NuScale Codes and Methods Framework Description Report

January 2013

Revision 0

Nonproprietary

NuScale Power, LLC

1100 NE Circle Blvd., Suite 350

Corvallis, Oregon 97330

www.nuscalepower.com

This page intentionally blank

PROPRIETARY INFORMATION NOTICE

THIS DOCUMENT DOES NOT CONTAIN PROPRIETARY INFORMATION.

COPYRIGHT NOTICE

CONTENTS

FIGURES

TABLES

FORWARD

This document is being submitted to the U.S. Nuclear Regulatory Commission (NRC) to facilitate discussions about the NuScale Power, LLC (NuScale) codes and methods framework during the pre-application review phase. This document provides a preliminary overview of the application of the codes and methods to be implemented by NuScale to evaluate anticipated operational occurrences (AOOs), postulated accidents (PAs), and perform core physics and thermal hydraulics analyses. A more detailed narrative is provided in the analysis of loss-of-coolant accident (LOCA) and rod ejection events. It is intended to provide an outline of the applicability of proprietary, special purpose, and existing codes and methods used extensively by the nuclear industry, with a focus on minor adaptations for analyzing the NuScale design. It is NuScale's intention to submit future pre-application reports to the NRC covering several of the topics discussed in this report.

NuScale is in the process of finalizing the NuScale power plant design and all of the design information presented in this report should be considered preliminary. Minor design changes are expected up until the submittal of the design certification application.

1.0 INTRODUCTION

This report presents the current selected codes and methods that will be used in the core design and safety analysis of the NuScale plant, which consists of one or more NuScale power modules. NuScale has selected codes that have been approved or are analogous to those approved by the NRC for conducting safety analyses of typical pressurized-water reactors (PWRs). Since evaluation of the suitability of codes and methods are typically event-specific, this report also presents a description and preliminary classification of the design basis events and plant conditions applicable to the NuScale module design. The goal of this report is three-fold:

1. Present a brief description of the codes identified thus far to perform core design and safety analyses.
2. Identify the applicability of the codes to transients and accidents identified for each analysis class.
3. Provide an overview of the methods implemented for the loss-of-coolant accident (LOCA) and rod ejection accidents.

The validation efforts employed to assess the adequacy of the code and its applications are discussed. This report does not identify specific acceptance criteria for the assessment of code-specific models.

Each major section of this report that discusses the use of a code outlines the application areas for analyses, regulatory requirements, the code chosen to perform the analyses, and a general description of code assessment efforts. A brief description of each section is given below.

Section 2.0 provides an overview of the NuScale module design and describes key design features of the plant to prevent and mitigate transients and accidents.

Section 3.0 provides an initial identification and classification of design basis events to be analyzed, along with a description of the applicability of each code.

Section 4.0 addresses the scope and application of the LOCA analyses to be performed with RELAP5 based on the design features of a NuScale module. Also addressed are the regulatory requirements and the application of the evaluation model development and assessment process (EMDAP) to ensure required features are included. The RELAP5 code is described based on its application domain along with modeling methods, code interfaces, and assessment efforts.

Section 5.0 discusses the NuScale in-house code, SubChannel Analyzer for NuScale Reactors (SCANR), to be used for subchannel analysis. The scope of the analyses and regulatory requirements is presented. A brief description of the code is given followed by methods used in analyses and interfaces with other codes. Lastly, the proposed assessment matrix and critical heat flux correlation development for SCANR are described.

Section 6.0 discusses the methods and codes used to perform an analysis of the neutronic behavior of the NuScale module. The codes chosen for these analyses are CASMO-5, CMSLINK, SIMULATE-5, and SIMULATE-3K. A brief description of each code is given. Lastly, the assessment efforts for the neutronics analysis codes are presented.

Section 7.0 presents the methods used to analyze a rod ejection accident with the regulatory requirements, the codes to be used (SIMULATE-3K, RELAP5, and SCANR), and an overview of the assessment efforts.

1.1 Abbreviations and Definitions

Table 1-1. Abbreviations

Term	Definition
AC	alternating current
AHF	actual heat flux
AIC	silver-indium-cadmium
AOO	anticipated operational occurrence
ARI	all-rods-in
ARO	all-rods-out
ASME	American Society of Mechanical Engineers
B&W	Babcock & Wilcox
BOC	beginning of cycle
BWR	boiling water reactor
CFR	code of federal regulations
CHF	critical heat flux
CMS	core management system
CNV	containment vessel
COBRA	coolant boiling in rod arrays
CRA	control rod assembly
CRC	commercial reactor critical
CVCS	chemical and volume control system
DC	design certification
DHRS	decay heat removal system
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DTC	Doppler temperature coefficient
EAB	exclusion area boundary
ECCS	emergency core cooling system
EFPD	effective full power day
EM	evaluation model
EMDAP	evaluation model development and assessment process
ENDF	evaluated nuclear data file
EOC	end-of-cycle
EPRI	Electric Power Research Institute
FTC	fuel temperature coefficient
FW	feedwater
GDC	general design criteria
GEST	Generator Separator Test
HCSG	helical coil steam generator
HFP	hot full power
HZP	hot zero power
IET	integral effects test
IHECSBE	International Handbook of Evaluated Criticality Safety Benchmark Experiments
IHERPBE	International Handbook of Evaluated Reactor Physics Benchmark Experiments

Term	Definition
ITC	isothermal temperature coefficient
LBLOCA	large-break loss-of-coolant accident
LEU	low enriched uranium
LOCA	loss-of-coolant accident
LPZ	low population zone
LWR	light-water reactor
MCNP	Monte Carlo N-Particle
MDC	moderator density coefficient
MDNBR	minimum departure from nucleate boiling ratio
MFW	main feedwater
MOC	middle of cycle
MTC	moderator temperature coefficient
NEA	Nuclear Energy Agency
NQA	nuclear quality assurance
NRC	U. S. Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OECD	Organisation for Economic Co-operation and Development
OSU	Oregon State University
PA	postulated accident
PCMI	pellet cladding mechanical interaction
PIRT	phenomena identification and ranking table
PNL	Pacific Northwest Laboratories
PWR	pressurized-water reactor
RCCA	rod cluster control assembly
RCP	reactor coolant pump
RCS	reactor coolant system
RELAP5	Reactor Excursion and Leak Analysis Program (version 5)
RG	regulatory guide
RPV	reactor pressure vessel
RRV	reactor recirculation valves
RVV	reactor vent valves
SAFDL	specified acceptable fuel design limit
SBLOCA	Small-break loss-of-coolant accident
SCANR	SubChannel Analyzer for NuScale Reactors
SDM	shutdown margin
SET	separate effects tests
SFCOMPO	Spent Fuel Composition Database (OECD)
SG	steam generator
SI	international system of units
SIET	Societa Italiana Esperienze Termoidrauliche
SPERT	Special Power Excursion Reactor Test
SRP	Standard Review Plan
TCA	tank-type critical assembly
TMI	Three Mile Island
VIPRE	Versatile Internals and Component Program for Reactors; EPRI

2.0 OVERVIEW OF NUSCALE MODULE DESIGN

The nuclear steam supply system (NSSS) module is a small, light-water-cooled, pressurized-water reactor (PWR). A NuScale power plant will provide customers with a scalable solution that allows a single facility to have up to 12 NSSS modules. The design takes advantage of existing design tools and available nuclear fuel options while leveraging the wealth of knowledge built through more than 50 years of practical application of light-water-cooled PWR technology.

The plant design provides a simple, highly reliable, and safe reactor that can be manufactured in the United States while maximizing the use of existing manufacturing capabilities and qualified "off-the-shelf" components.

The plant design meets these objectives by providing the following:

- Reliable and passively safe systems that are simple in design and operation
- Safety features that assure a core damage frequency significantly lower than the current light-water reactor fleet
- Reactor building designed to prevent penetration of a large commercial aircraft into the building while maintaining spent fuel pool integrity with the containment remaining intact, per 10 CFR 50.150
- 60-year design plant life
- Significantly reduced construction schedule compared to nuclear plants of a comparable output
- Modularization to enable in-shop fabrication of reactor and containment components
- Use of readily available conventional nuclear fuel

2.1 Nuclear Steam Supply System Module

The NSSS module is composed of a reactor core, a pressurizer, and two steam generators integrated within the reactor pressure vessel and housed in a compact steel containment vessel, as shown in Figure 2-1. The NSSS module's components are fabricated offsite and transported to the plant site.

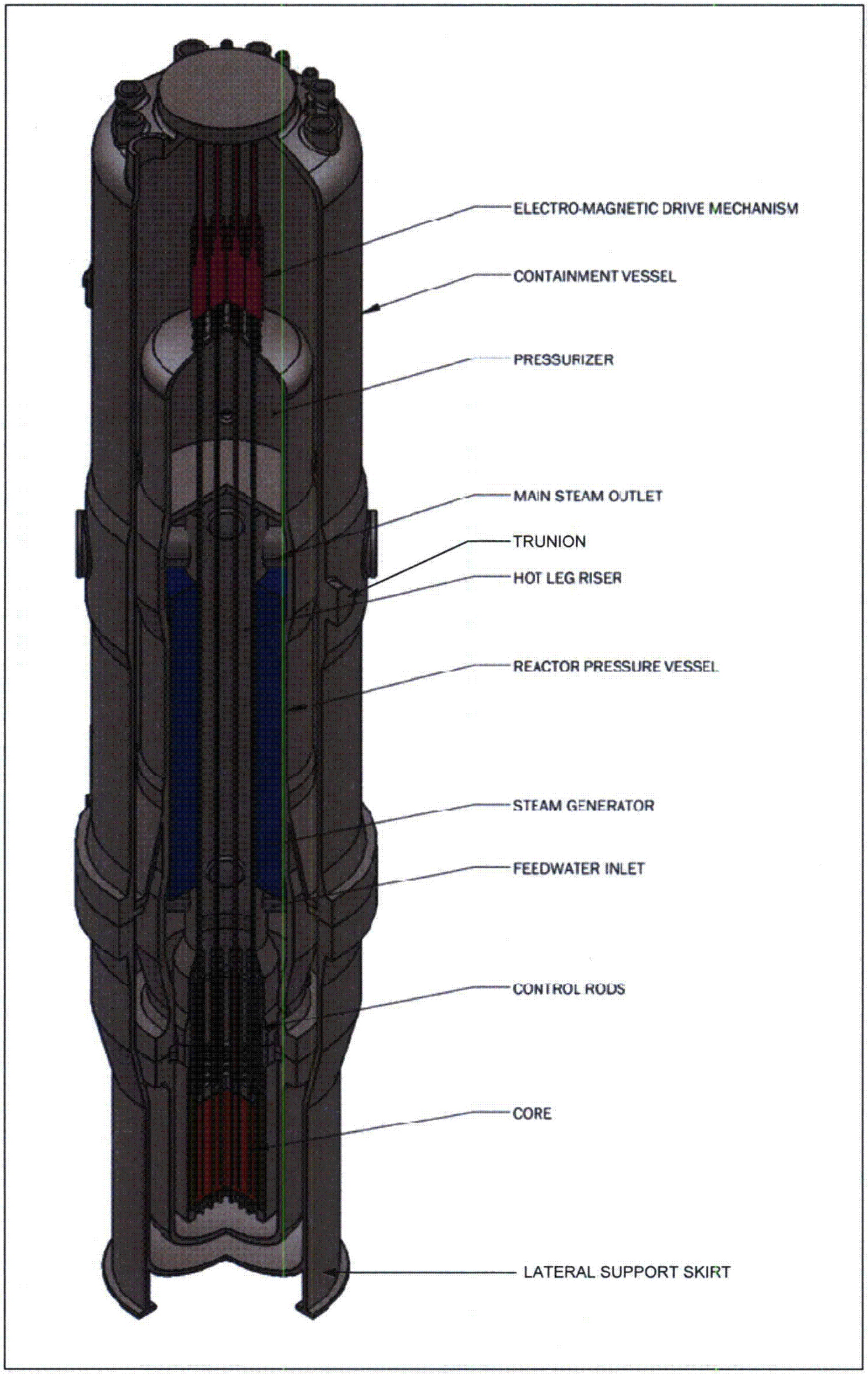


Figure 2-1. Cutaway view of the NuScale module

The NSSS module is designed to operate efficiently at full power conditions using natural circulation as the means of providing core coolant flow, eliminating the need for reactor coolant pumps. As shown in Figure 2-1, the reactor core is located inside a shroud connected to the hot leg riser. The reactor core heats reactor coolant causing the coolant to flow upward through the riser. When the heated reactor coolant exits the riser, it turns and flows downward over the tubes of the helical coil steam generator, which act as a heat sink. As the reactor coolant passes through the steam generator, it cools, increases in density, and naturally circulates downward to the reactor core inlet, where the cycle begins again.

NSSS modules are completely submerged in a reactor pool and protected by passive safety systems. Each NSSS module has a dedicated chemical and volume control system (CVCS), emergency core cooling system (ECCS), and decay heat removal system (DHRS).

Important features of the NSSS module include:

- A small, modular design that allows for incremental refueling outages and system maintenance
- An integral PWR NSSS that combines the reactor core, steam generator, and pressurizer within the reactor pressure vessel. Unlike a conventional PWR design, this design eliminates the external piping necessary to connect the steam generators and pressurizer to the reactor pressure vessel
- Buoyancy forces that drive natural circulation of the primary coolant, eliminating the need for reactor coolant pumps
- A reactor pressure vessel housed in a steel containment submerged in water. The water provides an effective passive heat sink for long-term emergency cooling
- The absence of reactor pressure vessel or containment penetrations below the top of the reactor core
- A steel containment operated at a deep vacuum eliminating the need for insulation on the reactor pressure vessel
- A modular design that can be built in a separate manufacturing facility, shipped to the site, and then installed, shortening the overall plant construction time resulting in significantly reduced costs for construction

A summary of the NuScale module design features are given in Table 2-1. The following subsections describe the major NuScale module components and systems.

Table 2-1. Features of the NuScale module

NuScale Design Feature	Primary Impact	Safety
Reactor coolant system integral to the reactor pressure vessel	No large primary coolant piping	Eliminates postulated large-break LOCA spectrum of accidents
Natural convection-cooled core	No reactor coolant pumps	Eliminates reactor coolant pump accidents, shaft breaks, pump seizure, missile generation and pump leaks
Once-through integrated steam generator with feedwater and steam inside the tube	Steam generator tubes are in compression	Improved steam generator tube integrity with steam generator tube rupture frequency reduced

NuScale Design Feature	Primary Impact	Safety
High design pressure and evacuated containment	All water lost from reactor vessel stays within containment and is returned to reactor vessel by passive means	No postulated design basis small-break LOCA capable of uncovering nuclear fuel
	Containment equilibrium pressure for worst case design basis accident remains below containment design pressure	Containment integrity assured (due to the metallic containment vessel, molten core concrete interaction is not possible).
	Subatmospheric pressure during normal operation	Increased steam condensation rates for containment heat removal during a postulated small-break LOCA. Any hydrogen postulated to be released is trapped in the containment vessel with little oxygen available to create a combustible mixture.
	No insulation on reactor vessel	Eliminates potential sump screen blockage and permits ex-vessel cooling
Low power core (160 MWt)	Reduces decay heat removal requirements	Enhances in-vessel retention; maintains low accident consequences; reduces fission product source term; simplifies emergency planning
Reactor pool with submerged NSSS and containment vessel	NSSS and containment vessel underwater	Provides passive long-term cooling and enhanced fission product retention
Passive safety systems	Safety systems cool and depressurize the containment vessel even in the event of loss of external power	Active safety systems are not required (low core damage frequency)

2.1.1 Reactor Pressure Vessel

The reactor pressure vessel consists of a steel cylinder with an inside diameter of approximately 9 ft and an overall height of approximately 60 ft and is designed for an operating pressure of approximately 1850 psig. To ensure safety and stability, ring forgings located at the steam and feed headers are thickened to provide reinforcement for the vessel nozzles. The upper and lower heads are elliptical, and the lower portion of the vessel has flanges to provide access for refueling (Reference 8.1.1).

The top of the upper head of the reactor pressure vessel provides support for the control rod drive mechanisms. Nozzles on the upper head provide connections for relief valves, reactor vent valves, and the pressurizer spray piping.

To provide a barrier between the saturated water in the pressurizer and the reactor coolant system fluid, a steel divider plate is located above the steam generator. The divider plate has orifices to limit the in and out surge of water in the pressurizer and to act as a thermal barrier.

2.1.2 Steam Generator

Each NSSS module uses two once-through helical-coil steam generators for steam production. The steam generators are located in the annular space between the hot leg riser and the reactor pressure vessel wall. The steam generator consists of tubes connected to upper and lower

plenums with tubesheets. Preheated feedwater enters the lower steam generator plenum through nozzles on the reactor pressure vessel (see Figure 2-2). As feedwater rises through the interior of the steam generator tubes, heat is added from the reactor coolant and the feedwater boils and exits the steam generator as superheated steam.

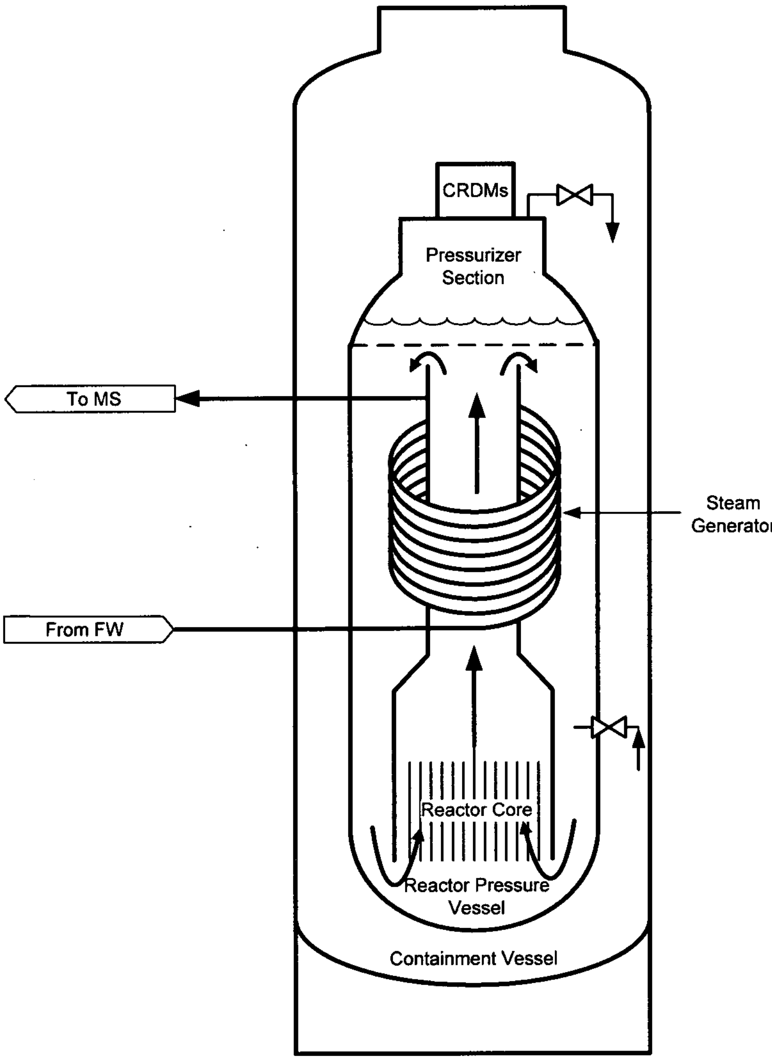


Figure 2-2. Steam generator and reactor flow

2.1.3 Pressurizer

The pressurizer is integrated into the reactor pressure vessel (RPV) and provides the primary means for controlling reactor coolant system pressure. It is designed to maintain a constant pressure during operation. Reactor coolant pressure is increased by applying power to a system of pressurizer heaters in the reactor pressure vessel head. The heater penetrations are installed above the pressurizer separator plate. Pressure in the reactor coolant system is reduced using spray provided by the CVCS.

2.1.4 Core and Fuel Assembly Design

The NuScale reactor design features a 37-fuel assembly core composed of conventional 17x17 square lattice array fuel assemblies. The core includes sixteen control rod assemblies (CRAs). Each fuel assembly includes 264 fuel pins, 24 control rods, and one instrument tube. The reactor design is provided in Figure 2-3. A table of preliminary reactor core design information is provided in Table 2-2 and Table 2-3 (Reference 8.1.2).

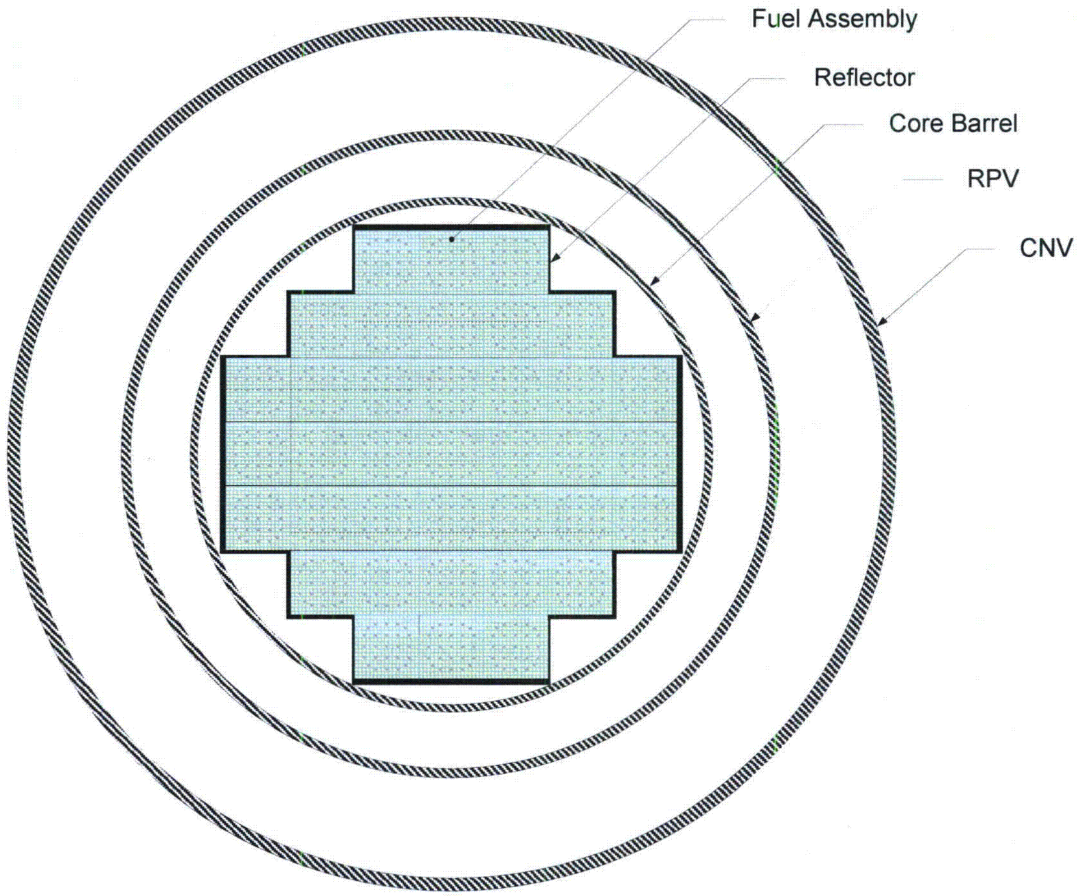


Figure 2-3. Radial cross-section view of the preliminary NuScale reactor module design

Table 2-2. Preliminary NuScale reactor core design characteristics

Design Parameters and Characteristics	Value	Units
Thermal power	160	MW_t
Number of fuel assemblies	37	-
Number of fuel pins per assembly	264	-
[[		

]]3(b)

Table 2-3. Preliminary NuScale fuel assembly and control rod assembly characteristics

Core Parameters	Dimensions
Fuel Pins (Standard 17 x 17 PWR Enriched UO_2 Fuel with Zircaloy Cladding)	
Rod outside diameter	0.374 in
Pellet outside diameter	0.322 in
Active height	78.74 in
Fuel Assembly (17x 17 Square Array)	
Assembly pitch	8.466 in
Pin pitch	0.496 in
Control Rods (B_4C Absorber)	
Absorber material diameter	0.339 in
Control rod outside diameter	0.378 in

2.1.4.1 Reactivity Control

Reactivity control is achieved using soluble boron and control rod assemblies.

2.1.4.1.1 Soluble Boron

The NuScale reactor core is cooled and moderated by light water primary coolant containing natural boron. Boron is a strong absorber of thermal neutrons and is dissolved in the primary coolant for reactivity control. The soluble boron is used throughout the cycle to compensate for the excess reactivity of the fresh fuel loaded into the core at the beginning of the cycle and is adjusted throughout the cycle to control slow reactivity changes resulting from burn-up effects.

The soluble boron is also adjusted to compensate for large reactivity changes during reactor heatup and cooldown.

2.1.4.1.2 Control Rod Assemblies

The NuScale reactor design incorporates a total of 16 identical CRAs for providing rapid shutdown and power distribution control. Each CRA spans only one fuel assembly and consists of 24 individual absorber rods integrated in a single spider assembly. The absorber rods are composed of silver-indium-cadmium absorber material, extending the full length of the core. Stainless steel tubes encapsulate the absorber material, isolating it from the reactor coolant.

The CRAs are used for reactor startup and shutdown, following load changes, and controlling small transient changes in reactivity. The CRAs can shut the reactor down at all times, even with the most reactive rod stuck out of the core.

The CRA's are grouped to form two control rod banks. Control group 1 consists of the inner four CRAs and is used for core reactivity regulation. Control group 2 includes the outer 12 CRAs and is used for reactor shutdown. All of the CRAs in control groups 1 and 2 are composed of the same materials and axial composition. The core layout showing the preliminary positions of the CRAs is provided in Figure 2-4.

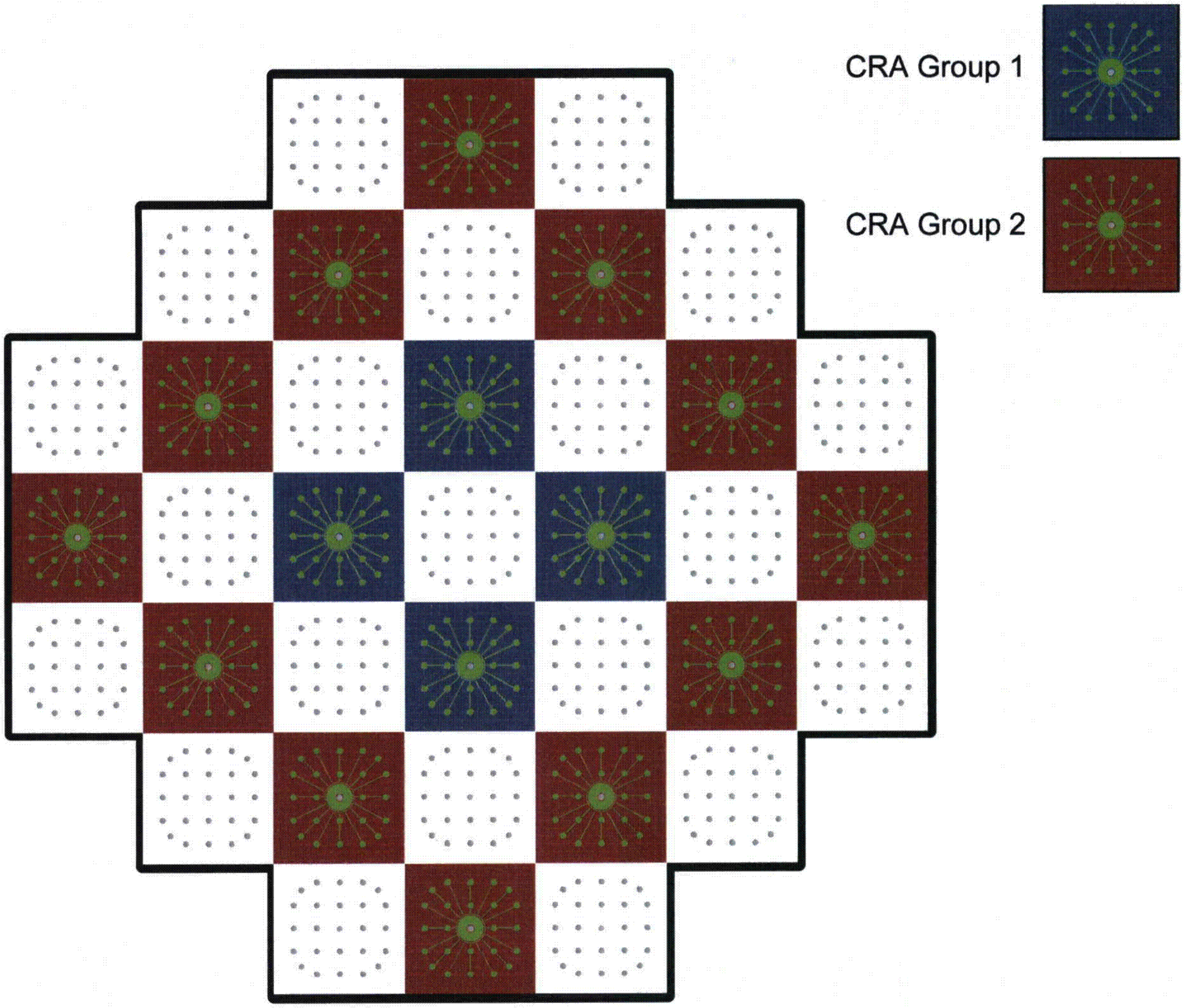


Figure 2-4. Preliminary control rod assembly map for the NuScale power reactor

2.1.4.2 Fuel Assembly Design

The NuScale reactor design uses a total of 37 fuel assemblies. Each fuel assembly is comprised of a conventional 17×17 square lattice array featuring a total of 264 fuel pins, 24 guide tubes for control absorber rods, and one instrument tube.

The fuel assembly design will utilize axial zoning. The purpose of the axial zoning is to maximize neutron economy and control the axial power shape and peaking. This is achieved by using reduced enrichment axial blankets that are on the top and bottom of the fuel stack. This lower enrichment creates an effective reflector for the neutrons within the core, reducing the axial leakage. The axial enrichment zones employed in the fuel assembly design are defined in Figure 2-5. They are illustrated as regions superimposed over the fuel assembly.

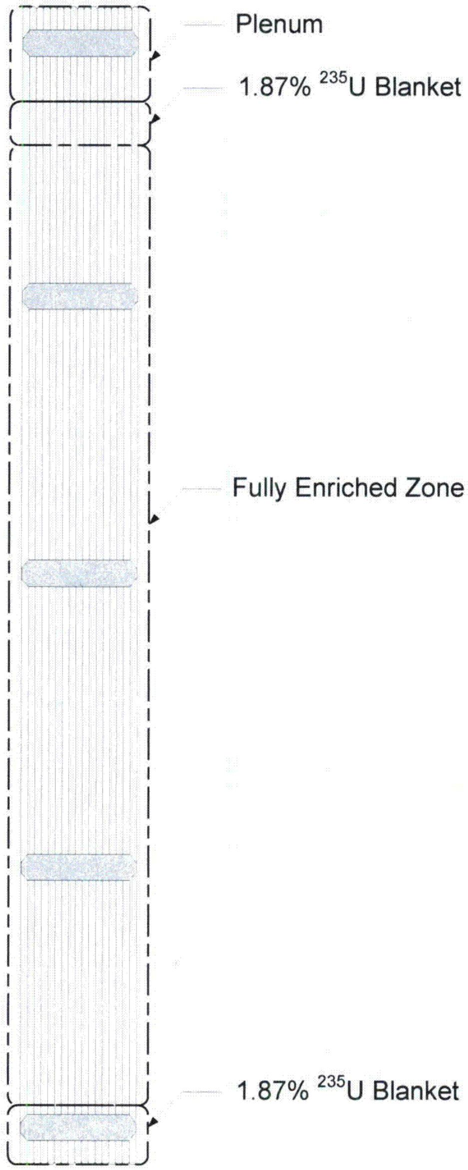


Figure 2-5. Fuel assembly axial composition

2.1.5 Chemical and Volume Control System

The CVCS is simple in design and is not required to function during or after an accident. During normal operation, the CVCS recirculates a portion of the reactor coolant through demineralizers to maintain reactor coolant cleanliness and chemistry. Reactor coolant inventory is controlled by injection of additional water when the pressurizer level is low or let down of reactor coolant to the liquid radioactive waste system when coolant inventory is high.

Boron concentration in the reactor coolant system is controlled by a feed and bleed process. Injection pumps provide borated water or clean demineralized water that is delivered into the reactor vessel with excess reactor coolant being letdown to the radioactive waste system.

2.2 Safety Features

Each NSSS module incorporates several simple, redundant, and independent safety features. These features are discussed in detail in the following sections.

2.2.1 Containment Vessel

The major safety functions of the containment vessel are to contain the release of radioactivity following postulated accidents, to protect the reactor pressure vessel and its contents from external hazards, and to provide an interfacing medium (reactor vessel to water, to containment vessel, to the pool) for decay heat removal following an accident or normal reactor shutdown.

Each containment vessel consists of a steel cylinder with an outside diameter of approximately 15 ft and an overall height of [[ ]][3(b)]. The containment vessel houses the reactor pressure vessel, control rod drive mechanisms, and associated components (Reference 8.1.1).

Flanges are provided on the containment vessel to allow for disassembly and to allow access to the reactor pressure vessel during refueling operations and maintenance. Manways are located in the upper head and in the vessel circumference to allow access to the steam generator headers during refueling outages. Penetrations located on the vessel upper head provide access for process piping to the reactor pressure vessel and to the containment vessel interior. Additional penetrations are provided for electrical and instrumentation connections.

The vessel is vertically and laterally supported by connection to the reactor pool walls. A support skirt attached to the containment vessel lower head allows the vessel to be supported laterally. Internal to the containment, the reactor vessel is laterally and vertically supported by connections to the containment vessel wall.

The containment vessel is submerged in the reactor pool, which provides a passive heat sink for the containment heat removal under LOCA conditions. Although not credited, the reactor pool provides an additional means of fission product retention beyond that of the fuel, fuel cladding, reactor pressure vessel, and the containment vessel for certain events. The containment vessel is designed to withstand the environment of the reactor pool as well as the high pressure and temperature of any design basis accident.

The containment vessel pressure is maintained at a deep vacuum under normal operating conditions. Maintaining a deep vacuum provides for reduced moisture that could contribute to component corrosion and impact the reliability of instrumentation and other systems within the containment vessel. The deep vacuum essentially eliminates convection heat transfer removing the need for "direct-contact" reactor pressure vessel insulation. Due to a lack of appreciable amounts of air, the deep vacuum also enhances steam condensation rates that would occur during an accident with ECCS actuation and would limit the formation of a combustible mixture of hydrogen and oxygen during a severe accident.

Following an actuation of the ECCS, heat removal through the containment vessel rapidly reduces the containment pressure and temperature and maintains them at less than design conditions. Steam is condensed on the inside surface of the containment vessel, which is passively cooled by conduction and convection of heat to the reactor pool water. Because the containment vessel is evacuated to a low absolute pressure during normal operation, few non-condensable gasses are present inside the containment vessel. This is beneficial, because the presence of non-condensable gasses has a tendency to reduce condensation heat transfer rates.

2.2.2 Decay Heat Removal System

The decay heat removal system (DHRS) provides secondary side reactor cooling when normal feedwater is not available. The system, as shown in Figure 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 steam generator loop. Each train is independently capable of removing 100 percent of the decay heat load and cooling the reactor coolant system. Each train has a passive condenser submerged in the reactor pool. The condensers are maintained with sufficient water inventory for stable operation.

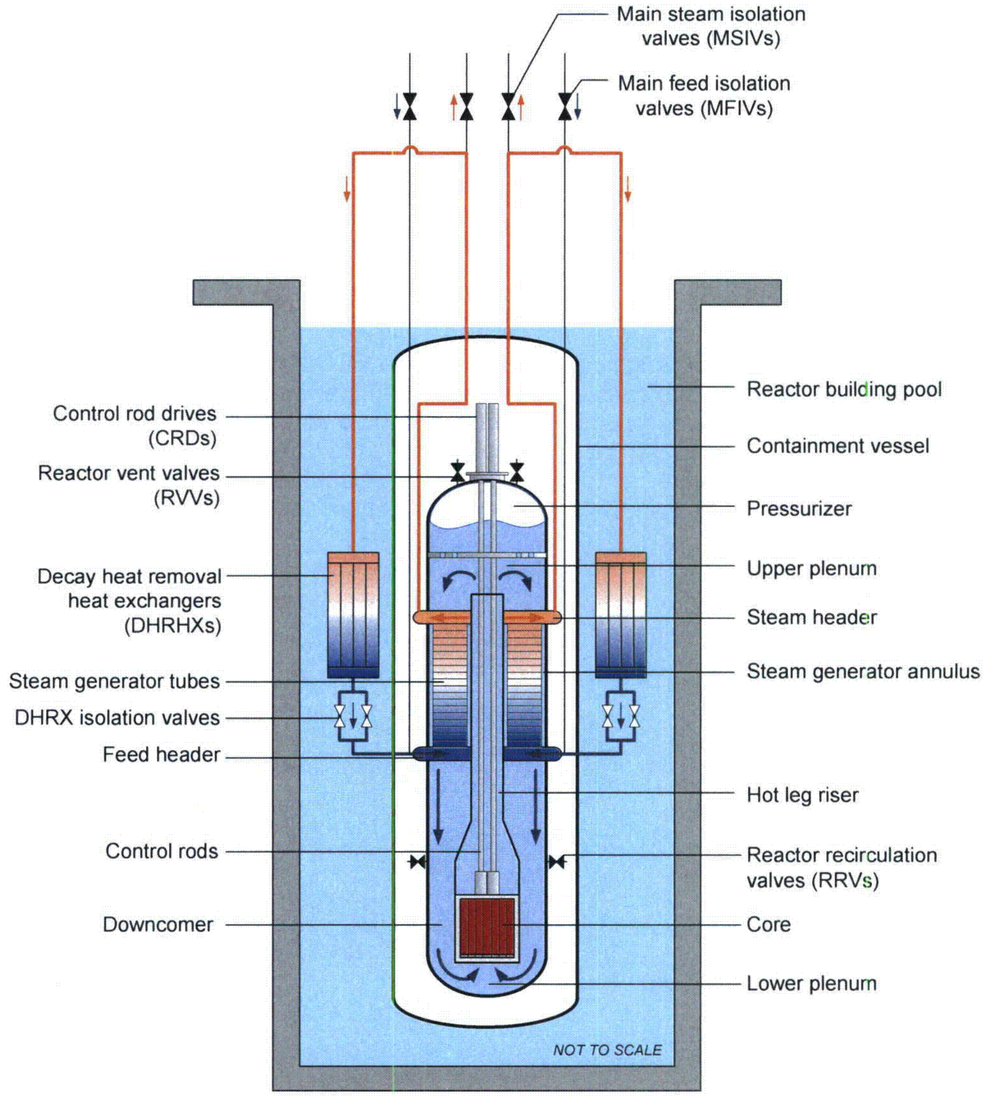


Figure 2-6. Decay heat removal system schematic

Upon receipt of an actuation signal, the DHRS valves open. This allows water from the decay heat removal condensers to flow into the steam generators and cool the reactor coolant as it boils. The steam then travels through the steam line back to the decay heat removal condenser where it is condensed by the reactor pool water, and the cycle is repeated. Heat is removed via the steam generators, thus preserving natural circulation within the reactor coolant system.

2.2.3 Emergency Core Cooling System

As shown in Figure 2-7, the emergency core cooling system (ECCS) consists of two independent reactor vent valves and two independent reactor recirculation valves. The ECCS provides a means of decay heat removal in the event of a loss of coolant accident or a loss of the main feedwater flow in conjunction with the loss of both trains of the DHRS.

The ECCS removes heat and limits containment pressure by steam condensation on, and convective heat transfer to, the inside surface of the containment vessel. It allows heat conduction through the containment vessel walls and heat conduction and convection to the water in the reactor pool. Long-term cooling is established via recirculation of reactor coolant to the reactor pressure vessel via the ECCS recirculation valves, which when opened provide a return flow of cooled water to the reactor.

The ECCS is initiated by opening the two (2) reactor vent valves in lines exiting the top of the reactor pressure vessel (the pressurizer region) and the two (2) reactor recirculation valves on lines entering the reactor pressure vessel in the downcomer region at a height above the core. Opening the valves allows a natural circulation path to be established. Water that is vaporized in the core leaves as steam through the reactor vent valves, is condensed and collected in the containment vessel, and is then returned to the downcomer region inside the reactor vessel through the reactor recirculation valves.

Following a LOCA or other condition resulting in an actuation of the ECCS, heat removal through the containment vessel rapidly reduces the containment pressure and temperature and maintains them at acceptably low levels for extended periods of time. Steam is condensed on the inside surface of the containment vessel, which is passively cooled by conduction and convection of heat to the reactor pool water. Since the containment vessel is evacuated to a low absolute pressure during normal operation, only a small amount of non-condensable gas will be present inside the containment vessel.

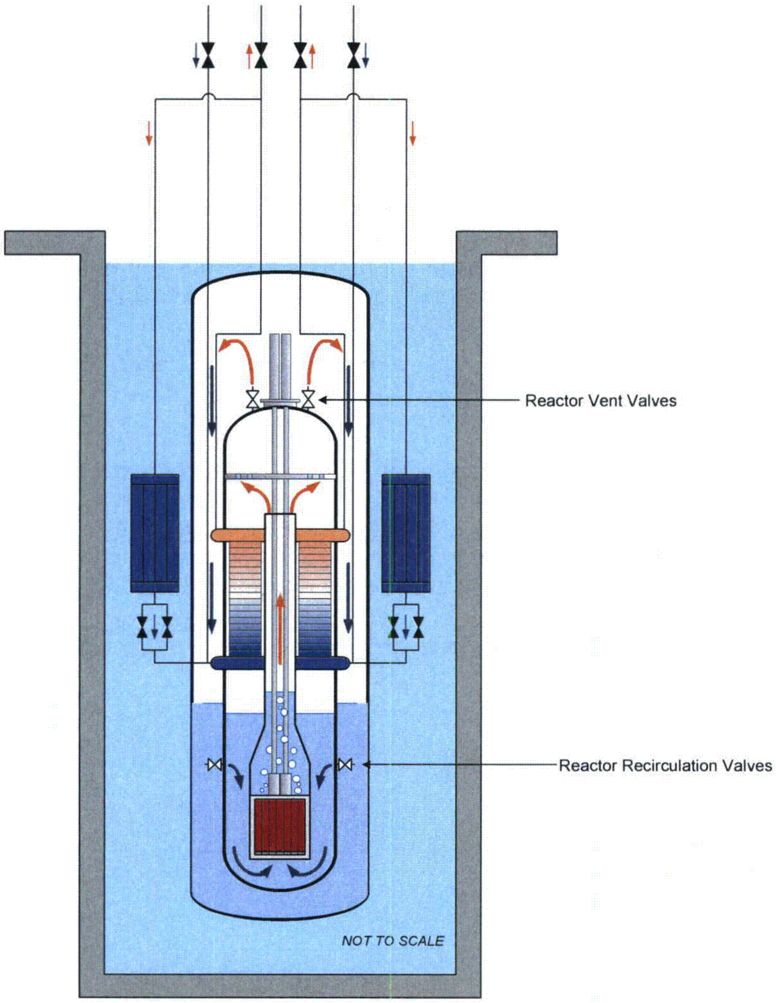


Figure 2-7. Emergency core cooling and containment heat removal system schematic

2.2.4 Reactor Pool

The reactor pool consists of a large, below-grade concrete pool with a stainless steel liner that provides stable cooling for the containment vessel for a minimum of 72 hours following any LOCA (Reference 8.1.1). During normal plant operations, heat is removed from the pool through a closed loop cooling system and ultimately rejected into the atmosphere through a cooling tower or other external heat sink. In an accident where offsite power is lost, heat is removed from the reactors and containments by allowing the pool to heat up and boil. Water inventory in the reactor pool is large enough to cool the reactors for at least 72 hours without adding water. After 72 hours, reactor building pool water boil-off, and ultimately passive air cooling of the containment vessels, provide adequate cooling for long-term decay heat removal.

3.0 NUSCALE MODULE TRANSIENT AND ACCIDENT IDENTIFICATION FOR NON-LOCAS

3.1 Scope

Normal operation includes plant heatup and cooldown, power level variation, and steady state operation. Anticipated operational occurrence (AOOs) are events in which the reactor plant conditions are disturbed from normal operation and are expected to occur one or more times during the lifetime of the plant. Postulated accidents (PAs) are unanticipated events that are not expected to occur over the life of the plant. This section identifies a preliminary list of design basis transients and accidents applicable to the NuScale design. Codes will be selected and methodologies will be developed to analyze these events.

3.2 Introduction

Initiating events are classified by event type and frequency of occurrence. In accordance with Chapter 15.0 of the Standard Review Plan (SRP) (Reference 8.1.3), the classification of an event is defined as either an AOO or a PA. NuScale has adopted this categorization approach such that AOOs are conditions of normal operation that are expected to occur one or more times during the life of the plant, and PAs are occurrences that are postulated, but are not expected to occur.

AOOs and PAs can be further sub-defined into seven types of events. These events are assigned to the following categories:

1. Increase in heat removal by the secondary system
2. Decrease in heat removal by the secondary system
3. Decrease in reactor coolant system (RCS) flow rate
4. Reactivity and power distribution anomalies
5. Increase in reactor coolant inventory
6. Decrease in reactor coolant inventory
7. Radioactive release from a subsystem or component

A key feature of the NuScale design that leads to differences in initiating events is that the NuScale module does not rely on pumps to circulate reactor coolant during normal operation, an AOO, or a PA. Therefore, the AOOs and PAs dealing with failure of reactor coolant pumps are not applicable to analyses of the NuScale design. Steam generator cooling and core heat addition provide the driving head for natural circulation flow in the reactor coolant system. As such, changes in the secondary side (such as an increase or decrease in steam generator heat removal) or a change in core power can have a direct corresponding effect on reactor coolant system flow. Changes in reactor coolant system flow will be included in the analyses of those events.

3.3 Regulatory Requirements and Guidance

During a transient caused by an AOO, the reactor core must be undamaged and able to return to normal operation as required by Chapter 15.0 of the SRP (Reference 8.1.3). Acceptance criteria generally applied to AOOs are given in Table 3-1.

Table 3-1. Summary of regulatory guidelines pertaining to non-LOCA AOO analysis

SRP	Criterion
15.0	**Primary Pressure Limits** Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.
	Fuel Cladding Integrity Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit, and that the centerline fuel melt limit is not exceeded.
	Event Evolution An AOO should not generate a postulated accident without other faults occurring independently or result in a consequential loss of function of the RCS or reactor containment barriers.

During a transient caused by a PA, the system could sustain damage to preclude resumption of plant operation. General acceptance criteria generally applied to Non-LOCA PAs are given in Table 3-2 as outlined by Chapter 15 of the SRP.

Table 3-2. Summary of regulatory guidelines pertaining to non-LOCA PA analysis

SRP	Criterion
15.0	**Primary Pressure Limits** Pressure in the RCS and main steam system should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.
	Fuel Cladding Integrity Fuel cladding integrity will be maintained if the minimum DNBR remains above the 95/95 DNBR limit. If this limit is exceeded then the fuel cladding is assumed to have failed as an input to the radiological consequence analysis.
	Fuel Peak Temperatures Calculations will demonstrate that peak fuel temperatures are below melting conditions. If this limit is exceeded then the fuel cladding is assumed to have failed as an input to the radiological consequence analysis.
	Radioactive Release The release of radioactive material shall not result in offsite doses in excess of the guidelines of 10 CFR Part 100.
	Event Evolution A postulated accident shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.

3.4 Codes Identified for Non-LOCA Transient and Accident Analyses

The tables in the following subsections list the identified AOOs and PAs to be evaluated as part of the NuScale safety analyses. These tables provide listings of the primary codes to be used for predicting compliance to specified acceptance criteria based upon the event type for each AOO or PA. The codes that provide input to the analysis codes are omitted from these tables and only the primary codes(s) are listed. The RELAP5 and SCANR code versions are proprietary NuScale codes that will be based on modifications to existing, well-established codes that are used in the nuclear-safety industry.

3.4.1 Increase in Heat Removal

Event	Category	Computer Code(s)
Reduction in feedwater temperature	AOO	[[
Increase in feedwater flow	AOO	
Increase in steam flow	AOO	
Inadvertent opening of a steam generator relief or safety valve	AOO	
Steam system piping failure (minor)	AOO	
Steam system piping failure (major)	PA	
Inadvertent actuation of decay heat removal system (DHRS)	AOO	
Containment flooding	AOO	
Loss of containment vacuum	AOO	]]3(a)

3.4.2 Decrease in Heat Removal

Event	Category	Computer Code(s)
Loss of external load	AOO	[[
Turbine trip	AOO	
Loss of condenser vacuum	AOO	
Closure of main steam isolation valves	AOO	
Loss of non-emergency AC power to station auxiliaries	AOO	
Loss of normal feedwater flow	AOO	
Feedwater system pipe break (minor)	AOO	
Feedwater system pipe break (major)	PA	]]3(a)

3.4.3 Decrease in Reactor Coolant System Flow

The NuScale module does not rely on pumps to circulate reactor coolant during normal operation, an AOO, or a PA. Therefore, the postulated accidents or scenarios dealing with failure of reactor coolant pumps are not applicable.

3.4.4 Reactivity and Power Distribution Anomalies

Event	Category	Computer Code(s)
Uncontrolled control rod assembly withdrawal from a subcritical or low power condition	AOO	[[
Uncontrolled control rod assembly withdrawal at power	AOO	
Control rod misoperation	AOO/PA	
Startup of inactive loop or recirculation loop at an incorrect temperature	AOO	
Inadvertent decrease in boron concentration in RCS	AOO	
Inadvertent loading and operation of a fuel assembly in an improper position	PA	
Inadvertent moderator cooldown	AOO	
Rod ejection	PA	
		]]3(a)

3.4.5 Increase in Reactor Coolant System Inventory

Event	Category	Computer Code(s)
Chemical and volume control system (CVCS) malfunction that increases reactor coolant inventory	AOO	[[ ]]3(a)

3.4.6 Decrease in Reactor Coolant System Inventory

Event	Category	Computer Code(s)
Inadvertent opening of pressurizer pressure relief valve	AOO	[[
Steam generator tube failure	AOO	
Inadvertent operation of emergency core cooling system (ECCS) that decreases reactor coolant inventory	AOO	]]3(a)

4.0 SYSTEMS THERMAL-HYDRAULIC LOCA ANALYSIS

4.1 Scope

RELAP5 has been chosen by NuScale for the small-break loss-of-coolant accident (SBLOCA) analysis. This section discusses the application of the LOCA analysis methodology to be performed with RELAP5 based on the design features of a NuScale module.

4.2 Introduction

RELAP5 is an eight field equation, multi-phase, and non-equilibrium system thermal-hydraulics code (Reference 8.1.4). The code contains additional field equations for the transport of non-condensable gases and solids transport, such as boron.

RELAP5 contains one and two-dimensional heat structures for radial and axial heat conduction through fuel pins, plates, steam generator tubes, and pipe and vessel walls. It also contains an extensive boiling curve with convective heat transfer correlations (Reference 8.1.4). The code has variable and Boolean (logical) trips, a control system that can be coupled to the independent variables, and control and trip variables for modeling all plant components and their interactions.

For loss-of-coolant (LOCA) analyses the Evaluation Model Development and Assessment Process (EMDAP) as defined by Regulatory Guide 1.203 will be followed (Reference 8.1.5).

The NuScale Integral Effects Tests (IET) at Oregon State University (OSU), The Societa Italiana Esperienze Termoidrauliche (SIET) Helical Coil Steam Generator Test Facility, and the SIET Generator Separator Test (GEST) Facility play important roles in the assessment of codes used for the analysis of NuScale LOCA events. These are discussed in Section 4.5.2.

4.3 Regulatory Requirements and Guidance

As part of the safety analysis of the NuScale reactor module, an analysis and evaluation of the emergency core cooling system (ECCS) performance following a postulated LOCA will be performed in accordance with the applicable requirements of 10 CFR 50, Appendix K for LOCA events. The acceptance criteria of 10 CFR 50.46 are listed in Table 4-1 below.

Table 4-1. Summary of acceptance criteria pertaining to loss-of-coolant analysis

SRP	Criterion
15.0	**Peak Cladding Temperature** The calculated maximum fuel element cladding temperature shall not exceed 2200 °F.
	Maximum Cladding Oxidation The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. *(additional language deleted for brevity)*
	Maximum Hydrogen Generation The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
	Coolable Geometry Calculated changes in core geometry shall be such that the core remains amenable to cooling.
	Long-Term Cooling After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

For LOCA analyses the EMDAP as defined by Regulatory Guide 1.203 will be followed (Reference 8.1.5). As part of the EMDAP, the evaluation model structure and codes will be able to model system and plant components along with the phenomena applicable for a LOCA. Since the plant behavior is not equally influenced by all its components during a transient for all phenomena, a Phenomena Identification and Ranking Table (PIRT) was generated as part of step four of the EMDAP which identifies the high ranked phenomenon from the SBLOCA PIRT.

4.4 LOCA Analysis Methods

The EM for analyzing LOCAs for the NuScale module is currently under development. The EM for LOCAs will use the RELAP5 computer code for analysis of the primary and secondary coolant system as well as the ECCS and the containment response. The system will be characterized for the entire spectrum of potential break sizes and locations. The LOCA analysis methodology is described in the subsequent subsections followed by a description of the application of the EMDAP in Section 4.5.

4.4.1 Loss-of-Coolant Accident Break Size Consideration

For the NuScale module design, all primary components are integrated into the reactor vessel. This makes consideration of the upper limit of break size in the NuScale module, as defined by 10 CFR 50.46(c), applicable to relatively small pipes. Thus, the design basis event for the NuScale reactor system will be a SBLOCA.

4.4.2 Small-Break Loss-of-Coolant Accident Location

The SBLOCA EM will also consider break location in addition to break-size spectrum. [[

]]3(b)

4.4.3 Passive Safety System Performance

A spectrum of breaks related to the failure of the RVVs, RRVs, primary safety valves, and CVCS connections will be analyzed to determine the limiting break location and size for the NuScale module. It will be demonstrated that events are adequately mitigated by the ECCS and associated equipment. Actuation of the ECCS is completely automated by the reactor protection system and requires no operator action or AC power. Nonetheless, operator actions that could potentially affect ECCS performance will be evaluated.

Scaled integral testing will be carried out to confirm passive safety system performance and to validate RELAP5 modeling. The initial list of integral effects tests that will be used for validation are shown in Table 4-6.

4.5 Loss-of-Coolant Accident Evaluation Model

Development of the NuScale LOCA EM will conform to the guidance provided in Regulatory Guide 1.203, which describes a process acceptable to the NRC for developing and accessing EMs used for analysis of design based transient and accident behavior of a nuclear power plant. An overview of the EMDAP is shown in Figure 4-1.

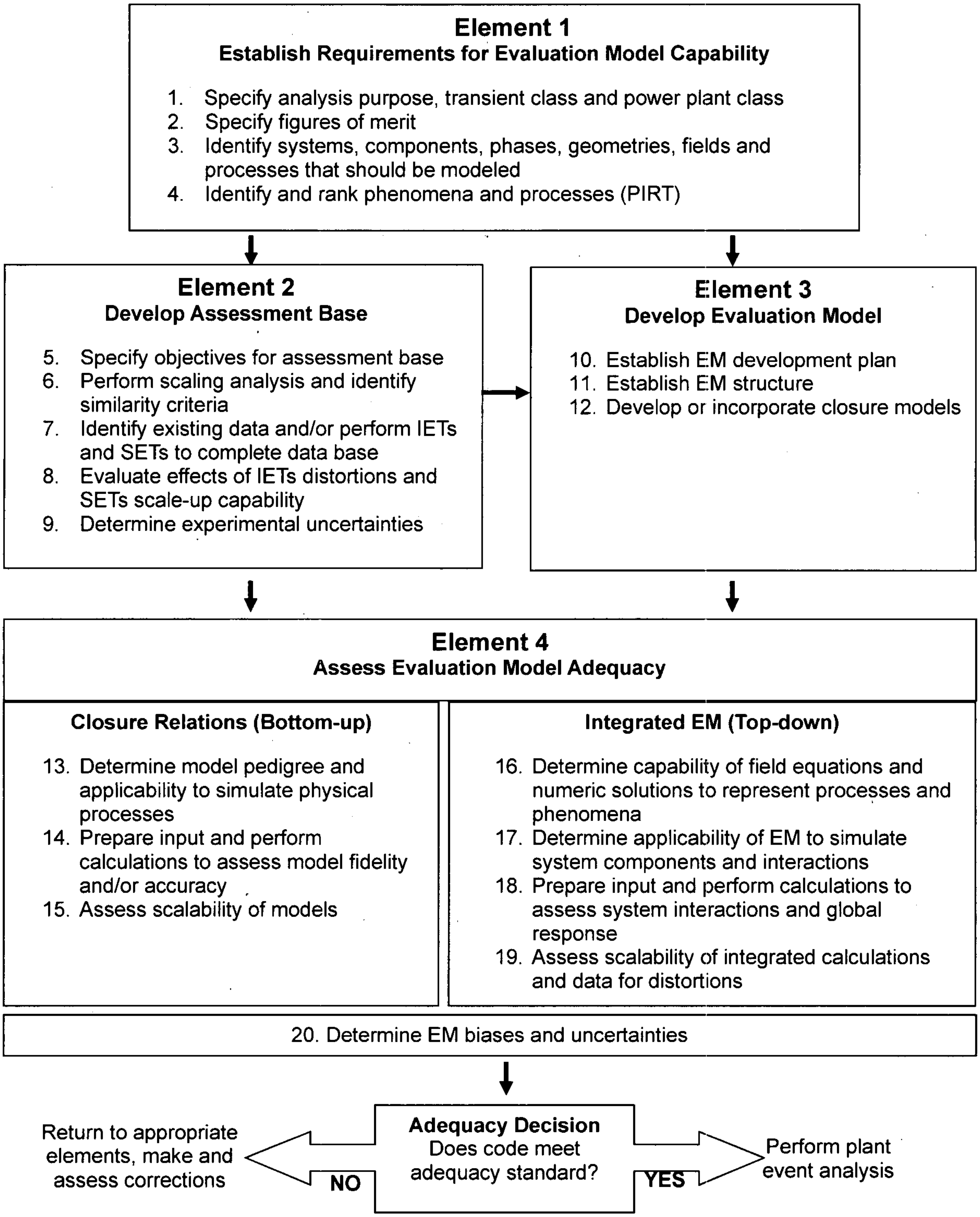


Figure 4-1. The EMDAP following Regulatory Guide 1.203

The arrows in Figure 4-1 show paths of iteration as the LOCA EM is developed. The first step in the process will be to compare the existing RELAP5 base code to available benchmark data in order to identify potential model deficiencies as applied to the NuScale design. Two PIRTs have thus far been developed (Reference 8.1.6) and will aid in identifying phenomenon of importance for analysis. After this initial evaluation, the base code will be taken under configuration management control and modified to include Appendix K requirements, heat transfer correlations applicable to the NuScale steam generator design and any other necessary modifications. This new code version will be "frozen" and benchmarked against the NuScale ECCS EM validation matrix. Results from the benchmarking will be evaluated and modeling or numerical solution improvements will be made, as appropriate. The modified code will then be benchmarked again. This process of model/code improvement will be repeated until the code and model performance is suitable for the intended purpose.

The following sections provide a more detailed discussion of Elements 1 through 4 of the EMDAP for the NuScale LOCA EM.

4.5.1 Element 1 – Establish Requirements for Evaluation Model Capability

Step 1 – Specify Analysis Purpose, Transient Class, and Power Plant Class

The first step in determining the requirements for the NuScale LOCA EM capability includes specifying the analysis purpose, identifying the transient class, and identifying the plant class that is to be analyzed. This step defines how and where the EM will be applied. The NuScale LOCA EM will conform to the following:

- *Analysis Purpose* - Demonstrate compliance with ECCS performance acceptance criteria 10 CFR 50.46.
- *Analysis Type* - Deterministic approach by implementation of modeling requirements of 10 CFR 50, Appendix K.
- *Transient Class* - SBLOCA (full break area spectrum and break location considered, as appropriate for the unique NuScale reactor system).
- *Plant Class* - Integrated NuScale pressurized-water reactor module.

Step 2 – Specify Figures of Merit

The purpose of the LOCA EM is to demonstrate compliance with the ECCS performance acceptance criteria listed in Section 4.3. Figures of merit were identified by the expert panels in the development of the two PIRTs (Reference 8.1.6). The figures of merit for each transient phase identified in the latest PIRT are described in Table 4-2 for two scenarios: (1) the inadvertent actuation of one of the RVV and (2) the inadvertent actuation of one of the RRVs. It is noted that both PIRTS are considered relevant to the NuScale design and further PIRTs may be developed if necessary.

Table 4-2. Figures of merit for the LOCA evaluation model

[[

Period	Figure of Merit	Description

]]3(b)

[[

Period	Figure of Merit	Description

]]3(b)

Step 3 – Identify Systems, Components, Phases, Geometries, Fields, and Processes that Should be Modeled

The purpose of this step is to identify the EM characteristics. It requires a detailed decomposition of the plant system into the system constituents. A preliminary decomposition for the NuScale SBLOCA EM is provided below:

1. System: containment vessel, reactor pressure vessel, decay heat removal system, and the reactor building pool.
2. Subsystems/Components: Table 4-3 shows the decomposition of the four plant system constituents into sub-systems, and components.

Table 4-3. Phenomena-specific systems, subsystems, and components

[[

Containment Vessel	Reactor Pressure Vessel	Decay Heat Removal System	Reactor Building
•			

]]3(b)

Step 4 – Identify and Rank Key Phenomena and Processes

The panel divided the NuScale module into systems, components, and processes as described previously in Step 3. For each of these systems and their subcomponents, the panel identified phenomena that may be expected to occur in the NuScale module during an SBLOCA. After the panel identified phenomena, they ranked each phenomenon relative to one or more of the figures of merit (e.g., core mixture level). Each phenomenon was assigned an importance rank of "High,"

"Medium," "Low," or "Inactive." In addition, panel members assessed and ranked the current knowledge level for the high-ranked phenomena. Numerical values were assigned to reflect knowledge levels and adequacy of data and analytical tools used to characterize the phenomena. The reader is referred to Reference 8.1.6 for detailed results of the two PIRTS performed. The following scales were used for ranking phenomena importance (Table 4-4) and knowledge levels (Table 4-5):

Table 4-4. LOCA PIRT phenomenon importance ranking

Importance Rank	Definition
High (H)	Significant influence on primary figure of merit
Medium (M)	Moderate influence on primary figure of merit
Low (L)	Small influence on primary figure of merit
Inactive (I)	Phenomenon not present or negligible

Table 4-5. LOCA PIRT knowledge level ranking

Knowledge Level	Definition
4	Well known/small uncertainty
3	Known/moderate uncertainty
2	Partially known/large uncertainty
1	Very limited knowledge/uncertainty cannot be characterized

Conducting a PIRT early on in the design process provided an opportunity for the PIRT results to inform the design and prioritize the focus on additional research efforts, as appropriate. After additional design and safety analysis work has been completed, the PIRT will be revisited to determine if the original importance rankings and knowledge levels assigned by the panel to each phenomenon remain appropriate.

4.5.2 Element 2 – Develop Assessment Base

Step 5, 6 and 7 – Specify Objectives for Assessment Base, Perform Scaling Analysis and Identify Similarity Criteria, and Identify Existing Data and/or Perform Integral Effects Tests (IETs) and Separate Effects Tests (SETs) to Complete the Database.

Review of the PIRT indicated that a number of the LOCA phenomena in the NuScale module are similar to and within the range of those in a typical PWR. The knowledge base for these phenomena is typically relatively high. In conjunction with the need for experiments, a detailed scaling analysis will be completed for the NuScale module, with the main purpose of generating design parameters for integral test facility construction and operation. This analysis follows a hierarchical top-down/bottom up methodology based on non-dimensional similitude between facilities. Results of scaling analyses will be used to update the existing OSU integral test facility to the latest NuScale module design.

An additional objective of the assessment base will be to address the needs for additional knowledge and/or prediction capabilities of the code that were identified for the high-importance and low/medium-knowledge phenomena identified in the preliminary NuScale LOCA PIRT. To obtain experimental data relevant to the unique features of the NuScale modules, three major test

programs are currently being carried out. One such facility is the NuScale Integral Test Facility at OSU, described in Section 4.5.2.1, where a wide range of AOO and SBLOCA tests are planned specifically for the NuScale module design certification activities. Others facilities include SIET and GEST test facilities in Italy described in Sections 4.5.2.2 and 4.5.2.3 respectively.

The initial matrix of existing and planned experiments that will be used to validate the combined physics in the RELAP5 code is shown in Table 4-6 as it relates to the PIRT and the NuScale reactor design. Table 4-7 is a preliminary list of NuScale IETs to support facility characterization and LOCA EM assessments.

Table 4-6. Initial RELAP5 validation matrix for SETs and IETs

[[

Plant System	Phenomena	Test or Reference(s)

Plant System	Phenomena	Test or Reference(s)

]]3(b)

Table 4-7. Planned NuScale IETs

Test Series	Facility Characterization	Descriptions
1	[[	[[
2		
3		
4		

Test Series	Facility Characterization	Descriptions
5		
6		
7		
8		
9		
10		
11		
12		
13	]]3(b)	]]3(b)

4.5.2.1 NuScale Integral Effects Tests at Oregon State University

The OSU test facility (Figure 4-2) provides integral system data for steady state and transient experiments. These experiments can be related to the operational, LOCA and Non-LOCA AOOs and PAs. The data is used for system characterization, model validation, informing operational procedures, and safety methodology development. OSU has significant testing capability having performed Department of Energy and NRC certification tests for the AP600 and AP1000 designs. A 10 CFR 50, Appendix B, NQA-1, 10 CFR 21 test program is in place with an extensive experience base. The NuScale IETs at OSU will include characterization tests, a spectrum of design-basis small-break LOCAs and Non-LOCA events. The facility will undergo modifications that include a containment upgrade, new instrumentation, and addition of a decay heat removal system.

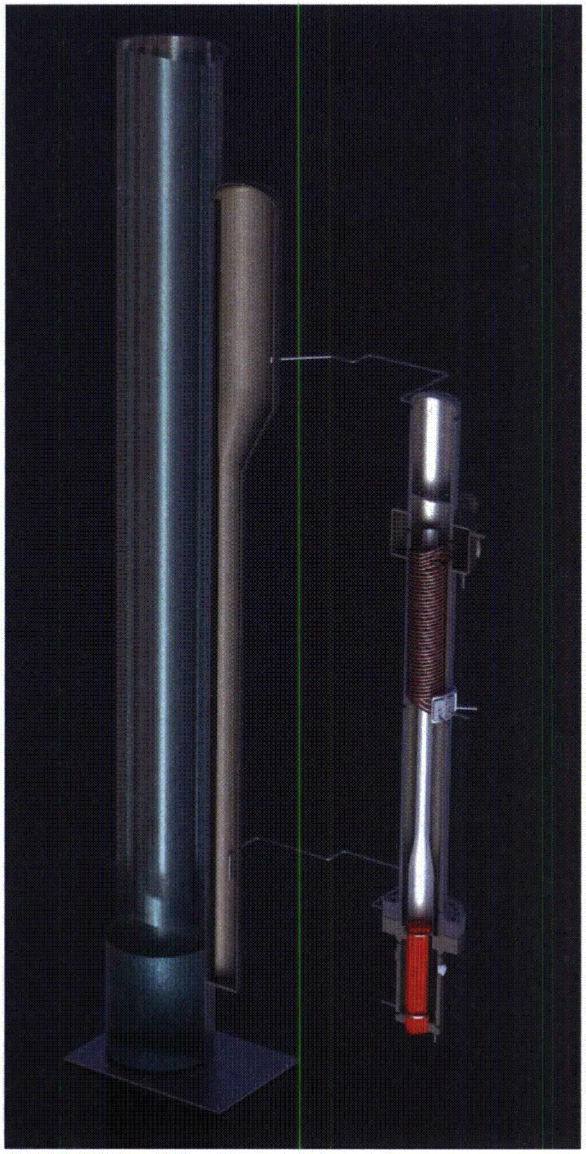


Figure 4-2. Schematic of the NuScale Integral Effects Test Facility at Oregon State University

4.5.2.2 SIET Helical Coil Steam Generator Test Facility

Since the NuScale module has helical coil steam generators (HCSGs) integral to the reactor vessel, the Societa Italiana Esperienze Termoidrauliche (SIET) Helical Coil Steam Generator Test Facility is another facility scheduled for use by NuScale. There are two independent HCSG tube bundles, allowing operation of half of the HCSGs under transient conditions (plant startup, main steam line break, steam generator tube rupture, etc.). Each HCSG tube bundle is intertwined with the other with alternating vertical sets of tubes.

SIET will conduct NuScale prototypic HCSG tests at facilities in Piacenza, Italy. The helical coil steam generator is an integral part of the DHRS and thus a part of the ultimate heat sink system. Its thermal-hydraulic performance directly impacts plant transient conditions during DHRS operation. Although existing HCSG heat transfer and flow models are available for similar configurations, new tests will be conducted to match the prototypic conditions of the NuScale design. The objective of the program is to obtain full-scale prototypic HCSG thermal-hydraulic performance data to validate NuScale computer codes. Testing will be conducted in full length tubes for a range of fluid and flow conditions.

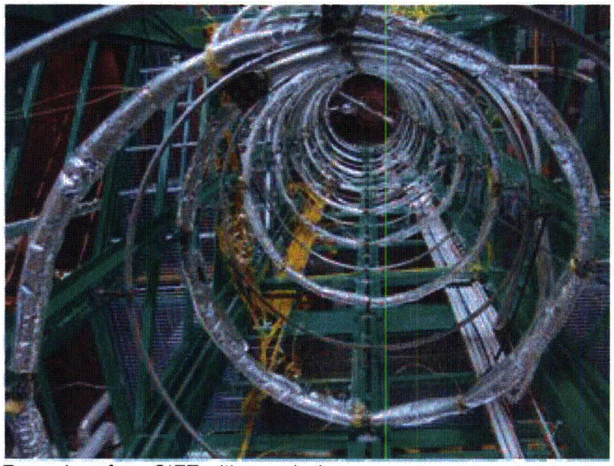
Two test facilities will be used to obtain HCSG data for

- Electric direct heated testing of helical coil tubes that allows for detailed investigations of thermal-hydraulic behavior
- Fluid heated testing of helical coil tubes allowing for testing in a prototypic bundle geometry with prototypic fluid heat transfer inside and outside of the tubes

The objectives of the electric direct heated tests are to obtain detailed thermal-hydraulic data for three highly instrumented parallel helical coil tubes. The tests will provide measurements and assessments of the following:

- Two-phase pressure drop
- Onset of boiling
- Dry-out location
- Average void fraction
- Radial and axial wall temperature profiles
- Axial fluid temperatures
- Wall heat flux
- Effects of tube inclination, and flow regime data
- Stability map for onset of density wave oscillations for flow in parallel tubes

Over 200 steady-state and instability tests will be conducted over a range of power varied from 20% to 110% of prototypic conditions. A schematic and photo of the electric direct heated test is shown in Figure 4-3.



Reproduce from SIET with permission

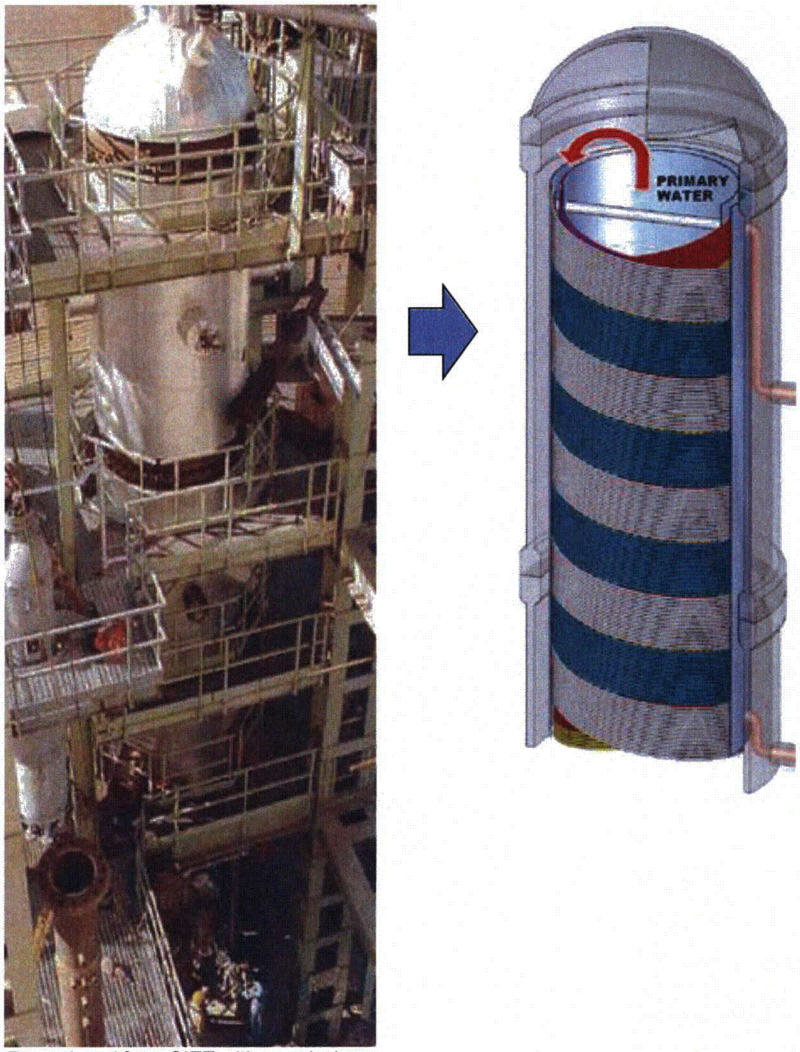
Figure 4-3. Schematic and photo of the SIET electric direct heated facility

4.5.2.3 SIET Generator Separator Test facility

The objective of the second set of tests at SIET, using the SIET Generator Separator Test (GEST) facility, is to obtain primary and secondary side thermal-hydraulic data for a full length NuScale HCSG under prototypic operating conditions. The tests will provide measurements and assessments of the following:

- Steam temperature and pressure at the steam header exit for prototypic feedwater flows and primary loop conditions
- Secondary side pressure drops (inside the tubes)
- Primary and secondary side fluid temperatures
- Bundle heat transfer data at prototypic heat fluxes, temperatures, steam pressures and flows
- Primary side fluid cross-flow heat transfer and pressure drop data for the NuScale tube bundle geometry
- Secondary side feedwater stability measurements for low power operations

Figure 4-4 shows the GEST facility.



Reproduced from SIET with permission

Figure 4-4. Generator Separator Test facility

Together with the Stern Laboratory Critical Heat Flux tests described in Section 5.8, the above NuScale module specific tests will provide a substantial portion of the separate-effects experimental data to validate the safety analysis codes.

Steps 8 & 9 – Evaluate Effects of IET Distortions, SET Scale-up Capability, and Determine Experimental Uncertainties as Appropriate

Comparisons will be made between the NuScale validation database and the experimental objectives of Step 5 of the EMDAP, and additionally to the phenomena identified in the NuScale PIRT. Distortions due to missing or atypical phenomena will be identified and their effects determined. If the effects are significant, the test with the unacceptable distortion may be eliminated from the assessment and replaced with another test(s) that better exhibits the important phenomena.

Experimental uncertainties for each of the tests will be determined, and any data displaying uncertainties deemed unacceptable will not be used in the SBLOCA EM assessment.

4.5.3 Element 3 – Develop Evaluation Model

Step 10 – Establish an Evaluation Model Development Plan

A development plan for RELAP5 will be created that addresses the following specific areas as defined by the "Engineering Procedure 3.4.2 - Development and Maintenance of Level B Software," NP-EP-0303-321.R0 (Reference 8.1.7):

- Software project plan
- Software design specification
- User documentation
- Test planning
- Configuration management plan
- Implementation
- Verification
- Validation
- Peer review
- Installation and acceptance testing
- Operation and maintenance

Step 11 – Establish Evaluation Model Structure

The NuScale LOCA EM is being developed with the capabilities to analyze the primary and secondary systems, DHRS, containment heat removal system, and ECCS as described in Step 3. This capability is achieved by basing the EM on a generally applicable systems code, RELAP5, developed over many years specifically for nuclear reactor thermal hydraulic analysis, and modifying the code such that it conforms to 10 CFR 50, Appendix K requirements.

Step 12 – Develop or Incorporate Closure Models

Closure models required by 10 CFR 50, Appendix K will be used (Reference 8.1.8) accordingly for the NuScale SBLOCA EM. When not specified by Appendix K, other appropriate closure models will be developed from the experimental database and incorporated into RELAP5. Closure models will be verified to be used within their range of applicability or justification provided for extending their use beyond their original range.

For a deterministic evaluation model, there are several conservative restrictions on which closure models are acceptable for use in different areas of the methodology. RELAP5 has the models that allow it to meet many of the requirements detailed in Appendix K as the bases for a LOCA EM without modification, and it can be modified to completely conform to Appendix K by identifying the applicable missing models and integrating them into the code. The restrictions listed in 10 CFR 50, Appendix K are grouped into four subsections:

- Sources of heat during the LOCA
- Swelling and rupture of the cladding and fuel rod thermal parameters
- Blowdown phenomena
- Post-blowdown phenomena and heat removal by the ECCS

4.5.4 Element 4 – Assess Evaluation Model Adequacy

The adequacy of the NuScale ECCS EM will be assessed in two parts. The first is a bottom-up assessment of the closure relations selected in RELAP5 (Steps 13 through 15 in Figure 4-1). The second will be a top-down assessment of the governing equations, numerical solution methods, and integrated performance of RELAP5 (Steps 16 through 19 in Figure 4-1). These two assessment methods are discussed in more detail in the following sections. In addition, the uncertainties and biases will be addressed in Step 20 of the EMDAP.

4.5.4.1 Closure Relations (Bottom-up)

Step 13 – Determine Model Pedigree and Applicability

The first step in this process will be a review of the RELAP5 source code to confirm that the models and correlations presented in the code manual (Reference 8.1.4) are consistent with the coding. This is a verification process step to ensure that the base code will fulfill its intended function. Any discrepancies will be noted and resolved.

The pedigree of each selected model for the EM will be evaluated by NuScale to determine whether the model, as implemented in RELAP5, is consistent with its pedigree and whether its use over the broad range of expected conditions is justified. From a modeling perspective, the main area of evaluation will be to determine if existing RELAP5 models are applicable for modeling the high-importance phenomena.

Step 14 – Prepare Input and Perform Calculations to Assess Model Fidelity and/or Accuracy

As described in RG 1.203, the fidelity of the EM relates to the completeness of validation, benchmarking, or some combination of these comparisons. The differences between calculated results and experimental data for PIRT phenomena will be evaluated and resolved.

Step 15 – Assess Scalability of Models

The scalability of models that are used to predict high-importance phenomena is assessed by using a dynamical scaling process outlined in the NuScale Dynamical System Scaling Methodology report submitted to the NRC (Reference 8.1.9).

4.5.4.2 Integrated EM (Top-down)

Step 16 & 17 – Determine Capability of Field Equations and Numeric Solutions to Represent Processes and Phenomena and the Applicability of EM to Simulate System Components and Interactions

The two-fluid model formulation used in RELAP5 will be evaluated by NuScale for its acceptability for characterizing PIRT identified phenomena. The evaluation will consider field equation pedigree, key concepts, and processes.

NuScale will also evaluate the numerical solution technique and take into consideration convergence, property conservation, and code stability. Code convergence and stability has been examined previously (Reference 8.1.10). However, a complete assessment of the field equations and numerical solution methods can only be performed after completing some prescribed subset of the EM validation matrix, which will assist NuScale in evaluating the integrated capability of RELAP5 to model the plant system components and interactions. This is done to confirm that the various EM options, special models, and inputs have the capability to model the major systems and how they interact.

Step 18 & 19 – Prepare Input and Perform Calculations to Assess System Interactions and Global Response of the NuScale Plant Model with Scalability of Calculations and Data for Distortions

RELAP5 will be assessed against a variety of integral facility experiments. Of particular importance is the capability of RELAP5 to model data from the NuScale IET at OSU. This facility models the major components of the NuScale reactor, including the containment and containment cooling pool with a plan to add a DHRS. The RELAP5 nodalization and option selections for the integral and component assessment will be consistent between the experiments and the full-scale plant model. Nodalization sensitivity studies will be performed in both the test facility and full plant model. The differences between the calculated results and experiments will be evaluated and resolved.

As part of this step, a steady state and transient base deck for the full plant model will be prepared for LOCA analysis. This base input model will provide the groundwork for analyses to be performed in Step 20 of the EMDAP.

Any unexplained differences between facility data and code calculations may indicate experimental and/or code scaling distortions. An assessment of the scalability of integral data and code calculations will be performed in accordance with NuScale's dynamical scaling methodology.

Step 20 – Determine Biases and Uncertainties

The complexity of the transient determines the scope and depth of the treatment of uncertainties in the EM. In Step 1 of the EMDAP, the purpose of the analysis was stated to demonstrate compliance with ECCS criteria in 10 CFR 50.46 and the analysis type selected was that described in Appendix K of 10 CFR 50. Conservative input parameters will be selected based on a limited assessment of biases and uncertainties. This evaluation will be conducted by running sensitivity cases. The cases will evaluate the effect on the calculated results due to (1) variations in noding, (2) predominate calculation phenomena, and (3) values of parameters over their applicable ranges. Where code sensitivities are identified, the choices made to provide a conservative input model shall be justified.

4.6 Adequacy Decision

As discussed in RG 1.203 the adequacy decision of the EM is the culmination of the EMDAP. During the EMDAP process, the adequacy of the NuScale LOCA EM will be continuously evaluated.

5.0 SUBCHANNEL ANALYSIS USING SCANR

5.1 Scope

NuScale is developing an in-house subchannel analysis code, Subchannel Analyzer for NuScale Reactors (SCANR), based on COBRA-IIIC (Reference 8.1.11). Since the NuScale reactor has unique design features and operating conditions, such as natural circulation flow through the core along the primary loop, it is necessary to replace some of the existing COBRA-IIIC models with more improved and appropriate ones for implementation into SCANR. Some of the major differences between SCANR and the existing subchannel analysis codes, including COBRA-IIIC include:

- Local-parameter-based rod-bundle critical heat flux (CHF) correlation, which is applicable to the NuScale reactor low flow conditions, to be developed using the CHF data obtained from the CHF tests performed by Stern Laboratories
- Single-phase convection wall heat transfer coefficient correlation applicable to the NuScale reactor natural circulation conditions
- More robust and efficient numerical solution scheme (overall solution algorithm and matrix solver) applicable to the NuScale reactor low flow conditions

This section describes the development and use of SCANR for steady-state core thermal-hydraulic analysis, transient, and accident analysis.

5.2 Introduction

The main purpose of SCANR is to generate local thermal-hydraulic conditions throughout the reactor core using the subchannel analysis approach with a two-phase mixture flow model. SCANR will be used for steady-state and transient thermal-hydraulic analyses for the NuScale reactor core under single- and two-phase flow conditions.

The primary intended applications of SCANR are to generate local thermal-hydraulic conditions for CHF tests in developing a CHF correlation and to provide local thermal-hydraulic conditions throughout the reactor core in calculating the minimum departure from nucleate boiling ratio (MDNBR). SCANR can also provide more realistic boundary conditions, such as the axial profiles for the coolant temperature and wall heat transfer coefficient in a limiting subchannel, for the fuel rod performance analyses.

DNBR calculations require detailed information about the local thermal-hydraulic conditions in the reactor core, because most of the CHF correlations for PWRs are local parameter based. Currently, the most reliable and practical design and analysis tool for generating the local core thermal-hydraulic conditions is a subchannel analysis code. SRP 4.4, "Thermal and Hydraulic Design," specifies that subchannel analysis codes should be used to calculate local fluid conditions within fuel assemblies for use in PWR DNB correlations.

[[

]]3(b)

[[

]]3(b)

Figure 5-1. Minimum departure from nucleate boiling ratio calculation process

5.3 Regulatory Requirements and Guidance

The NuScale core thermal-hydraulic design is based on the following General Design Criteria (GDC) from 10 CFR 50, Appendix A (Reference 8.1.12): GDC 10, GDC 11 and GDC 12. In particular, a key acceptance criteria to meet in the PWR core thermal-hydraulic design and safety analyses is that there should be at least a 95% probability at the 95% confidence level (95/95) that the hot fuel rod in the reactor core will not experience a departure from nucleate boiling (DNB) during normal operation or anticipated operational occurrences (AOOs). In other words,

the MDNBR should be greater than the 95/95 DNBR limit based on an acceptable CHF correlation.

5.4 Subchannel Analysis Code—SCANR

SCANR is a subchannel analysis code based on a two-phase mixture flow model. The governing equations for the coolant subchannels are formulated based on the subchannel analysis approach and the two-phase mixture flow model, which are able to be reduced to the single-phase liquid flow model.

[[

]]3(b)

The NuScale reactor is a PWR and is designed such that the core will always be covered with coolant for the design basis accident conditions. [[

]]3(b)

SCANR also solves the conduction equation for the fuel rods to obtain the fuel rod temperature distribution in the temporal and spatial (radial and axial) domains.

5.4.1 SCANR Field Equations

The governing equations of SCANR for the coolant subchannels and the fuel rods are based on those for COBRA-IIIC.

The governing field equations for the coolant subchannels include the following:

- [[

]]3(b)

5.4.2 SCANR Closures Relations

The physical models or constitutive relations to close the governing field equations for SCANR include the following:

- [[

]]3(b)

5.4.3 SCANR Numerics

[[

]]3(b)

5.4.4 SCANR Capabilities

SCANR will provide user convenience features, including the following:

- Input/output unit conversions between the British and SI unit systems
- MDNBR iterations on operating parameters for set point analyses
- Automatic generation of subchannel and lumped channel models for DNB analyses

5.5 SCANR Inputs and Outputs

Since SCANR is a component analysis code, focusing on the reactor core, it requires boundary conditions from other codes, such as the core axial and radial power distributions, system pressure, and core inlet temperature and flow distributions.

With those boundary conditions, SCANR generates the local thermal-hydraulic conditions in the reactor core to be used in the minimum DNBR calculation. Some of the local thermal-hydraulic conditions generated by SCANR are used as boundary conditions for the fuel rod performance analysis.

5.5.1 Steady-State Core Thermal-Hydraulic Model

[[

]]3(b)

[[

]]3(b)

Figure 5-2. Example nodalization with SCANR

5.5.2 Transient and Accident Core Thermal-Hydraulic Models

[[

]]3(b)

5.5.3 Conservatisms

[[

]]3(b)

5.6 Interfaces with Other Codes

Figure 5-3 shows the interfaces of SCANR with other NuScale design and safety analysis codes. For the NuScale reactor safety analyses, [[

]]3(b) In the figure, the green boxes around SCANR represent the major data flow among the different codes.

5.7 SCANR Assessment

Based on the operating conditions of the NuScale reactor, and with consideration of the NuScale LOCA and Non-LOCA PIRTs (Reference 8.1.6 and 8.1.14) the SCANR validation matrix has been established as shown in Table 5-1. The validation matrix provides the test category, test identification, the subchannel analysis codes that were validated against tests in literature, measured test data, and the SCANR features to be validated. [[

]]3(b)

[[

]]3(b)

Figure 5-3. Code interfaces and data flow

Table 5-1. SCANR validation matrix

[[

]]3(b)

5.8 Critical Heat Flux Correlation Development and Testing

5.8.1 Critical Heat Flux Tests

For the departure from nucleate boiling (DNB) analyses for the NuScale reactor core with the subchannel analysis code SCANR, an applicable CHF correlation is needed. Since the NuScale reactor has unique design features and operating conditions, such as low flow conditions from natural circulation cooling, the existing rod-bundle CHF correlations are not directly applicable. It is therefore necessary to perform new CHF tests to obtain data for developing a CHF correlation applicable to the unique NuScale design. The CHF tests for developing a CHF correlation will be performed under steady-state and transient conditions at Stern Laboratories. The steady-state CHF test condition matrix covers the following parameter ranges:

- [[
-
-

]]3(b) The CHF test configurations are shown in Table 5-2.

Table 5-2. Critical heat flux test configurations

[[

]]3(b)

For conventional pressurized-water reactors, CHF correlations developed under steady-state conditions are used in transient reactor conditions. Limited transient CHF testing is performed to confirm the validity of the correlation based on steady-state data. Similar limited transient CHF testing will be performed by Stern Laboratories.

5.8.2 Mixing Tests

For subchannel analyses to provide local thermal-hydraulic conditions for both the CHF correlation development and the DNBR calculations, one of the most sensitive empirical parameters to be supplied as an input is the turbulent mixing parameter. This parameter depends on specific fuel assembly design characteristics, including the grid spacer design. The thermal mixing tests will be performed by Stern Laboratories.

5.8.3 Pressure Drop Tests

For subchannel analyses, the information about the hydraulic resistances along the rod bundle is also needed as an input. Therefore, it is necessary to perform pressure drop tests to evaluate the wall friction factor and grid spacer loss coefficients. The pressure drop tests will be performed by Stern Laboratories.

5.8.4 NuScale Critical Heat Flux Correlation Development and Application

The core thermal design process for the NuScale reactor core is outlined in Figure 5-4.

[[

]]3(b)

Figure 5-4. Core thermal design process

[[

]]3(b)

6.0 NEUTRONICS

6.1 Scope

This section provides an overview of the core physics and neutronics calculation methods that will be employed in support of design certification of the NuScale reactor module. The descriptions provided in this section are intended to provide a high-level overview of the calculations and methods that will be utilized.

It is noted that the methodology description of the nuclear analysis codes provided in Section 6.4 is focused on identifying and describing the individual codes that will be employed and the specific interfaces between each code. The fundamental physics methods that are utilized by the codes as part of the calculation algorithms are not addressed in this report.

6.2 Regulatory Requirements

The nuclear design of the NuScale reactor is performed to meet the regulatory requirements of the General Design Criteria (GDC) from 10 CFR 50, Appendix A. Specifically, the requirements that pertain to the nuclear design from a neutronics perspective are GDC 10, GDC 11, GDC 12, GDC 13, GDC 20, GDC 25, GDC 26, GDC 27, and GDC 28 (Reference 8.1.12).

6.3 Core Design Targets

Preliminary core design targets have been established for the NuScale reactor core as shown in Table 6-1. The preliminary core design targets are based on values that are within the experience base of the industry and are intended to provide confidence that fuel loading patterns that meet these targets will retain significant margin against the final design limits that will be determined from the safety analysis.

The core design targets shown in Table 6-1 are used as input to the core physics and safety analysis calculations. The results of these calculations, in conjunction with the uncertainty values derived from the nuclear analysis code validation work, will be used to establish final core design limits.

Table 6-1. Preliminary core design targets

[[

]]3(b)

6.4 Neutronics Analysis Codes

6.4.1 CMS Code Suite

The CMS code suite is developed by Studsvik Scandpower and is widely used throughout the industry for the analysis and design of boiling water reactors (BWRs) and pressurized-water reactors. The calculation approach in CMS is the conventional multi-stage approach, which consists of a detailed neutron transport theory calculation to generate group-wise nuclear data, and a coarse diffusion calculation that uses the group-wise data in the reactor core design and analysis process. Figure 6-1 shows the calculation flow in CMS.

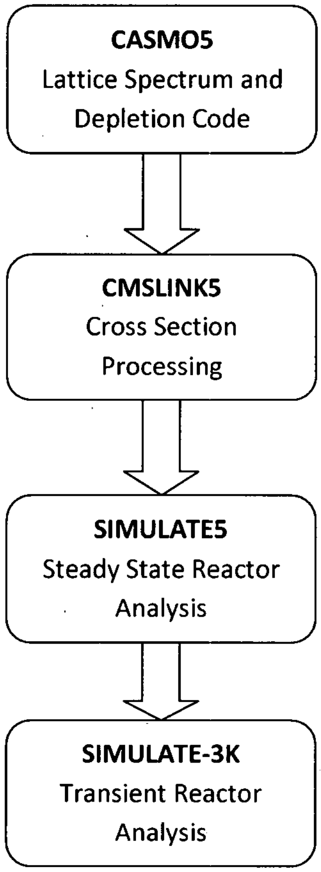


Figure 6-1. CMS code suite computational components

The four major CMS components (Reference 8.1.45) are the two-dimension (2D) lattice transport code CASMO-5, the data processing code CMSLINK5, the three-dimension (3D) steady state nodal diffusion code SIMULATE-5, and the 3D reactor transient code SIMLUATE-3K. The following subsections describe each of these modules in more detail.

6.4.1.1 CASMO-5

The main purpose of CASMO-5 (Reference 8.1.46) is to generate the nuclear data that is required by SIMULATE-5 (Reference 8.1.47 and 8.1.48) in order to simulate the reactor under all possible operating conditions. CASMO-5 is therefore used to perform fuel design calculations and to generate nuclear data for each type of fuel assembly under various conditions of core burnup, boron concentration, fuel temperature, rodded and un-rodded conditions, etc. CASMO-5 functionalizes the assembly cross section data as a function of reactor state into a single data library. This library is used by SIMULATE-5 to select the appropriate nuclear data according to the reactor state being analyzed.

CASMO-5 is a multi-group 2D neutron transport theory code with depletion capabilities for calculating BWR and PWR fuel assemblies (Reference 8.1.46).The code handles a geometry consisting of cylindrical fuel rods of varying composition in a square pitch array with allowance for fuel rods loaded with gadolinium or erbium, integral fuel burnable absorber, as well as cluster

control rods, in-core instrument channels, water gaps, and cruciform control rods in the regions separating fuel assemblies.

The method of characteristics transport solution within CASMO-5 is used to further condense and homogenize (spatially average) the detailed cross sections into group-wise nodal diffusion data for SIMULATE-5. This process is performed for each assembly for each reactor operating condition. To facilitate this, CASMO-5 provides an automated capability that allows for generating a set of data as a function of reactor state parameters (Reference 8.1.45). This data is used in conjunction with an interpolation scheme in SIMULATE-5 to access the cross section as a function of reactor state point. CASMO-5 produces a data library for each assembly type. This data is then processed into a single binary library by the linking code CMSLINK5.

6.4.1.2 CMSLINK5

CMSLINK5 is the linking code that processes the CASMO-5 data into a single nuclear data library for use in SIMULATE-5 and SIMULATE-3K. The code collects the following data from the CASMO-5 data files: multigroup macroscopic/microscopic nodal cross sections, multigroup submesh macroscopic cross sections, detector, pin power reconstruction, kinetics, isotopic, spontaneous fission, and thermal-hydraulic data (Reference 8.1.49).

CMSLINK5 is capable of processing hot and cold data for standard PWR conditions with and without burnable poison, pulled and reinserted burnable poison for PWR segments, standard hot and cold BWR data, and standard hot and cold PWR and BWR reflector data. CMSLINK produces the scoping libraries for use in core design studies.

The parameterization of the cross sections are automated in CMS and the relevant statepoint dependent data generated in CASMO-5. The cross section parameterization model is quite comprehensive and the cross sections are tabulated as a function of burnup (exposure), moderator temperature, spacer grids, historical and instantaneous fuel temperature, boron concentration, control rod insertion, xenon concentration and several other parameters that affect the cross sections (Reference 8.1.49). The functionalization of the cross sections and k_{inf} data are output to a single library.

CMLSINK5 is also used to prepare the nuclear data for a varying number of energy groups for use in SIMULATE-5 ranging from one to eight groups. It uses flux weighting to collapse multi-group CASMO-5 data to the desired group-wise structure in SIMULATE-5.

6.4.1.3 SIMULATE-5

SIMULATE-5 is Studsvik Scandpower's next generation core solver and is a 3D multi-group nodal diffusion code for the analysis of BWRs and PWRs (Reference 8.1.48). The code is used to perform core design and safety parameter analysis. The main goal of SIMULATE-5 is to compute the spatial and energy dependent flux and pin power distribution over the reactor core and to accurately deplete and compute the reactor nuclide inventory.

SIMULATE-5 contains several improvements over SIMULATE3. The improvements were driven by the need to model the strong heterogeneous effects in modern reactor cores that result from the use of mixed oxide (MOX) fuel and burnable poisons. SIMULATE-5 also contains an improved pin power reconstruction model (in conjunction with CASMO-5) as well as a more sophisticated thermal-hydraulics model based on COBRA-IIIC. SIMULATE-5 has also been fully integrated into the existing CMS code system and the transient code SIMULATE-3K is capable of reading SIMULATE-5 data for use in transient analyses (Reference 8.1.48).

In SIMULATE-5, each assembly in the core is typically subdivided into 24-25 axial homogeneous material nodes. Provision is made for both axial and radial material heterogeneities that may

occur inside a node. In the axial direction, the heterogeneities due to spacer grids, control rods, enrichment and burnable absorber zoning, and staggered assembly heights are accounted for. In the radial direction, the fuel assembly is divided into a 5 x 5 planar submesh (Reference 8.1.48).

The neutronics and thermal-hydraulics module are coupled in SIMULATE-5. This coupling allows for a better approximation to computing the fuel and coolant temperatures and the interaction of feedback effects. The PWR thermal hydraulics solver models the entire region between the lower and upper core plates. The BWR and PWR core portion of the thermal-hydraulics models are treated essentially identically, with each assembly having an active channel and a number of parallel bypass channels. For each axial node of a channel, the total mixture mass, steam mass, mixture enthalpy, and mixture momentum balance equations are solved (a four-equation model). The 3D fuel temperatures are computed in the thermal-hydraulics module by solving the radial Fourier heat conduction equation.

The major inputs to SIMULATE-5 are reactor dimensions, material data, fuel assembly descriptions, control rod bank description, reactor power, and thermal-hydraulic data. SIMULATE-5 solves for reactor flux and power per assembly node and computes assembly nuclide inventories as a function of exposure. In addition, SIMULATE-5 uses the nodal data in conjunction with one-dimensional (1D) SIMULATE-5 axial data and 2D CASMO-5 data to synthesize (reconstruct) the assembly pin-by-pin flux and power data. It also computes a series of power peaking factors over the core.

6.4.1.4 SIMULATE-3K

SIMULATE-3K is an advanced, nodal code for transient analysis of both PWRs and BWRs (Reference 8.1.50). SIMULATE-3K can be used to model and analyze a variety of reactor transients including rod ejection, bank withdrawal transients, dropped rod analysis, and boron dilution events.

SIMULATE-3K is based on SIMULATE-3 (the predecessor of SIMULATE-5) and explicitly couples both the neutronics and the thermal-hydraulics calculations for each assembly in the core. SIMULATE-3K solves the transient 3D, two-group neutron diffusion equation. Intra-nodal flux and power distributions within each node are used to compute the power, fuel temperatures, and enthalpies for every axial level of every fuel pin in the core during transients.

The neutronics are essentially the same as that used in SIMULATE-5, except that only a two-group diffusion calculation is performed. The thermal module consists of a pin conduction model for the calculation of the pin surface heat flux and internal temperature distribution. The fuel pin heat generation rate is directly coupled to the pin power from the neutronics calculation. In turn, the thermal-hydraulic module provides the neutronics calculation with the appropriate feedback that can be used to construct new cross sections.

The thermal-hydraulic channel model uses a 5-equation model for the liquid and vapor phase for mass and energy conservation. For cases in which there is relative acceleration between the vapor and liquid phase, a 6-equation model is used.

6.4.2 Monte Carlo N-Particle Transport Code

The general purpose Monte Carlo N-Particle (MCNP) transport code (Reference 8.1.51) will be used to support the validation of the CMS code suite by affording code-to-code comparisons between the results of the CMS code and those of MCNP. The MCNP code is used for this purpose because of its high fidelity exact geometrical treatment.

MCNP is a general-purpose, continuous-energy, generalized-geometry, time-dependent, coupled neutron/photon/electron Monte Carlo transport code. It can be used in several transport modes:

neutron only, photon only, electron only, combined neutron/photon transport where the photons are produced by neutron interactions, neutron/photon/electron, photon/electron, or electron/photon. The neutron energy regime is from 10-11 MeV to 20 MeV for all isotopes and up to 150 MeV for some isotopes, the photon energy regime is from 1 keV to 100 GeV, and the electron energy regime is from 1 KeV to 1 GeV. For neutrons, all reactions given in a particular cross section evaluation such as the evaluated nuclear data file (e.g., ENDF/B-VII) are accounted for. Thermal neutrons are described by both the free gas and $S(\alpha,\beta)$ scattering models to account for molecular binding effects below 4 eV. For photons, the code accounts for incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption, absorption in pair production with local emission of annihilation radiation, and bremsstrahlung. In the neutron transport mode the code can perform fixed source calculations or calculate eigenvalues for critical systems.

6.4.3 Calculation Inputs and Outputs

6.4.3.1 CASMO-5 Inputs and Outputs

CASMO-5 is a 2D lattice physics transport code used in the design of the NuScale reactor core. This code is used to create the fuel assembly and reflector segments that comprise the 3D model of the core that is built in SIMULATE-5. CASMO-5 generates group data for normal operating parameters, as well as for perturbations, called branching cases, of many of the parameters. Parameters that may vary during the operation of the reactor are perturbed so that group data exists under many of the sets of conditions that may occur. Group data for these branching cases is calculated for the depletion of the assembly.

6.4.3.1.1 Input Model

The CASMO-5 model for the assemblies in the NuScale reactor core has a standard structure containing user-supplied information, including

- Assembly and pin geometry
- Control rod dimensions and absorbers
- Materials
- Fuel enrichment, density, and burnable poison loading
- Fuel pin and guide tube layout
- Fuel and moderator temperature
- Operating pressure
- Power density

6.4.3.1.2 Inputs to SIMULATE

Group data is calculated in branching cases for individual assemblies over the requested range of exposure and stored in a card image file. These files are collected for each assembly type and processed by CMSLINK in order to create a single library file containing all group data in a single location. The library file is the major product of CASMO-5 and CMSLINK that serves as input to the core simulator, SIMULATE-5.

6.4.3.2 SIMULATE-5 Inputs and Outputs

SIMULATE-5 is used to construct 3D models of the NuScale reactor core using the group data library generated by CASMO-5 and CMSLINK in the solution of a nodal diffusion code. Cycle

depletions produce 3D flux data at each depletion step, which is used to derive information related to power distribution, peaking, and other relevant neutronics values. Additional nuclear physics calculations with results important to safety analysis of the reactor core can be performed at any depletion step.

6.4.3.2.1 Input Model

In SIMULATE-5, the NuScale reactor core is described in a base model. This model sets up the reactor geometry, operating conditions, and calculations to be performed by specifying the following:

- Fuel assembly data
- Assembly and reflector radial layout
- Axial zoning of each assembly type
- Previous cycle information
- Spacer grid geometry and location
- Control rod locations and axial description
- General operating data
- Temperature correlations
- Axial mechanical description
- Thermal-hydraulic calculation models
- Requested depletion and calculations

Using the information described above, SIMULATE-5 performs core depletions, control rod worth and shutdown calculations, power distribution calculations, models core follow, and determines other information used in safety analysis.

6.4.3.3 SIMULATE-3K

The neutronics calculations relevant to the transient analysis code SIMULATE-3K are described in Section 7.0.

6.5 Neutronics Code Validation

6.5.1 Approach

The purpose of neutronic code validation efforts is to provide confidence in the ability of the neutronics codes to predict the neutronic behavior of the NuScale reactor core. The code validation also quantifies the accuracy of the code predictions for parameters of interest. NuScale will validate the CMS code suite by performing benchmark calculations of suitable reactor physics experiments and comparing the calculated and experimental results. Suitable experiments are identified as those having values of one or more key parameters that are similar to those of the intended NuScale application domain.

The following provides a representative list of the physics parameters for which bias and uncertainty values will be calculated:

- [[

]]3(b)

For certain key parameters, or in cases where the combined effects of multiple parameters are of interest, experimental data may be unavailable. In these cases, a comparison will be made between the results of the CMS code and those of a second code system, which, by virtue of prior validation and methodology, is known to capture the effects of the parameters with high fidelity (e.g. MCNP).

For NuScale applications, this code-to-code comparison generally applies to consideration of the potential anisotropic flux at the edges of the NuScale core. In this case, the accuracy of the diffusion theory methodology underlying the CMS code suite may be assessed by comparison with the exact geometrical treatment afforded by the general purpose Monte Carlo code MCNP (Reference 8.1.51).

6.5.2 Benchmark Data

Benchmarks are selected in order to validate one or more of five broad categories of physical phenomena:

- Reactor statics, including k_{eff} and fission/power distributions
- Materials and cross section libraries, including depletion isotopic concentrations
- Experiments with geometrical similarity to the NuScale reactor core
- Reactor dynamics, including reactivity coefficients related to rod worth, temperature, boron concentration, etc.
- General benchmark experiments with characteristics that may expose trends in bias and bias uncertainty or provide verification of code performance against known solutions

A total of four benchmark groups have been selected for analysis to support the neutronics codes validation:

- Critical benchmark experiments
- Experimental reactor benchmarks
- Commercial reactor critical (CRC) benchmarks
- Spent fuel composition benchmarks

These four benchmark groups are described in the subsections that follow and include a description of the benchmark set being considered for analysis in each group.

6.5.2.1 Static Critical Benchmark Experiments

[[

]]3(b)

Table 6-2. [[ ]]3(b)

[[

]]3(b)

[[

]]3(b)

6.5.2.2 Dynamic Experimental Reactor Benchmarks

The International Handbook of Evaluated Reactor Physics Benchmark Experiments (Reference 8.1.53) includes a collection of evaluated experimental reactor benchmarks relevant to the general area of reactor dynamics.

Several experiments documented in the IHERPBE are considered to provide good analogues of the NuScale reactor design and thus are considered to represent suitable benchmarks for the purpose of establishing nuclear uncertainty factors. Table 6-3 provides a list of the IHERPBE experiments that are being considered for benchmarking the MCNP and CMS nuclear analysis codes.

Table 6-3. [[ ]]3(b)

[[

]]3(b)

6.5.2.3 [[ ]]3(b)

6.5.2.3.1 [[ ]]3(b)

[[

]]3(b)

Table 6-4. [[ ]]3(b)

[[

]]3(b)

[[ ]]3(b)

6.5.2.3.2 [[ ]]3(b)

[[

]]3(b)

6.5.2.4 Commercial Reactor Critical Benchmarks

A commercial reactor critical (CRC) state-point is either a hot zero-power critical condition attained after sufficient cooling time to allow the fission product xenon inventory to decay, or at-power equilibrium critical condition where xenon worth has reached a stable value.

[[

]]3(b)

Table 6-5. Commercial reactor critical benchmark experiments

[[

]]3(b)

6.5.2.5 Spent Fuel Composition Benchmarks

[[

]]3(b)

Table 6-6. [[ ]]3(b)

[[

]]3(b)

The data provided in Table 6-6 provides an empirical means of validating the depletion models of the CMS code suite.

6.5.3 Code-to-Code Comparative Analysis

The preceding sections document the code-to-code validation approach of the CMS code suite calculations that will support the core design and safety analysis of the NuScale reactor module. Those sections included a description of the explicit experimental and commercial benchmarks that are being considered for the validation analysis. However, due to the unique design of the NuScale reactor core, it is recognized that the experimental data may, in some cases, be unavailable or insufficient. In these cases, a comparison is made between the results of the CMS code and those of a second code system, which, by virtue of prior validation and methodology, is known to capture the physics with high fidelity.

[[

]]3(b)

The neutronic parameters that are examined in the code-to-code analysis can be split into two groups; global parameters and local parameters.

[[

]]3(b)

6.5.4 Uncertainty Values and Areas of Applicability

The objective of the CMS code suite validation exercise is to establish biases and uncertainties for components or parameters of interest that contribute to significant effects on the total uncertainty of the CMS model to predict core physics parameters over a demonstrated range of area of applicability with a 95%/95% confidence level.

The development of biases and uncertainties for the NuScale reactor core physics calculations will use an objective, rigorous, graded approach. In such an approach, the effort in quantifying an uncertainty component must be proportional to the contribution of that component to the total uncertainty. A graded approach will focus the uncertainty analysis consideration and thus more time on those components that contribute to a large effect on total uncertainty.

6.5.4.1 Code Bias

The code bias is calculated using the statistical technique that is most appropriate to the data population established. In cases where the data population follows a normal distribution, bias values will be derived using a 95/95 confidence level and determined as a one-sided tolerance limit in accordance with common statistical techniques.

7.0 ROD EJECTION ACCIDENT

7.1 Scope

The control rod ejection accident assumes that one control rod is rapidly ejected from the reactor due to a failure of a control rod drive mechanism housing mounted near the top of the reactor vessel. This event results in a reactivity excursion and a loss of primary coolant through the breach of the pressure boundary. The initial conditions for the event vary with the reactor critical from zero to full power operation, and at any time during the fuel cycle. For the NuScale design, only Control Rod Group 1 (Figure 2-4) consisting of four control rods located in symmetric locations in the reactor, is inserted with the reactor critical. Consequently, there is only one core location where a control rod can be ejected. Insertion limits for Group 1 as a function of power level will limit the ejected rod worth and thereby limit the consequences to acceptable values.

7.2 Regulatory Requirements and Guidance

Standard Review Plan (SRP) Section 15.4.8 (Reference 8.1.57) provides NRC guidance on an acceptable methodology for performing the rod ejection analysis. 10 CFR 50, Appendix A (Reference 8.1.12), General Design Criterion 13, "Instrumentation and Control," and General Design Criterion 28, "Reactivity Limits," are applicable to the rod ejection event.

GDC 13 specifies design criteria for instrumentation to detect and actuate automatic mitigation of the rod ejection accident. GDC 28 specifies design criteria on reactivity insertion resulting from the rod ejection event to ensure that the consequences are acceptable. Specifically, GDC 28 requires the reactivity control system to be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary greater than limited local yielding nor sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. Regulatory Guide 1.77 (Reference 8.1.58), "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," was also considered during the development of the NuScale rod ejection methodology.

The regulatory acceptance criteria for the rod ejection accident are specified in SRP Section 4.2, Appendix B (Reference 8.1.59), "Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents," and in SRP Section 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)." SRP Section 4.2, Appendix B revises the acceptance criteria in SRP Section 15.4.8 to account for the effects of fuel exposure. It is noted that SRP Section 4.2, Appendix B contains interim limits that may be revised prior to becoming final. The rod ejection methodology will be revised as necessary based on any changes associated with the future final acceptance criteria. Due to the rod ejection being classified as an accident, failure of the cladding barrier is allowed provided that the radiological consequences meet the following criteria:

- Table 6, "Accident Dose Criteria" for EAB and LPZ dose criteria of Regulatory Guide 1.183 (Reference 8.1.60)
- Appendix H, "Assumptions for Evaluating the Radiological Consequences of a PWR Rod Ejection Accident"
- 10 CFR 50.67 (Reference 8.1.61), "Accident Source Term" (for control room dose criteria)
- Core coolability is maintained
- The stresses resulting from the peak pressure value from the event must remain within "Service Limit C" of the ASME Boiler and Pressure Vessel Code (Reference 8.1.62).

With regard to the core coolability criteria, the NuScale rod ejection methodology and analyses will maintain the cladding barrier intact with regard to pellet-cladding mechanical interaction

(PCMI) by meeting the Δcal/gm limit of SRP Section 4.2, Appendix B, Figure B-1, and by showing no incipient melting of the fuel. Therefore, methodology to model fuel dispersion and interaction with the coolant following failure of the cladding due to PCMI or fuel melting is not required and is not included in the NuScale rod ejection methodology. Core coolability is also ensured by demonstrating that any cladding ballooning that occurs due to overheating caused by exceeding the departure from nucleate boiling ratio (DNBR) limit (for initial power levels >5%) or by exceeding the 170/150 cal/gm enthalpy limit (for zero initial power level), along with cladding internal pressure exceeding the reactor coolant system pressure, is acceptable.

7.3 Rod Ejection Analysis Codes

7.3.1 [[ ]]3(b)

[[

]]3(b)

7.3.2 [[ ]]3(b)

[[

]]3(b)

7.3.3 [[ ]]3(b)

[[

]]3(b)

7.4 [[ ]]3(b)

[[

]]3(b)

[[

]]3(b)

Figure 7-1. [[ ]]3(b)

[[

]]3(b)

[[

]]3(b)

Figure 7-2. [[ ]]3(b)

[[

]]3(b)

[[

]]3(b)

Figure 7-3. [[ ]]3(b)

[[

]]3(b)

[[

]]3(b)

Figure 7-4. [[ ]]3(b)

7.4.1 [[ ]]3(b)

[[

]]3(b)

7.4.2 [[ ]]3(b)

[[

]]3(b)

7.4.3 [[ ]]3(b)

[[

]]3(b)

[[

]]3(b)

Figure 7-5. [[ ]]3(b)

[[

]]3(b)

7.5 [[ ]]3(b)

7.5.1 [[ ]]3(b)

[[

]]3(b)

7.5.2 [[ ]]3(b)

[[

]]3(b)

[[

]]3(b)

Figure 7-6. [[ ]]3(b)

[[

]]3(b)

8.0 REFERENCES

8.1 Referenced Documents

8.1.1 NuScale Power, LLC, "NuScale Plant Design Overview," NP-ER-0000-1198-R0.

8.1.2 SIMULATE Base Model Core Design for NuScale Power Reactor, NP-EC-A021-1510-RA.

8.1.3 U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," NUREG-0800, Chapter 15.0, Revision 3, 2007.

8.1.4 The RELAP5© Code Development Team, "RELAP5© Code Manual Volume IV: Models and Correlations," INEEL-EXT-98-00834, Revision 4.0, June 2012.

8.1.5 U.S. Nuclear Regulatory Commission, "Transient and Accident Analysis Methods," Regulatory Guide 1.203, December 2005.

8.1.6 NuScale Power, LLC, "NuScale Module Small-Break Loss-of-Coolant Accident Phenomena Identification and Ranking Table," NP-TR-0610-289-R1.

8.1.7 NuScale Power, LLC, "Development and Maintenance of Level B Software," NP-EP-0303-321.

8.1.8 *U.S. Code of Federal Regulations,* "ECCS Evaluation Models," Appendix K to Part 50, Chapter 1, Title 10, "Energy," (10 CFR 50).

8.1.9 NuScale Power, LLC, "The Dynamical System Scaling (DSS) Methodology," NP-TR-1010-867-R1.

8.1.10 Shieh, A. S., V. H. Ransom, "RELAP5/MOD3 Code Manual Volume 6: Validation of Numerical Techniques in RELAP5/MOD3.0," Nuclear Systems Analysis Operations, Information Systems Laboratories, Inc., October 2010.

8.1.11 Rowe, D. S., "COBRA IIIC: A Digital Computer Program for Steady State and Transient Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements," BNWL-1695, Battelle, Pacific Northwest Laboratories, Richland, WA, March 1973.

8.1.12 *U.S. Code of Federal Regulations*, "General Design Criteria for Nuclear Power Plants," Appendix A to Part 50, Chapter 1, Title 10, "Energy," (10 CFR 50, Appendix A).

8.1.13 Wheeler, C. L., C. W. Stewart, R. J. Cena, D. S. Rowe, and A. M. Sutey, "COBRA-IV-I: An Interim Version of COBRA for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores," BNWL-1962, Battelle, Pacific Northwest Laboratories, Richland, WA, March 1976.

8.1.14 NuScale Power, LLC, "Non-LOCA PIRT for the NuScale Power Module," NP-RP-0304-210-RA.

8.1.15 Marinelli V., L. Pastori (CNEN-Italy), and B. Kjellen (AB Atomenergi-Sweden), "Experimental Investigation on Mass Velocity Distribution and Velocity Profiles in an LWR Rod Bundle," *Transactions of the American Nuclear Society*, Vol. 15, pp. 413-415, 1972.

8.1.16 Creer, J. M., D. S. Rowe, J. M. Bates, and A. M. Sutey, "Effects of Sleeve Blockages on Axial Velocity and Intensity of Turbulence in an Unheated 7×7 Rod Bundle," BNWL-1965, Battelle Pacific Northwest Laboratories, Richland, WA, January 1976.

8.1.17 Chelemer, H., P. T. Chu, and L. E. Hochreiter, "THINC-IV: An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, Westinghouse Electric Corporation, Pittsburgh, PA, June 1973.

8.1.18 Marshall, R. C. and R. P. Letendre, "Influence of Inlet Geometry on Flow in the Entrance Region of a Nuclear Reactor Rod Bundle," ASME Paper 68-WA/HT-34, *American Society of Mechanical Engineers*, New York, NY, 1968.

8.1.19 Quigley, M. S., C. A. McMonagle, and J. M. Bates, "Investigation of Combined Free and Forced Convection in a 2x6 Rod Bundle," BNWL-2216, Battelle Pacific Northwest Laboratories, Richland, WA, July 1977.

8.1.20 Bates, J. M. and E. U. Khan, "Investigation of Combined Free and Forced Convection in a 2x6 Rod Bundle During Controlled Flow Transients," PNL-3135, Battelle Pacific Northwest Laboratories, Richland, WA, October 1980.

8.1.21 Lahey, R. T., Jr., B. S. Shiralkar, and D. W. Radcliffe, "Two-Phase Flow and Heat Transfer in Multirod Geometries: Subchannel and Pressure Drop Measurements in a Nine-Rod Bundle for Diabatic and Adiabatic Conditions," GEAP-13049, General Electric Company, San Jose, CA, March 1970.

8.1.22 Lahey, R. T., Jr., B. S. Shiralkar, D. W. Radcliffe, and E. E. Polomik, "Out-of-Pile Subchannel Measurements in a Nine-Rod Bundle for Water at 1000 psia," *Progress in Heat and Mass Transfer*, Vol. 6, pp. 345-363, Pergamon, London, 1972.

8.1.23 Herkenrath H., W. Hufschmidt, U. Jung, and F. Weckermann, "Experimental Investigation of the Enthalpy and Mass Flow Distribution in 16-Rod Clusters with BWR-PWR Geometries and Conditions," Final Report EUR 7575 EN, Joint Research Center Ispra, Italy, 1981.

8.1.24 Gustafsson, B., R. Harju, and O. Imset, "Flow and Enthalpy Distribution in a 9-Rod Bundle," European Two-Phase Flow Group Meeting, Harwell, England, June 3-7, 1974.

8.1.25 Castellana, Frank S. and Joseph E. Casterline, "Subchannel Flow and Enthalpy Distributions at the Exit of a Typical Nuclear Fuel Core Geometry," *Nuclear Engineering and Design*, Vol. 22, pp. 3-18, 1972.

8.1.26 Nuclear Energy Agency, *NEA/NRC Benchmark based on NUPEC BWR Full-size Fine-mesh Bundle Tests (BFBT)*, December 13, 2012, http://www.oecd-nea.org/science/wprs/egrsltb/BFBT/.

8.1.27 Nuclear Energy Agency, *Benchmark based on NUPEC PWR Sub-channel Bundle Tests*, December 14, 2012, http://www.oecd-nea.org/science/wprs/egrsltb/PSBT/.

8.1.28 Nylund, O., et al., "FRIGG Loop Project: Hydrodynamic and Heat Transfer Measurements on a Full-Scale Simulated 36-Rod Marviken Fuel Element with Uniform Heat Flux Distribution," FRIGG-2, R4-447/RTL-1007, ASEA and AB Atomenergi, Stockholm, Sweden, 1968.

8.1.29 Nylund, O., et al., "FRIGG Loop Project: Hydrodynamic and Heat Transfer Measurements on a Full-Scale Simulated 36-Rod Marviken Fuel Element with Nonuniform Heat Flux Distribution," FRIGG-4, ASEA and AB Atomenergi, Stockholm, Sweden, 1968.

8.1.30 Christensen, H., "Power-to-Void Transfer Functions," ANL-6385, Argonne National Laboratory, Argonne, IL, 1961.

8.1.31 Martin, R., "Measurement of the Local Void Fraction at High Pressure in a Heating Channel," *Nuclear Science and Engineering*, Vol. 48, pp. 125-138, 1972.

8.1.32 "Subcooled Boiling Data from Rod Bundles," 1003383, EPRI, Palo Alto, CA, September 2002.

8.1.33 Rohsenow, W. M. and J. M. Clark, "Heat Transfer and Pressure Drop Data for High Heat Flux Densities in Water at High Sub-Critical Pressures," Institute of Mechanical Engineers, 1951.

8.1.34 Bennett, J. A. R., et al., "Heat Transfer to Two-Phase Gas-Liquid Systems; Part 1: Steam-Water Mixtures in the Liquid Dispersed Region in an Annulus," AERE-R-3159, 1956.

8.1.35 Schrock, V. E. and L. M. Grossman, "Forced Convection Boiling Studies," Forced Convection Vaporization Project - Final Report 73308, UCX2182, University of California, Berkeley, November 1959.

8.1.36 Bennett, A. W., et al., "Heat Transfer to Steam-Water Mixture Flowing in Uniformly Heated Tubes in Which the Critical Heat Flux has Been Exceeded," AERE-R5373, 1967.

8.1.37 Shiralkar, B. S., E. E. Polomik, R. T. Lahey, Jr., et al., "Transient Critical Heat Flux - Experimental Results," GEAP-13295, General Electric Company, San Jose, CA, April 1972.

8.1.38 Matzner, B. and J. E. Casterline, "The Effect of Length and Pressure on the Critical Heat Flux for a Closely Spaced 19-Rod Bundle in Forced Convection Boiling," TID-22539, Columbia University, New York, December 1965.

8.1.39 General Electric Company, "Deficient Cooling, 12th Quarterly Progress Report, April 1 - June 30, 1972," GEAP-10221-12, San Jose, CA, July 1972.

8.1.40 Weisman, J., et al., "Experimental Determination of the Departure from Nucleate Boiling in Large Rod Bundles at High Pressures," *Chemical Engineering Progress Symposium*, Series 82, Vol. 64, AIChE Heat Transfer Conference, Seattle, WA, 1968.

8.1.41 Rosal, E. R., et al., "High Pressure Rod Bundle DNB Data with Axially Nonuniform Heat Flux," *Nuclear Engineering and Design*, Vol. 31, pp. 1-20, 1974.

8.1.42 Holman, J. P., "Heat Transfer," 3rd ed., McGraw-Hill, New York, 1963; V. S. Arpaci, "Conduction Heat Transfer," Addison-Wesley, Reading, MA, 1966.

8.1.43 Lanning, D. D. and E. R. Bradley, "Irradiation History and Interim Post Irradiation Data for Instrumented Fuel Assembly (IFA)-432," NUREG/CR-3071, PNL-4543, Battelle Pacific Northwest Laboratories, Richland, WA, 1984.

8.1.44 Stewart, C. W., J. M. Cuta, S. D. Montgomery, J. M. Kelly, K. L. Basehore, T. L. George, and D. S. Rowe (Battelle, Pacific Northwest Laboratories); G. C. Gose and J. L.

Westacott (Computer Simulation & Analysis, Inc.), "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, Volume 2: User's Manual," EPRI NP-2511-CCM-A, Volume 2, Revision 4.4, Electric Power Research Institute, Palo Alto, CA, February 2011.

8.1.45 Studsvik Scandpower CMS Training Course for NuScale Power Inc., June 25-29, 2012, Training Course Notes. SSP-07/431, CASMO-5 User Manual, Revision 4, March 2012.

8.1.46 CASMO-5 Methodology Manual, SSP-08/405 Rev 1, December 2011.

8.1.47 SIMULATE-5 User Manual, SSP-10/438 Rev 3, January 2012.

8.1.48 SIMULATE-5 Methodology Manual, SSP-10/465 Rev 2, June 2011.

8.1.49 CMSLINK5 User's Manual, SSP -10/437 Rev 1 January 2012.

8.1.50 SIMULATE-3K Models & Methodology, SSP-98/13 Rev 7, July 2011.

8.1.51 X-5 Monte Carlo Team, "MCNP - A General Monte Carlo N-Particle Transport Code, Version 5", LA-CP-03-0245, Los Alamos National Laboratory, April, 2003, Revised February, 2008.

8.1.52 Organization for Economic Cooperation and Development, "International Handbook of Evaluated Criticality Safety Benchmark Experiments," NEA/NCS/DOC(95)03, September 2010 Edition.

8.1.53 Organization for Economic Cooperation and Development, "International Handbook of Evaluated Reactor Physics Benchmark Experiments," NEA/NCS/DOC(2006)1, March 2010 Edition.

8.1.54 Lewman, L.W., "Urania-Gadolinia: Nuclear Model Development and Critical Experiment Benchmark," B&W 1810, DOE/ET/34212-41, Babcock & Wilcox, 1984.

8.1.55 Baldwin, M.N. and M. E. Stern, "Physics Verification Program, Part III, Task 4, Summary Report," Babcock & Wilcox Report BAW-3647-20, March 1971.

8.1.56 Kurosawa,M., Naito,Y., Sakamoto,H., and Kaneko,T., "The Isotopic Compositions Database System on Spent Fuels in Light Water Reactors (SFCOMPO)," JAERI-Data/Code 96-036, Japan Atomic Energy Research Institute, February 1997.

8.1.57 U.S Nuclear Regulatory Commission, "Standard Review Plan, Spectrum of Rod Ejection Accidents (PWR)," NUREG-0800, Section 15.4.8, Revision 3, March 2007.

8.1.58 U.S Nuclear Regulatory Commission, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactor," Regulatory Guide 1.77, May 1974.

8.1.59 U.S Nuclear Regulatory Commission, "Standard Review Plan, Interim Acceptance Criteria and guidance for the Reactivity Initiated Accidents" NUREG-0800, Section 4.2, Appendix B, Revision 3, March 2007.

8.1.60 U.S Nuclear Regulatory Commission, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Regulatory Guide 1.183, July 2000.

8.1.61 *U.S. Code of Federal Regulations*, "Accident source term," Section 50.67, Part 50, Chapter 1, Title 10, "Energy," (10 CFR 50.67).

8.1.62 American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section III, "Nuclear Power Plant Components, 2007, New York, NY.

8.1.63 UFSAR Chapter 15 Transient Analysis Methodology, DPC-NE-3005-A, Revision 3b, Duke Energy, July 2009.

8.1.64 Multi-Dimensional Reactor Transients and Safety Analysis Physics Parameters Methodology, DPC-NE-3001-A, Revision 0a, Duke Energy, May 2009.

8.1.65 Finnemann, H., H. Bauer, A. Galati and R. Martinelli, "Results of LWR Core Transient Benchmarks," NEA/NSC/DOC(93)25, OECD Nuclear Energy Agency, October 1993.

8.1.66 Studsvik Scandpower, Inc. "LWR Core Reactivity Transients. SIMULATE-3K Models and Assessments," SSP-04/443, Revision 3, July 2011.

8.1.67 McCardell, R.K., D. L. Herbon and J. E. Houghtaling, "Reactivity Accident Tests Results and Analyses for the SPERT III E-Core A Small Oxide-Fueled, Pressurized Water Reactor," IDO-17281, March 1969.

8.1.68 Studsvik Scandpower, Inc., "Qualification of CASMO-5 / SIMULATE-3K Against the SPERT-III E-Core Cold Start-Up Experiments," PHYSOR-2012, April 2012.