NuScale Module Small-Break Loss-of-Coolant Accident Phenomena Identification and Ranking Table

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Nonproprietary

Panel Members:

Steve Congdon (General Electric, retired)

Tom George (Zachry/NAI)

Craig Peterson (CSA, Inc.)

Jose Reyes (NuScale Power, LLC)

Gregg Swindlehurst (GS Nuclear Consulting, LLC)

Graham Wallis, Chair (Creare, Inc.)

Facilitator:

Kent Welter (NuScale Power, LLC)

Assistant Facilitator:

Tristan McDonald (NuScale Power, LLC)

NuScale Power, LLC

1100 NE Circle Blvd Suite 350

Corvallis, Oregon 97330

www.nuscalepower.com

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Executive Summary

This document describes the development of the small-break loss-of-coolant accident (SBLOCA) phenomena identification and ranking table (PIRT) for the NuScale Power, LLC (NuScale) passive modular integral power reactor. The purpose of this PIRT is to provide an assessment of the relative importance of phenomena that may occur in the NuScale module during accident conditions in relation to specified figures of merit. This assessment is part of the process prescribed by Regulatory Guide 1.203 (Reference 7.1.1) and will support development of a detailed evaluation model to serve as the calculational framework for analysis of SBLOCA scenarios in accordance with the acceptance criteria of 10 CFR 50.46 (Reference 7.1.2).

The SBLOCA PIRT involved convening a panel comprised of industry experts, the NuScale chief technical officer, and NuScale staff to facilitate in its development for two SBLOCA scenarios. The panel received a briefing on the NuScale design, both SBLOCA sequence of events, and a computer code prediction of the response of the NuScale module for the SBLOCA scenarios. The panel then followed the PIRT process by first identifying the structures, systems, and components (SSC) of the module that were associated with the SBLOCA scenario. The SBLOCA scenario was then separated into phases with each phase representing a distinct process-dominated time period. Figures of merit were then selected for each phase. Specifically, the figures of merit were chosen to be quantifiable measures of the system's potential to meet regulatory safety limits. Phenomena were identified for each SSC for each phase, and the phenomena were ranked considering their level of importance relative to the figures of merit. The panel established a knowledge ranking for each of the phenomena for each phase as well.

A key insight from the PIRT process is to identify the high importance phenomena that have a low knowledge level and, or, high uncertainty. After analysis of the PIRT results from both scenarios investigated the following phenomena fall into this category:

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These high importance and low knowledge level phenomena were used by NuScale to support the development of the NuScale SBLOCA evaluation model, and to determine the need for additional separate effects and scaled integral testing. All of the results of the PIRT provide valuable information for consideration during design activities and analytical model development.

1.0 Introduction

1.1 Purpose

This document describes the development of the small-break loss-of-coolant accident (SBLOCA) phenomena identification and ranking table (PIRT) for the NuScale Power, LLC (NuScale) passive modular integral power reactor. Two scenarios were addressed in development process of the PIRT. SBLOCA scenario 1 was initiated by the inadvertent actuation of one of the RVVs on top of the reactor vessel and SBLOCA scenario 2 was initiated by the inadvertent opening of one of the RRVs. The purpose of this PIRT is to provide an assessment of the relative importance of phenomena that may occur in the NuScale module during accident conditions in relation to specified figures of merit. This assessment is part of the process prescribed by Regulatory Guide 1.203 (Reference 7.1.1) and will support development of a detailed evaluation model to serve as the calculational framework for analysis of SBLOCA scenarios in accordance with the acceptance criteria of 10 CFR 50.46 (Reference 7.1.2).

A preliminary LOCA PIRT was conducted in 2008. The results of the preliminary LOCA PIRT are included as Appendix B, and the resumes of the preliminary PIRT panel members are included in Appendix C. The 2008 reactor design included features that are no longer present or have been modified in the current design including:

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The current SBLOCA PIRT addresses all of the above design changes and was not guided or influenced by the results of the preliminary LOCA PIRT. The process followed was essentially the same. The results of the preliminary LOCA PIRT supplement the new results and remain of value to NuScale except for the parts that are no longer applicable due to the above design changes. Future design activities and SBLOCA evaluation model development will consider the results of the PIRTs with focus on the high importance and low knowledge level phenomena.

1.2 Scope

This PIRT is applicable to the SBLOCA transient behavior in the NuScale passive modular integrated reactor design, which incorporates all components of the primary system inside a single reactor vessel and does not require active means of circulating the primary coolant.

1.3 Abbreviations and Definitions

Table 1-1. Abbreviations and definitions

Term	Definition
CCFL	counter-current flow limitation
CFR	Code of Federal Regulations

Term	Definition
CHF	critical heat flux
CHRS	containment heat removal system
CVCS	chemical and volume control system
DHRHX	decay heat removal heat exchanger
DHRS	decay heat removal system
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
ECCS	emergency core cooling system
KL	knowledge level
LBLOCA	large-break loss-of-coolant accident
LOCA	loss-of-coolant accident
LWR	light water reactor
NRC	Nuclear Regulatory Commission
PIRT	phenomena identification and ranking table
PWR	pressurized water reactor
RCS	reactor coolant system
RPV	reactor pressure vessel
RRV	reactor recirculation valve
RVV	reactor vent valve
SBLOCA	small-break loss-of-coolant accident
SSC	systems, structures, and components

2.0 Facility Description

2.1 General Information

The NuScale module (Figure 2-1) is an integrated light water reactor (LWR) with a power rating of approximately 45 MWe (160 MWt). The module contains passive safety features with the pressurizer, steam generator, hot leg riser, cold leg, and core fully integrated into the reactor pressure vessel. A steel containment vessel envelopes the reactor pressure vessel. The containment vessel is partially evacuated during power operation and is capable of withstanding high pressures during accident conditions. The entire module and containment are submerged in a pool of water. The reactor building pool is a stainless steel-lined concrete pool shared by all of the operating modules. The module is covered by an individual concrete biological shield, and all of the modules and pool are enclosed in a single confinement building.

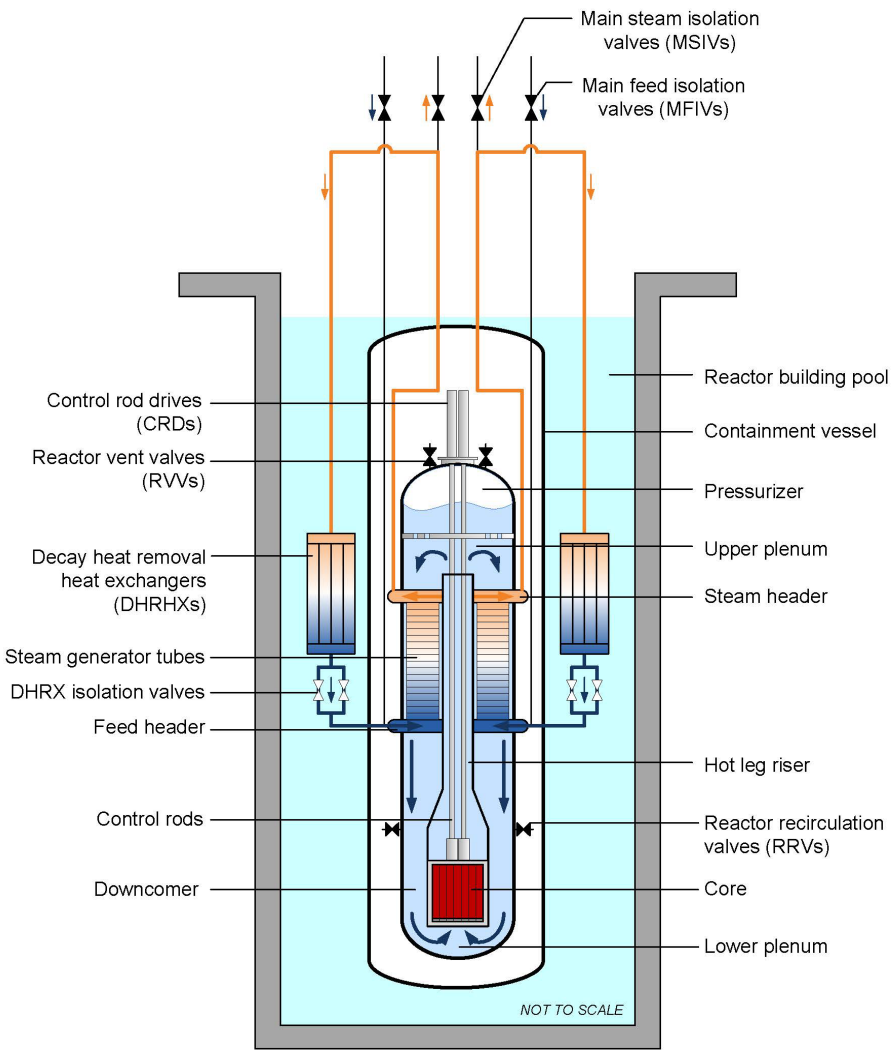

Figure 2-1. NuScale module

The NuScale design relies on passive safety systems and incorporates all large piping paths into the reactor vessel. The use of passive safety systems for decay heat removal, emergency core cooling, and containment cooling eliminates external power requirements under accident conditions. The NuScale modules, control room, and spent fuel pool are all located below grade and housed in controlled-access buildings.

The core is located inside a shroud connected to the hot leg riser. Subcooled water enters the core, where it is heated and then flows vertically into the riser section. Circulation continues as hot water exits the riser into the upper plenum and then turns downward into the annulus housing the steam generators. Hot water in the annulus between the riser and the inside wall of the reactor vessel is cooled by the steam generator tubes. The cooled and higher density water descends through the downcomer into the lower plenum, then re-enters the core.

2.1.1 Steam Generator

The NuScale steam generator is a helical coil, once-through heat exchanger located in the annular space between the hot leg riser and the reactor vessel inside wall. Feedwater enters the tubes at the bottom and superheated steam exits at the top. Two redundant, independent sets of steam generator tube banks occupy the steam generator region.

2.1.2 Pressurizer

The pressurizer, located in the upper head of the reactor pressure vessel, provides reactor coolant system (RCS) pressure control. It is designed to operate in conjunction with the system and the chemical volume and control system (CVCS) to control RCS temperature and pressure during heatup/cooldown and during power transitions, as well as to maintain a constant pressure during steady-state power operation.

A baffle region is located above the steam generator region to provide a barrier between the saturated fluid within the pressurizer and the subcooled RCS fluid. This baffle region limits the temperature of fluid that may surge into or out of the pressurizer region by mixing and heating the fluid.

2.1.3 Nuclear Core

The NuScale module nuclear core consists of 37 fuel assemblies arranged in a 17 x 17 square array**.** The core includes 16 control rod clusters. Each fuel assembly includes 264 fuel pins, 24 control rods, and one instrument tube. The active fuel region is two meters in height.

2.1.4 Containment Vessel

The steel containment vessel is an engineered safety feature that is dry and partially evacuated under normal operating conditions. This configuration minimizes moisture problems that could cause component corrosion and impact the reliability of instrumentation and other systems within containment. The partial vacuum reduces heat transfer from the reactor pressure vessel without the use of conventional insulation during operations. Due to a lack of appreciable amounts of air, the partial vacuum also enhances steam condensation rates on the vessel following a SBLOCA and reactor vent valve (RVV) actuations, and reduces the potential of critical combustible concentrations of hydrogen and oxygen mixtures in the event of a severe accident.

2.1.5 Decay Heat Removal System

The DHRS, a passive engineered safety feature, transfers decay heat from the reactor to the reactor pool via the steam generators. The DHRS provides cooling during transients and accidents that result in a loss of normal feedwater. It has two independent trains, each capable of sufficient heat transfer to prevent potential fuel damage.

The decay heat removal heat exchanger (DHRHX) is submerged in the reactor pool. The inlet to the DHRS is connected to the main steam line. Valves at the exit of the DHRHX automatically open to actuate the system to return condensate back to the steam generator feedwater header. Steam produced in the steam generators is condensed in the DHRHX and the condensate returns to the inlet of the steam generator in a closed loop mode of operation. For DHRS operation without a LOCA or opening of the RVVs, similar to normal operation, the primary coolant is cooled by convection heat transfer as it flows over the steam generator tubes. The steam generator removes heat from the reactor coolant in the reactor vessel annulus, creating a density difference between the hotter, lower-density coolant inside the riser and the cooler, higher-density coolant in the downcomer. This density difference causes natural circulation of the reactor coolant in the same manner as during normal operation, but at a reduced flow rate. For DHRS operation with a LOCA or following actuation of the RVVs, the RCS inventory is reduced, and the primary coolant is cooled initially by convection and then later by condensation on the steam generator tubes. The condensate flows downward and returns to the reactor vessel lower plenum for cooling the core.

2.1.6 Containment Heat Removal System

Following a postulated break in the primary or secondary systems or an actuation of the RVVs, steam released into the evacuated containment vessel is condensed on the inside surface of the containment vessel. The containment heat removal system (CHRS) then transfers the energy to the reactor building pool by conduction and convection heat transfer modes. The CHRS, a passive engineered safety feature, consists of the reactor building pool and the evacuated containment vessel which enhances condensation. The reactor building pool consists of a large, below-grade stainless steel lined concrete pool that is designed to provide cooling of the containment vessels for at least 72 hours following any design basis event without any active heat removal from the pool. Following a postulated break in the primary or secondary systems, steam released into the containment would be condensed on the inside surface of the containment vessel, which is cooled by conduction and convection heat transfer to the reactor building pool.

2.1.7 Emergency Core Cooling System

The ECCS, an engineered safety feature, consists of two independent RVVs, two independent reactor recirculation valves (RRVs), and the CHRS. The ECCS provides a means of core decay heat removal following a SBLOCA. The ECCS is automatically initiated by the RPS causing the opening of the RVVs to create a release path for the primary coolant to flow into the containment vessel. This release path is in addition to the SBLOCA release path. The primary coolant collects at the bottom of the containment vessel after the steam phase condenses on the containment vessel. After a sufficient depth of water accumulates in the containment vessel the RRVs automatically open. Opening the RRVs creates a flow path for the water in the containment vessel to flow into the downcomer of the reactor vessel. This establishes a natural circulation loop whereby water that is boiled in the core flows up through the RVVs, is condensed and collected in the containment vessel, and is then returned through the RRVs into the downcomer for stable long-term core cooling.

3.0 SBLOCA Scenario Description

3.1 Scenario Selection

Regulations require that all potential LOCA break sizes and locations be analyzed when evaluating the performance of the ECCS for compliance with 10 CFR 50.46 (Reference 7.1.2). Because of the large range of break sizes and locations in a standard pressurized water reactor, system processes and related phenomena that are present at one end of the break spectrum are not present at the other end. Typically, separate LOCA evaluation methodologies are licensed for large-break loss-of-coolant-accidents (LBLOCAs) and for SBLOCAs.

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The RPV and containment pressure response is shown in Figure 3-1. This pressure response curve is characteristic of any blowdown situation where the RPV and containment pressures eventually equilibrate.

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Figure 3-1. RVV LOCA RPV and containment pressure response

3.2 Sequence of Events

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Table 3-1. Sequence of events – Scenario 2 RRV SBLOCA

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4.0 PIRT Process

The PIRT process is a systematic way of gathering information from experts on a specific subject and ranking the importance of information to meet decision-making objectives. It has been applied to many nuclear technology issues to help guide research and development activities to satisfy regulatory requirements.

The PIRT process can be broken into nine distinct steps:

1. Issues
2. Objectives
3. Database
4. Hardware and scenario
5. Figures of merit
6. Phenomena identification
7. Importance ranking
8. Knowledge-level ranking
9. Documentation

A high-level description of the PIRT process is provided in Sections 4.2 through 4.10.

4.1 PIRT Panel Biographies

The following individuals participated either as panel members, a panel chair, or panel facilitators in the NuScale SBLOCA PIRT development. Biographical information for each participant is provided in Appendix A.

- Mr. Steve Congdon (Member)
- Dr. Tom George (Member)
- Mr. Craig Peterson (Member)
- Dr. José N. Reyes (Member)
- Mr. Gregg Swindlehurst (Member)
- Dr. Graham Wallis (Chair)
- Dr. Kent B. Welter (Facilitator)
- Mr. Tristan McDonald (Assistant Facilitator)

4.2 Step 1 – Issues

The issues to be addressed by this PIRT were the identification of the attributes of the design and the response of the NuScale module following a SBLOCA that may affect nuclear safety and may require additional analytical or experimental information.

4.3 Step 2 – Objectives

The objectives of the SBLOCA PIRT were to:

- Establish the figures of merit

- Identify the phenomena important to nuclear safety
- Rank the phenomena relative to the figures of merit
- Rank the knowledge level for each phenomenon
- Determine the high-importance/low knowledge level phenomena to focus the development of the SBLOCA evaluation model and to determine additional design and testing requirements

4.4 Step 3 – Database

The database for the SBLOCA PIRT consisted of the following:

- The list of phenomena developed by the preliminary LOCA PIRT panel in 2008
- A presentation on the design of the NuScale module
- A discussion of the response of the NuScale module to the SBLOCA Scenario 1, which was a spurious opening of one RVV at full power
- A presentation of a GOTHIC code simulation of the response of the NuScale module to the SBLOCA Scenario 2, which was a spurious opening of one RRV at full power

4.5 Step 4 – Hardware and Scenario

The hardware for the PIRT is the NuScale module as described in Section 2.0. The PIRT panel evaluated two SBLOCA scenarios. {{

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4.6 Step 5 – Figures of Merit

The PIRT panel established the figures of merit described in Table 4-1 for use in ranking the phenomena for each phase and for each system, structure, or component.

Table 4-1. Figures of merit

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4.7 Step 6 – Phenomena Identification

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Table 4-2. Systems, structures, components, and processes

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4.8 Step 7 – Importance Rankings

The PIRT panel then ranked the phenomena relative to the figures of merit for each of the three phases as applicable. Each phenomenon was assigned one of the importance rankings described in Table 4-3 below.

Table 4-3. Importance rankings

Importance Ranking	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

4.9 Step 8 – Knowledge Level Rankings

The PIRT panel assessed and ranked the current knowledge level for all phenomena except those ranked inactive (I). Table 4-4 presents the numerical scale used for knowledge levels.

Table 4-4. Knowledge levels

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

4.10 Step 9 – Documentation

The documentation of the PIRT consists mainly of this report. Additional documents provided to the PIRT team are archived in the NuScale document system.

4.11 PIRT Results for Scenario 1 – RVV Failure SBLOCA

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Table 4-5. NuScale module SBLOCA PIRT results – Scenario 1 RVV SBLOCA

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4.12 PIRT Results for Scenario 2 – RRV Failure SBLOCA

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Table 4-6. NuScale module SBLOCA PIRT results – Scenario 2 RRV failure

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5.0 Discussion of Phenomena and Ranking Rationales

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5.1 Containment Vessel Phenomena

5.1.1 Flow Through RVV

Note: This phenomenon is the mass and energy release entering the containment vessel, resulting from a SBLOCA or from ECCS actuation of the RVV. The phenomena in the RVV piping are discussed in Section 5.2.3.

5.1.1.1 Description of Phenomena

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5.1.2 Flow Through RRV

Note: This phenomenon is the Scenario 2 – RRV SBLOCA mass and energy release.

5.1.2.1 Description of Phenomena

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5.1.2.2 PIRT Panel Importance and Knowledge Level Ranking

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5.1.3 Water Films on Surfaces with Condensation

5.1.3.1 Description of Phenomena

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5.1.3.2 PIRT Panel Importance and Knowledge Level Ranking

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5.1.4 Liquid Level

5.1.4.1 Description of Phenomena

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5.1.4.2 PIRT Panel Importance and Knowledge Level Ranking

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5.1.4.3 Scenario 2 – RRV SBLOCA Differences

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5.1.5 Containment Vessel Heat Transfer

5.1.5.1 Description of Phenomena

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5.1.5.2 PIRT Panel Importance and Knowledge Level Ranking

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5.1.6 Concentration/Combustion of Radiolytic Gases

5.1.6.1 Description of Phenomena

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5.1.6.2 PIRT Panel Importance and Knowledge Level Ranking

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5.1.7 Heat Transfer To/From/In Reactor Pressure Vessel

5.1.7.1 Description of Phenomena

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5.1.7.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2 Reactor Pressure Vessel

5.2.1 Pressurizer

5.2.1.1 Water Holdup/Draining/CCFL

5.2.1.1.1 Description of Phenomena

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5.2.1.1.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.1.2 Flashing

5.2.1.2.1 Description of Phenomena

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5.2.1.2.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.1.3 Level Swell

5.2.1.3.1 Description of Phenomena

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5.2.1.3.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.2 RRV Pressure Drop/Flow

Note: This phenomenon is the pressure drop and recirculation flow through the RRV after the ECCS actuates the RRV.

5.2.2.1 Description of Phenomena

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5.2.2.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.2.3 Scenario 2 – RRV SBLOCA Differences

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5.2.3 RVV Flow/Pressure Drop in Piping

5.2.3.1 Description of Phenomena

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5.2.3.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.4 Integral Phenomena

5.2.4.1 Natural Circulation

5.2.4.1.1 Description of Phenomena

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5.2.4.1.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.4.2 Interrupted Natural Circulation

5.2.4.2.1 Description of Phenomena

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5.2.4.2.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.4.2.3 Scenario 2 – RRV SBLOCA Differences

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5.2.4.3 Intermittent Natural Circulation

5.2.4.3.1 Description of Phenomena

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5.2.4.3.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.5 Reactor Core

5.2.5.1 3D Flow Distribution

5.2.5.1.1 Description of Phenomena

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5.2.5.1.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.5.2 Void Distribution

5.2.5.2.1 Description of Phenomena

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5.2.5.2.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.5.2.3 Scenario 2 – RRV SBLOCA Differences

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5.2.5.3 Flow Stagnation

5.2.5.3.1 Description of Phenomena

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5.2.5.3.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.5.3.3 Scenario 2 – RRV SBLOCA Differences

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5.2.5.4 Flashing

5.2.5.4.1 Description of Phenomena

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5.2.5.4.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.5.4.3 Scenario 2 – RRV SBLOCA Differences

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5.2.5.5 Boiling

5.2.5.5.1 Description of Phenomena

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5.2.5.5.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.5.5.3 Scenario 2 – RRV SBLOCA Differences

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5.2.5.6 Stored Energy in Fuel Rods/Fuel Pin Properties

5.2.5.6.1 Description of Phenomena

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5.2.5.6.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.5.7 Decay Heat/Delayed Fission Power

5.2.5.7.1 Description of Phenomena

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5.2.5.7.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.5.8 Subchannel Spacer Effects/Subcooled Boiling

5.2.5.8.1 Description of Phenomena

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5.2.5.8.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.5.9 Control Rod Insertion

5.2.5.9.1 Description of Phenomena

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5.2.5.9.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.6 Hot Leg Riser

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5.2.6.1 Flashing

5.2.6.1.1 Description of Phenomena

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5.2.6.1.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.6.1.3 Scenario 2 – RRV SBLOCA Differences

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5.2.6.2 Phase Slip/Flow Regimes

5.2.6.2.1 Description of Phenomena

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5.2.6.2.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.6.2.3 Scenario 2 – RRV SBLOCA Differences

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5.2.7 Downcomer

5.2.7.1 3D Flow Distribution/Fluid Mixing/Boron Mixing

5.2.7.1.1 Description of Phenomena

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5.2.7.1.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.7.1.3 Scenario 2 – RRV SBLOCA Differences

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5.2.7.2 Heat Transfer To/From/In Reactor Pressure Vessel

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5.2.8 Lower Plenum

5.2.8.1 3D Flow Distribution/Fluid Mixing/Boron Mixing/Stratification

5.2.8.1.1 Description of Phenomena

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5.2.8.1.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.8.1.3 Scenario 2 – RRV SBLOCA Differences

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5.2.8.2 Heat Transfer To/From/In Reactor Pressure Vessel

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5.2.9 Steam Generator

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5.2.9.1 Shell Side Heat Transfer/Condensation

5.2.9.1.1 Description of Phenomena

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5.2.9.1.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.9.2 Tube Side Boiling

5.2.9.2.1 Description of Phenomena

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5.2.9.2.2 PIRT Panel Importance and Knowledge Level Ranking

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5.2.9.3 Tube Side Flow/Pressure Drop

Note: This section combines single-phase and two-phase flow and pressure drop panel rankings that were identical.

5.2.9.3.1 Description of Phenomena

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5.2.9.3.2 PIRT Panel Importance and Knowledge Level Ranking

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5.3 Decay Heat Removal System

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5.3.1 Initial Inventory/Startup

5.3.1.1 Description of Phenomena

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5.3.1.2 PIRT Panel Importance and Knowledge Level Ranking

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5.3.2 Heat Transfer Across DHRHX Tubes

5.3.2.1 Description of Phenomena

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5.3.2.2 PIRT Panel Importance and Knowledge Level Ranking

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5.3.3 Condensation/Non-Condensable Gas Inside Tube

5.3.3.1 Description of Phenomena

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5.3.3.2 PIRT Panel Importance and Knowledge Level Ranking

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5.3.4 Heat Transfer to the Pool Via DHRHX/Pool Boiling

5.3.4.1 Description of Phenomena

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5.3.4.2 PIRT Panel Importance and Knowledge Level Ranking

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}}3(a)-(c)

5.3.5 Flow/Pressure Drop

5.3.5.1 Description of Phenomena

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}}3(a)-(c)

5.3.5.2 PIRT Panel Importance and Knowledge Level Ranking

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}}3(a)-(c)

5.3.6 Buoyancy/Flow Distribution

5.3.6.1 Description of Phenomena

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5.3.6.2 PIRT Panel Importance and Knowledge Level Ranking

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}}3(a)-(c)

5.3.7 Leakage In/Out of System

5.3.7.1 Description of Phenomena

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5.3.7.2 PIRT Panel Importance and Knowledge Level Ranking

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5.4 Reactor Building Pool

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6.0 Conclusions

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Table 6-1. Importance rankings

Importance Ranking	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 6-2. Knowledge levels

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

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Although this report does not focus on the medium and lower importance phenomena, all of the results presented in the PIRT tables should be considered in the development of the SBLOCA evaluation model and the planning of system and component testing programs.

A preliminary LOCA PIRT was conducted in 2008. The 2008 design included features that are no longer present or have been modified in the current design that is the subject of this report. The current SBLOCA PIRT was not guided or influenced by the results of the preliminary LOCA PIRT. The PIRT process followed was essentially the same. The results of the preliminary LOCA PIRT supplement the new results and remain of value except for those parts that are no longer applicable due to design changes. Future design activities and SBLOCA evaluation model development will consider the results of both PIRTs and focus on the high importance/low knowledge level phenomena.

7.0 References

7.1.1 U.S. Nuclear Regulatory Commission, “Regulatory Guide 1.203, Transient and Accident Analysis Methods,” Regulatory Guide 1.203, Office of Standard Development, Washington, DC, December 2005.

7.1.2 U.S. Code of Federal Regulations, “Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors,” Section 50.46, Part 50, Chapter 1, Title 10 (CFR 50.46).

7.1.3 NuScale Power, LLC, 2008, “Customer Requirements Document,” NP-ER-0301-010-DRAFT. *[Proprietary]*

7.1.4 GOTHIC Thermal Hydraulic Analysis Package, Version 8.0, Volumes 1-4, NAI 8907-02, Rev. 20, January 2012.

Appendix A. SBLOCA PIRT Panel Biographies

Mr. Steve Congdon

Retired from GE Nuclear Energy, Wilmington, NC in 2000 where he served as Manager of Advanced Engineering and Manager of Fuel Technology; 2001-2011 served as rehired pensioner at GE Nuclear Energy, providing assistance in BWR plant monitoring technology, qualification of GE nuclear and thermal hydraulic methods and writing NRC licensing reports supporting GE Power Uprate activities.

Dr. Tom George

Senior Consulting Engineer, Numerical Applications Division Zachry Nuclear Engineering. Dr. George specializes in numerical modeling in engineering mechanics. Primary developer for the GOTHIC code for general-purpose thermal-hydraulic analysis. Participated in wide range of safety analyses using GOTHIC to support vender and utility licensing efforts. Developed a computer program to track smoke propagation and fire growth in buildings. Developed a mechanical response model that predicts the elastic and plastic properties of restructuring sphere pac nuclear fuel and the stress-strain distribution in the fuel and cladding. Developed models for heat and mass transfer. Developed numerical techniques for thermal-hydraulic analysis in three-dimensional curvilinear or generalized coordinate systems for complex geometries.

Mr. Craig Peterson

President of Computer Simulation and Analysis, Inc. in Idaho Falls, ID since 1999; former Nuclear Safety Analysis and Software Applications Manager at Energy Incorporated; over 30 years experience in commercial nuclear power industry working on a wide variety of projects for numerous organizations.

Dr. José N. Reyes

Dr. José N. Reyes is the Henry and Janice Schuette Endowed Chair Professor and Head of the Department of Nuclear Engineering and Radiation Health Physics at Oregon State University (OSU). He is an internationally recognized expert on passive safety system design, testing, and operation for nuclear power plants. He currently serves as a United Nations International Atomic Energy Agency (IAEA) technical expert on passive safety systems. Dr. Reyes successfully established a 17-nation Coordinated Research Program on Passive Safety Systems for the IAEA and also developed and directed a course on natural circulation and passive safety systems at the International Center for Theoretical Physics in Trieste, Italy. Dr. Reyes was the OSU principal investigator for the Westinghouse AP600 and AP1000 certification test programs sponsored by the U.S. Nuclear Regulatory Commission (NRC), U.S. Department of Energy, and Westinghouse.

Prior to joining the faculty at OSU, Dr. Reyes worked 10 years as a thermal hydraulics research engineer in the NRC's Reactor Safety Division. He is the author of numerous technical papers and has given lectures and keynote addresses to professional nuclear organizations in the United States, Europe, and Asia. He received a B.S. degree in Nuclear Engineering from the University of Florida and an M.S. and Ph.D. in Nuclear Engineering from the University of Maryland.

Mr. Gregg Swindlehurst

Consultant at GS Nuclear Consulting, LLC in Charlotte, NC; Former Safety Analysis Manager at Duke Power; 24 years of supervision/management in electric utility nuclear engineering organizations; 30 years of experience with pressurized water reactor transient and accident analysis methodology development and application using RETRAN, RELAP5, VIPRE, GOTHIC,

SIMULATE, and FALCON codes; Served on the McGuire and Catawba corporate Nuclear Safety Review Boards and the subcommittee that reviews all license amendment requests prior to submittal for NRC review; Pioneered the use of engineering-quality transient and accident and containment analysis codes for validating training simulator software.

Dr. Graham Wallis (Chair)

Dr. Graham Wallis served as a member of the Thayer School faculty from 1962 to 2001. He was selected by the American Society of Mechanical Engineers (ASME) to receive the 1994 Fluids Engineering Award "for extensive research in the field of two-phase flow, and for writing on the extension of potential flow theory to two-phase flows." He is an internationally recognized expert on fluid dynamics, two-phase flow, thermodynamics, heat and mass transfer, nuclear power, and energy conversion.

Dr. Wallis is the author of the book, *One-Dimensional Two-Phase Flow,* and over 150 publications and reports on aspects of two-phase flow. He has taught courses in linear systems, fluids, thermodynamics, heat and mass transfer, nuclear reactor design, nuclear reactor engineering, energy conversion, and field theory. He received a B.A. and Ph.D. in Mechanical Engineering from Cambridge University, as well as an M.S. degree in Mechanical Engineering from MIT, and is a Fellow at Trinity College of Cambridge.

Dr. Kent B. Welter (Facilitator)

Prior to joining NuScale Power, LLC, Dr. Welter was Acting Chief of the Code Development Branch/Office of Research at the NRC. He has led teams of thermal-hydraulics and neutronics experts to develop and maintain safety analysis codes and has worked in partnership with universities, laboratories, and other national and international research centers involved in nuclear system safety analysis research and testing. Dr. Kent Welter is a Principal Engineer at NuScale Power where he is the manager of the Safety Analysis Group.

He is the author and co-author of over 20 papers and technical reports. He is the past chair of the American Nuclear Society Young Members Group, Vice-Chair/Chair-Elect of the American Nuclear Society Environmental Sciences Division, and was a representative member of the U.S. Delegation to the United Nations World Summit on Sustainable Development in Johannesburg. In 2007, Dr. Kent Welter received the Scientific Achievement Award from the Oregon Institute of Technology for his contributions and professional involvement in the area of advanced reactor thermal-hydraulics. He received a B.S in Mechanical Engineering Technology from Oregon Institute of Technology and a Ph.D. in Nuclear Engineering from OSU.

Mr. Tristan McDonald (Assistant Facilitator)

Mr. Tristan McDonald has been employed as a Safety Analysis Engineer in the Safety Analysis Group at NuScale Power since 2010 Previously he worked at the Fort Calhoun Nuclear Station in the Reactor Performance Analysis Group. He has experience with the GOTHIC, S-RELAP5, XCOBRA-IIIC, and CFD codes He received a B. S. in Nuclear Engineering from the University of New Mexico in 2007.

Appendix B. Preliminary LOCA PIRT Results Table

Table B-1. Preliminary NuScale module LOCA PIRT results

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Appendix C. Preliminary PIRT Panel Biographies

Dr. Lawrence E. Hochreiter

Dr. Lawrence E. Hochreiter is a former Professor of Nuclear and Mechanical Engineering at the Pennsylvania State University. Dr. Hochreiter spent 26 years working in the Nuclear Energy Systems Divisions at Westinghouse, primarily in the Nuclear Safety area. In 1972, he was appointed Manager of Safeguards Development, where he supervised light water reactor (LWR) safety research as applied to pressurized water reactors (PWRs). These experiments included large full-length rod bundle blowdown film boiling, level swell, and reflood heat transfer tests; NRC/Westinghouse Full-Length Emergency Core Heat Transfer reflooding experiments; scaled steam/water mixing tests; and Westinghouse transient departure from nucleate boiling tests.

Dr. Hochreiter also was responsible for development and integration of the AP600 (an advanced PWR design) safety testing and analysis efforts, which supported AP600 design certification and licensing. He was directly involved in model development, refinement, and validation of the Westinghouse safety analysis computer codes for small-break loss-of-coolant accident (SBLOCA), large-break loss-of-coolant accident (LBLOCA), long-term cooling, and containment analysis for this passive plant design. He developed several of the initial Phenomena Identification and Ranking Tables (PIRTs) for the AP600 LBLOCA and SBLOCA transient and containment analyses. He received a B.S. in Mechanical Engineering from the University of Buffalo, and an M.S. and a Ph.D. in Nuclear Engineering from Purdue University.

Dr. Mujid S. Kazimi

Dr. Mujid S. Kazimi has been a Professor of Nuclear and Mechanical Engineering at Massachusetts Institute of Technology (MIT) since 1976, and served as Head of the Department of Nuclear Science and Engineering there from 1989 to 1997. He is the founding and current Director of the Center for Advanced Nuclear Energy Systems at MIT. He has extensive experience in design and safety analysis of nuclear fission reactors, fusion technology devices, and high-level radioactive waste storage facilities. He received a B.Eng. in Nuclear Engineering from the Alexandria University of Egypt, as well as an M.S. and Sc.D in Nuclear Engineering from MIT.

Dr. Kord S. Smith

As a cofounder of the U.S. branch of Studsvik Scandpower, Dr. Kord S. Smith has engaged in research, development, deployment, marketing, training, and support of reactor physics software tools that are widely used in the commercial LWR industry since 1984. He has helped develop mathematical physics models and computer codes that are capable of performing the steady state and transient physics analysis required for core design, fuel management, safety analysis, and NRC licensing of commercial nuclear reactors. He received a B.S. in Nuclear Engineering from Kansas State University and an M.S. and Sc.D in Nuclear Engineering from MIT.

Dr. José N. Reyes

Dr. José N. Reyes is the Henry and Janice Schuette Endowed Chair Professor and Head of the Department of Nuclear Engineering and Radiation Health Physics at Oregon State University (OSU). He is an internationally recognized expert on passive safety system design, testing, and operation for nuclear power plants. He currently serves as a United Nations International Atomic Energy Agency (IAEA) technical expert on passive safety systems. Dr. Reyes successfully established a 17-nation Coordinated Research Program on Passive Safety Systems for the IAEA and also developed and directed a course on natural circulation and passive safety systems at the International Center for Theoretical Physics in Trieste, Italy. Dr. Reyes was the OSU principal

investigator for the Westinghouse AP600 and AP1000 certification test programs sponsored by the U.S. Nuclear Regulatory Commission (NRC), U.S. Department of Energy, and Westinghouse.

Prior to joining the faculty at OSU, Dr. Reyes worked 10 years as a thermal hydraulics research engineer in the NRC's Reactor Safety Division. He is the author of numerous technical papers and has given lectures and keynote addresses to professional nuclear organizations in the United States, Europe, and Asia. He received a B.S. degree in Nuclear Engineering from the University of Florida and an M.S. and Ph.D. in Nuclear Engineering from the University of Maryland.

Dr. Brent Boyack

Biography not submitted.

Dr. Graham Wallis (Chair)

Dr. Graham Wallis served as a member of the Thayer School faculty from 1962 to 2001. He was selected by the American Society of Mechanical Engineers (ASME) to receive the 1994 Fluids Engineering Award "for extensive research in the field of two-phase flow, and for writing on the extension of potential flow theory to two-phase flows." He is an internationally recognized expert on fluid dynamics, two-phase flow, thermodynamics, heat and mass transfer, nuclear power, and energy conversion.

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Prior to joining NuScale Power, LLC, Dr. Welter was Acting Chief of the Code Development Branch/Office of Research at the NRC. He has led teams of thermal-hydraulics and neutronics experts to develop and maintain safety analysis codes and has worked in partnership with universities, laboratories, and other national and international research centers involved in nuclear system safety analysis research and testing. Dr. Kent Welter is a Principal Engineer at NuScale Power where he is the manager of the Safety Analysis Group.

He is the author and co-author of over 20 papers and technical reports. He is the past chair of the American Nuclear Society Young Members Group, Vice-Chair/Chair-Elect of the American Nuclear Society Environmental Sciences Division, and was a representative member of the U.S. Delegation to the United Nations World Summit on Sustainable Development in Johannesburg. In 2007, Dr. Kent Welter received the Scientific Achievement Award from the Oregon Institute of Technology for his contributions and professional involvement in the area of advanced reactor thermal-hydraulics. He received a B.S in Mechanical Engineering Technology from Oregon Institute of Technology and a Ph.D. in Nuclear Engineering from OSU.

Dr. Eric P. Young (Assistant)

Dr. Eric Young is a Safety Analysis Engineer at NuScale Power, LLC, where he is in charge of development of the LOCA evaluation methodology for the NuScale plant. He received a B.S in Mechanical Engineering Technology from Oregon Institute of Technology and an M.S. and Ph.D. in Nuclear Engineering from OSU.