System Features

Section 7.0: Instrumentation and Controls – Introduction and Overview

- Changes from DCA
  - Elimination of the remote shutdown station (across all of Chapter 7)
    - Alternate operator workstations allow for plant monitoring outside the main control room

- Results from audit and RAI review
  - No audit items or RAIs specific to Section 7.0 [1]

[1] One audit item in Chapter 15 related to COL Item 7.0-1

Section 7.1: Fundamental Design Principles

- Changes from DCA
  - Module protection system (MPS) setpoint changes due to changes in operating pressure and temperature, and updated safety analysis
  - Emergency core cooling system (ECCS) actuation changes due to updated safety analysis
  - Additional decay heat removal system (DHRS) and reactor trip system (RTS) actuations due to updated safety analysis
  - Addition of an 8-hour timer for ECCS actuation to add supplemental boron if needed to maintain subcriticality
  - Adoption of Institute of Electrical and Electronics Engineers (IEEE) Standard 497-2016 as endorsed by Regulatory Guide 1.97, Revision 5, and the addition of Type F post-accident monitoring variables

- Results from audit and RAI review
  - Removal of a note from Figure 7.1-1aa regarding inadvertent actuation block (A-7.1-1)

Section 7.2: System Features

- Changes from DCA
  - Information from the Advanced Sensor Technical Report cited in the DCA is incorporated into SDAA Section 7.2.16
  - Change from digital to analog sensors
    - Reactor pressure vessel riser level, containment vessel water level, reactor coolant system pressure, pressurizer pressure
    - Diversity and defense-in-depth (D3) and coping analyses updated in Section 7.1
  - Reduction in quantity of reactor coolant system temperature sensors based on updated engineering evaluation
    - Analysis of the reactor coolant flow determined that streaming effects do not require the use of multiple sensors per quadrant

- Results from audit and RAI review
  - No audit items or RAIs specific to Section 7.2

SDAA Topical Report – Design of the Highly Integrated Protection System Platform TR-1015-18653-P-A Revision 2

- NRC approved Topical Report (June 2017)
- No changes since ACRS engagement in 2023

SDAA Technical Report – NuScale Instrument Setpoint Methodology Technical Report TR-122844 Revision 0

- NRC reviewed as part of Chapter 7
- No changes since ACRS engagement in 2023

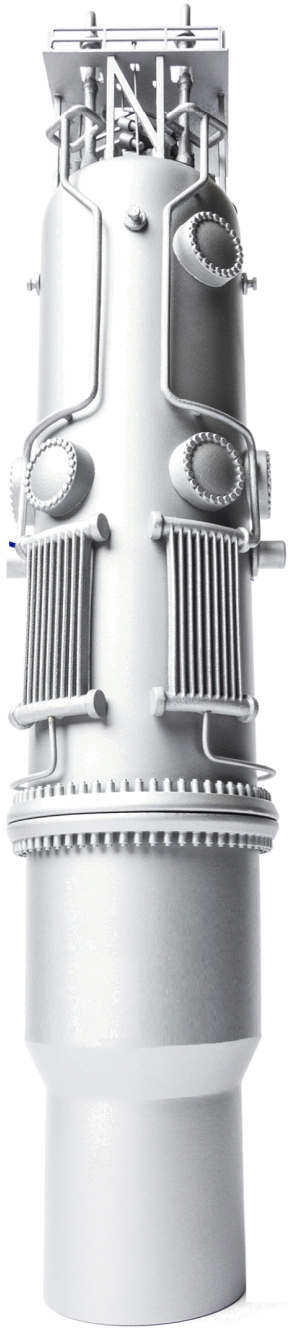
COL items

- No change to COL Items in Chapter 7 [1]

[1] One audit item in Chapter 15 related to COL Item 7.0-1

Acronyms

ACRS	Advisory Committee on Reactor Safeguards
COL	Combined License
D3	Diversity and Defense-in-depth
DCA	Design Certification Application
DHRS	Decay Heat Removal System
ECCS	Emergency Core Cooling System
IEEE	Institute of Electrical and Electronics Engineers
MPS	Module Protection System
NRC	Nuclear Regulatory Commission
RAI	Request for Additional Information
RTS	Reactor Trip System



Chapter 9

Auxiliary Systems

August 22, 2024

Presenters: Sarah Turmero, Freeda Ahmed, and Jordan Green

Chapter 9: Auxiliary Systems

- Section 9.1 Fuel Storage and Handling
- Section 9.2 Water Systems
- Section 9.3 Process Auxiliaries
- Section 9.4 Air Conditioning, Heating, Cooling, and Ventilation Systems
- Section 9.5 Other Auxiliary Systems (Lighting, Communication, and Fire Protection)
- Section 9A Fire Hazards Analysis

Section 9.1: Fuel Storage and Handling

- Section 9.1.1 Criticality Safety of Fresh and Spent Fuel Storage and Handling
- Section 9.1.2 New and Spent Fuel Storage
- Section 9.1.3 Pool Cooling and Cleanup System
- Section 9.1.4 Fuel Handling Equipment
- Section 9.1.5 Overhead Heavy Load Handling Systems

Section 9.1.1 Criticality Safety of Fresh and Spent Fuel Storage and Handling

- Criticality safety of fuel storage is addressed by COL Item 9.1-1.
- 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.
- Audit and RAI Results:
  - Design of reactor flange tool is responsibility of COL applicant. Fuel remains in the lower reactor pressure vessel (RPV). Criticality of fuel while in the RPV is discussed in Section 4.3.2.6. (A-9.1.1-1)
  - COL Item 9.1-2 requires applicants to perform criticality analysis of fuel racks (A-9.1.1-2)
  - Criticality safety design for refueling pool is described in Section 9.2.5 (A-9.1.1-3)

Section 9.1.2 New and Spent Fuel Storage

- Fuel storage is addressed by COL Item 9.1-2.
- 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.
- Audit and RAI Results:
  - Demonstrated that the spent fuel pool (SFP) has >30 days of water above the top of fuel. (RAI 9.1.2-1.1)
  - Clarified the seismic classification between dry dock gate and the dry dock gate support in Section 9.1.2 and Section 9.1.3 (RAI 9.1.2-1.2)

Section 9.1.3 Pool Cooling and Cleanup System

- Design changes from DCA:
  - Combined the spent fuel pool cooling system, the reactor pool cooling system, the pool cleanup system, and the pool surge control system into a single pool cooling and cleanup system.
    - Major components remain the same (filters, demineralizers, surge control tank) with the exception of the pumps and heat exchangers, which have been reduced from five trains to three.
  - Pool leakage detection system wall leak channels attach to the steel-plate composite walls.
- Audit and RAI Results:
  - None specific to Section 9.1.3.

Section 9.1.4 Fuel Handling System

- Design changes from DCA:
  - New fuel elevator capable of handling irradiated fuel for inspection purposes
  - New fuel jib crane classification changed from ASME NUM-1 Type 2 to ASME NUM-1 Type 1A, single failure proof
- Audit and RAI Results:
  - None specific to Section 9.1.4

Section 9.1.5 Overhead Heavy Load Handling Systems

- Design Changes from DCA:
  - Reactor Building crane (RBC) capacity increased from 850 tons to 950 tons
  - Module lifting adapter in DCA design was removed and is now integral to the RBC
  - RBC auxiliary hoist capacity increased from 15 tons to 40 tons
  - Changes to automated control system software reduces probability of operator error
  - Added additional jib cranes designed to ASME NUM-1 Type 1A
  - Removal of heavy load exclusion zone above the SFP
- Change to 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.

Section 9.1.5 Overhead Heavy Load Handling Systems, continued

- Audit and RAI Results:
  - Clarifies use of ASME NUM-1 within Section 9.1.5 including demonstrating compliance with Regulatory Guide 1.244 Position C.1 (A-9.1.5-1, A-9.1.5-2, A-9.1.5-8)
  - Justifies deviation of ASME NOG-1 design factor for plate buckling and the methodology for determining spacing of transverse stiffeners (RAI 9.1.5-3)
  - Eliminates heavy load exclusion zone terminology in 9.1.5, 15.7.5, and 17.4 (RAI 9.1.5-6)

Section 9.2: Water Systems

- Section 9.2.1 Station Service Water System - Not applicable to US460
- Section 9.2.2 Reactor Component Cooling Water System
- Section 9.2.3 Demineralized Water System
- **Section 9.2.4 Potable and Sanitary Water Systems**
- **Section 9.2.5 Ultimate Heat Sink**
- Section 9.2.6 Condensate Storage Facilities
- **Section 9.2.7 Site Cooling Water System**
- Section 9.2.8 Chilled Water System
- **Section 9.2.9 Utility Water Systems**

Section 9.2.4 Potable and Sanitary Water Systems

- Changes from DCA
  - Potable and Sanitary Water System piping (including loop seals) penetrating the control room envelope changed from Seismic Category II (SC-II) to Seismic Category I (SC-I).

- Audit and RAI Results:
  - Removal of COL Item 9.2-2 concerning source and pre-treatment methods of potable water
  - Removal of COL Item 9.2-3 concerning sanitary waste storage and disposal
  - The potable and sanitary water systems serve no safety-related functions, are not credited for mitigation of design-basis accidents, and have no safe shutdown functions. Site-specific characteristics do not impact ability to meet the identified requirements (A-9.2.4-1)

Section 9.2.5 Ultimate Heat Sink (UHS)

- Changes from DCA
  - The number of modules was reduced from 12 to six, reducing the inventory of the UHS due to a smaller footprint.
  - UHS level lowered from 68 ft to 53 ft from bottom of module.

- No Audit questions or RAIs specific to Section 9.2.5

Section 9.2.7 Site Cooling Water System

- Changes from DCA
  - Changed from one-loop open system to a two-loop system in SDAA to better maintain water quality for plant users
    - The two-loop system consists of a closed loop that removes heat from plant loads and an open cooling tower loop that rejects heat to the environment.
  - Removed COL Item 9.2-4 concerning long-term corrosion and fouling
    - The site cooling water system serves no safety-related functions, is not credited for mitigation of design-basis accidents and has no safe shutdown functions. Site-specific characteristics do not impact ability to meet the identified requirements.

- Audit and RAI Results:
  - Discussed that utility water provides makeup to the tower basin and demineralized water provides makeup to the closed loop (A-9.2.7-1)

Section 9.2.9 Utility Water Systems (UWS)

- Changes from DCA
  - Removed COL Item 9.2-5 concerning identification of a site-specific water source and water treatment system
    - This item was written for a previous revision of Regulatory Guide 1.206, which does not apply to the SDAA. The UWS meets GDCs 5, 60, 64 and 10 CFR 20.1406.

- Audit and RAI Results:
  - The UWS provides raw water to the demineralized water system, site cooling water and fire protection for general washdown use. The system function determines the system chemistry controls. The selected source of raw water has no impact on the safety-related structures systems and components (SSC) (A-9.2.9-1)
  - Discussed UWS piping in the vicinity of safety-related or SC-I SSC; and protective measures to avoid impact on system from flooding (A-9.2.9-2)

Section 9.3: Process Auxiliaries

- Section 9.3.1 Compressed Air Systems
- Section 9.3.2 Process Sampling System
- Section 9.3.3 Equipment and Floor Drain Systems
- **Section 9.3.4 Chemical and Volume Control System**
- Section 9.3.5 Standby Liquid Control System
- **Section 9.3.6 Containment Evacuation System**
- Section 9.3.7 Containment Flooding and Drain System

Section 9.3.4 Chemical and Volume Control System (CVCS)

- Design changes from DCA:
  - Module heatup system modified to use an electric heater in lieu of an auxiliary boiler

- Audit and RAI Results:
  - Clarified how CVCS complies with General Design Criterion (GDC) 4 (A-9.3.4-1)
  - Revised demineralized water system isolation valves to Quality Group C (RAI 9.3.4-1)
  - Added the maximum boron concentration for the boron addition system to prevent boric acid precipitation (RAI 9.3.4-2)
  - Provided additional design information regarding flow-restricting venturis to support probabilistic risk assessment (RAI 9.3.4-3)

Section 9.3.6 Containment Evacuation System (CES)

- Design changes from DCA:
  - The CES inlet pressure instrumentation and connecting piping, up to and including isolation valves, are designed to SC-I standards (SC-III in DCA), which ensures these components maintain capability to perform their function during and after a safe shutdown earthquake.
  - The same section of piping was increased to containment design pressure (Table 19.1-3) to act as a diverse independent backup to the CIVs in support of probabilistic risk assessment. (RAI 19.1-52)

- Audit and RAI Results:
  - Containment pressure correlates to a reactor coolant system leak rate (A-9.3.6-1)
  - Correction factor to account for water vapor bypass is calculated (A-9.3.6-2)
  - Vacuum pump removal of water vapor resulting from leaks inside containment (A-9.3.6-3)

Section 9.4: Air Conditioning, Heating, Cooling, and Ventilation Systems

- Section 9.4.1 Control Room Area Ventilation System
- Section 9.4.2 Reactor Building and Spent Fuel Pool Area Ventilation System
- Section 9.4.3 Radioactive Waste Building Ventilation System
- Section 9.4.4 Turbine Building Ventilation System
- Section 9.4.5 Engineered Safety Feature Ventilation System

Section 9.5: Other Auxiliary Systems

- **Section 9.5.3 Lighting Systems**
- **Section 9.5.2 Communication Systems**
- **Section 9.5.1 Fire Protection Program**

Section 9.5.3 Lighting Systems

- Changes from the DCA
  - Main control room (MCR) has dedicated emergency lighting that is continuously on

- Audit and RAI Results:
  - Clarified illumination levels for normal and emergency lighting (RAI 9.5.3-1)
  - Explained how manual fire suppression would be handled with emergency lighting (A-9.5.1-2)
  - Clarified illumination levels outside MCR (RAI 9.5.3-3)

Section 9.5.2 Communication System

- Changes from the DCA
  - Sound-powered telephone system was removed
  - Health physics network added to the communication system
  - Removed COL Item 9.5-2 concerning the location of security power equipment within a vital area
    - Now part of the standard plant design

- No Audit items or RAIs specific to Section 9.5.2

Section 9.5.1 Fire Protection Program

- Changes from DCA
  - Building layout change

- Audit and RAI Results:
  - Confirmed containment cable design attributes (A-9.5.1.2.4-1)
  - Explained that fixed emergency lighting is not required for post-fire safe shutdown functions or alternative safe shutdown functions (A-9.5.1-2 & A-9.5.3-1)
  - Discussed structural and electrical raceway fire barrier requirements (A-9.5.1-1)

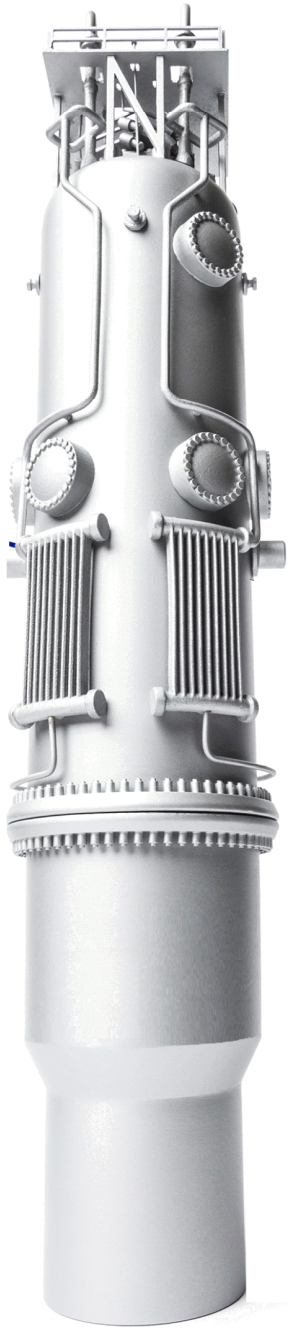
Section 9A: Fire Hazards Analysis

- Design changes from DCA
  - Building layout change

- Audit and RAI Results:
  - Clarified safe shutdown requirements vs capabilities concerning MCR evacuation (A-9A.6.4.1-1 & -2)
  - Clarified the fire hazards analysis to state which rooms did not contain safe shutdown equipment and discussed how propagation is mitigated (RAI 9A.5-1)

Acronyms

ASME	American Society of Mechanical Engineers
CES	Containment Evacuation System
CIV	Containment Isolation Valve
COL	Combined License
CVCS	Chemical and Volume Control System
DCA	Design Certification Application
GDC	General Design Criterion
MCR	Main Control Room
RAI	Request for Additional Information
RPV	Reactor Pressure Vessel
RBC	Reactor Building Crane
SFP	Spent Fuel Pool
SC-I	Seismic Category I
SC-II	Seismic Category II
SSC	Structures, Systems, and Components
SDAA	Standard Design Approval Application
UHS	Ultimate Heat Sink
UWS	Utility Water System



Chapter 12
Radiation Protection

August 22, 2024

Presenter: Erik Slobe

Chapter 12: Radiation Protection

- Section 12.1 Ensuring that Occupational Radiation Exposures Are as Low as Reasonably Achievable
- Section 12.2 Radiation Sources
- Section 12.3 Radiation Protection Design Features
- Section 12.4 Dose Assessment
- Section 12.5 Operational Radiation Protection Program

Section 12.1 Ensuring that Occupational Radiation Exposures Are as Low as Reasonably Achievable

- Same methodology as the Design Certification Application (DCA)

Section 12.2 Radiation Sources

- Same methodology as DCA
- Updated source term information in Tables 12.2-1 through Table 12.2-31
  - Updated for change in cycle length, increase in burnup rate, change in thermal power, and change in number of NuScale Power Modules
  - Design basis failed fuel fraction is applied to one reactor for shared system source terms (11.1).
- Audit results
  - Dose rate for workers on the fuel handling machine (A-12.2.1.8-1)
  - Decay of N-16 to insignificant levels in the chemical volume control system flow path (A-12.2-5)
  - Source terms for components of the low conductivity waste (LCW) processing skid, including LCW filters, ion exchanger, accumulators, and LCW polishers (A-12.2-3)

Section 12.3 Radiation Protection Design Features

- Same methodology as DCA
- Differences from DCA
  - No very high radiation areas
  - Reduction in number of reactor building and radwaste building shield doors
  - Reduction in fixed radiation monitors
  - Removal of COL Item 12.3-5 on additional area radiation monitors
    - Design criteria for area radiation monitors are included in Section 12.3.4.2
  - Removal of COL Item 12.3-8 on radiation shielding for shield wall penetrations
    - Completion of more detailed shielding analyses
- Audit Results
  - Break pot tanks in phase separator tank and spent resin storage tank vent lines replaced with hooded vents (A-12.3.1.1-2)
- RAI Results
  - Shielding based on nominal concrete equivalent gamma attenuation (RAI 12.3-1)
  - Radiation monitors under the bioshield are post accident monitoring system B, C, and F variables (RAI 12.3.4.2-1)

Section 12.4 Dose Assessment

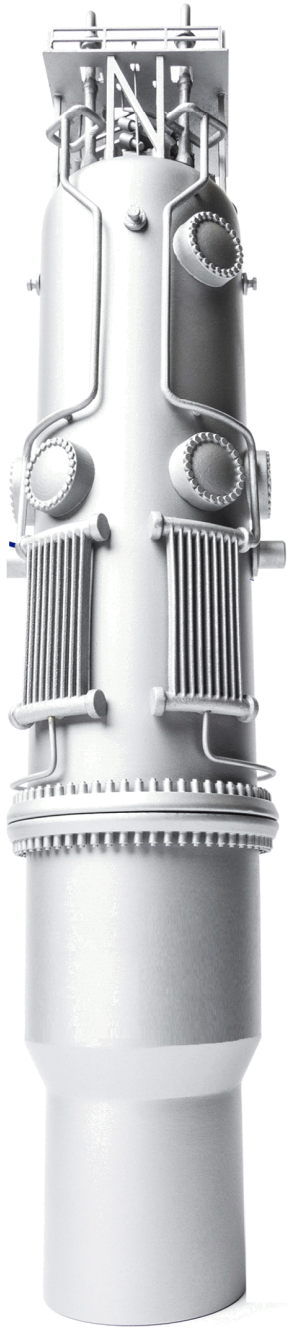
- Same methodology as DCA
- Changes from DCA
  - Updated for change in cycle length, increase in burnup rate, change in thermal power, change in number of Nuscale Power Modules, building layout changes, and operational optimizations
  - Vital areas for post-accident actions do not include areas for initiating combustible gas monitoring
- Audit Results
  - An update to COL Item 12.4-1 includes changing the dose to construction workers from co-located existing operating NuScale Power Plants to a responsibility of the applicant. (A-12.4.1.9-1)

Section 12.5 Operational Radiation Protection Program

- No changes from DCA

Acronyms

COL	Combined License
DCA	Design Certification Application
LCW	Low Conductivity Waste
NRC	Nuclear Regulatory Commission
RAI	Request for Additional Information



Chapter 18

Human Factors Engineering

August 22, 2024

Presenter: Doug Bowman

Introduction

- Chapter 18 Overview
  - Section 18.1 – Human Factor Engineering (HFE) Program Management
  - Section 18.2 – Operating Experience Review (OER)
  - **Section 18.3 – Functional Requirements Analysis and Function Allocation (FRA/FA)**
  - **Section 18.4 – Task Analysis (TA)**
  - **Section 18.5 – Staffing and Qualifications (S&Q)**
  - **Section 18.6 – Treatment of Important Human Actions (TIHAs)**
  - **Section 18.7 – Human-System Interface (HSI) Design**
  - Section 18.8 – Procedure Development
  - Section 18.9 – Training Program Development
  - Section 18.10 – Human Factors Verification and Validation (V&V)
  - Section 18.11 – Design Implementation (DI)
  - Section 18.12 – Human Performance Monitoring
- There are not slides for the areas that have not changed
- Other Items

Comparison of HFE Program for US600 DCA and US460 SDAA

Program Element	US600 DCA	US460 SDAA
Operating Experience Review	RSR Submitted	IP Submitted
Functional Requirements Analysis and Function Allocation	RSR Submitted	IP Submitted
Task Analysis	RSR Submitted	IP Submitted
Staffing and Qualifications	RSR Submitted	RSR Submitted
Treatment of Important Human Actions	RSR Submitted	RSR Submitted
Human-System Interface Design	RSR Submitted	IP Submitted
Procedure Development	COL Activity	COL Activity
Training Program Development	COL Activity	COL Activity
Verification and Validation	IP Submitted/RSR Submitted	IP Submitted
Design Implementation	COL Activity	IP Submitted
Human Performance Monitoring	COL Activity	COL Activity

- Implementation plan (IP) describes methodology
- Results summary report (RSR) describes methodology and results
- NUREG-0711 allows for submittal of an IP or RSR
- DCA: submitted RSRs for all HFE elements that are predecessors to V&V
- SDAA: NuScale is following the traditional model for HFE

Section 18.3 – Functional Requirements Analysis and Function Allocation

- The purpose of this element is to verify those functions needed to satisfy the plant's safety and commercial goals, and the assignment of those functions to personnel and automation, takes advantage of human and machine strengths and avoids human and machine limitations
- **As the FRA/FA process has matured, a single, combined and interlinked database has been developed that aligns the HFE task analysis, FRA/FA database, the Operator Training Task Analysis and the Plant Operating Procedure set**

Section 18.4 – Task Analysis

- TA identifies specific tasks (human actions) required to satisfy the functions from the FRA/FA element
- Similar process to US600 DCA
  - Now combined in a single, interlinked database as described in Section 18.3

Section 18.5 – Staffing and Qualifications

- S&Q determines the number and qualification of licensed operators required for safe and reliable plant operation
  - **Minimum staffing requirement is one licensed reactor operator and two licensed senior reactor operators**
- Changes from the DCA:
  - For the DCA: Minimum staffing requirements are located in the DC rule (Part 52 App. G)
  - **For the SDAA: Technical basis and approach for minimum staffing requirements is approved topical report → NuScale Control Room Staffing Plan, TR-0420-69456-NP-A**
  - Previously reviewed by ACRS (ML21139A226 and ML21139A232 [April 2021])

Section 18.6 – Treatment of Important Human Actions

- Identification of IHAs within the scope of Chapters 7,15, and 19
- Probabilistic Risk Assessment (Chapter 19) determines risk-important human actions
- Deterministic human actions are those credited in Chapter 15 and D3 (diversity and defense-in-depth) coping analyses of Chapter 7 (e.g., those required for long-term decay heat removal or reactivity control)
- US460 Standard Design does not have IHAs
- Changes from the DCA:
  - DCA included two IHAs → No longer RIHAs in the SDAA (see Chapter 19)
  - **The DCA IHAs are still addressed and mitigating strategies accounted for in the current SDAA generic technical guidelines, HSI design, HFE and training task analysis**

Section 18.7 – HSI Design

- HSI design element establishes the HSI design
- Substantially similar main control room and HSI as the US600
  - **US460 changes are in response to ISV and design changes from the US600**

Other Items

New HFE-related ITAAC:

02.	The MCR design incorporates HFE principles that reduce the potential for operator error.	An integrated system validation (ISV) test is performed in accordance with the Verification and Validation Implementation Plan.	A report exists and concludes that acceptance criteria associated with each ISV test scenario are satisfied upon initial performance of the scenarios or upon remediation of failures.

SDAA Audit

- 20 audit items successfully resolved
- Included virtual demonstration of simulator and staff review of databases

Acronyms

ACRS	Advisory Committee on Reactor Safeguards
COL	Combined License
D3	Diversity and Defense-in-depth
DCA	Design Certification Application
DI	Design Implementation
FRA/FA	Functional Requirements Analysis and Function Allocation
HSI	Human-System Interface
IHA	Important Human Action
IP	Implementation Plan
ISV	Integrated System Validation
MCR	Main Control Room
OER	Operating Experience Review
RIHA	Risk Important Human Action
RSR	Results Summary Report
S&Q	Staffing and Qualifications
SDAA	Standard Design Approval Application
TA	Task Analysis
TIHA	Treatment of Important Human Actions
V&V	Verification and Validation