NuScale Preliminary Loss-of-Coolant Accident (LOCA) Thermal-Hydraulic and Neutronics Phenomena Identification and Ranking Table (PIRT)

March 2011

Revision 1

Nonproprietary

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ABSTRACT

A panel of industry experts has been convened, and the NuScale Power, Inc., (NuScale) design structures, systems, and components have been identified. The level of detail is commensurate with that needed to identify phenomena affecting the system condition with respect to specific figures of merit relating to a small-break loss-of-coolant accident (SBLOCA) in the NuScale design. These figures of merit were identified by dividing the SBLOCA scenario into three periods, each of which represents a distinct process-dominated time frame. Specifically, the figures of merit were chosen because they are quantifiable measures of the system's potential to meet regulatory safety limits, and the phenomena were ranked by considering their level of importance in determining the figures of merit.

This document describes the development of this SBLOCA phenomena identification and ranking table (PIRT) for the NuScale passive integral power reactor. The purpose of this PIRT is to provide an independent assessment of the relative importance of phenomena that may occur in the NuScale module during the postulated accident conditions in relation to specified figures of merit. This independent assessment is part of the process prescribed by Regulatory Guide 1.203 (Reference 1) and will support development of a detailed evaluation model to serve as the calculational framework for analysis of postulated SBLOCA accidents in accordance with the evaluation required by 10 CFR 50.46 (Reference 2).

1.0 DEFINITIONS / ACRONYMS

CFR	Code of Federal Regulations
CHF	Critical Heat Flux
CHRS	Containment Heat Removal System
DHRS	Decay Heat Removal System
ECCS	Emergency Core Cooling System
LBLOCA	Large-Break Loss-of-Coolant Accident
LOCA	Loss-of-Coolant Accident
LWR	Light Water Reactor
MPa	Megapascals
MSIV	Main Steam Isolation Valve
MWe	Megawatts Electric
MWt	Megawatts Thermal
NRC	Nuclear Regulatory Commission
OSU	Oregon State University
PIRT	Phenomena Identification and Ranking Table
psia	pounds per square inch absolute
SBLOCA	Small-Break Loss-of-Coolant Accident
RELAP5	Reactor Excursion and Leak Analysis Program (version 5)
RRV	Reactor Recirculation Valve
RVV	Reactor Vent Valve

2.0 GENERAL INFORMATION

2.1 Purpose

This document describes the development of this small-break loss-of-coolant accident (SBLOCA) phenomena identification and ranking table (PIRT) for the NuScale Power, Inc., (NuScale) passive integral power reactor. The purpose of this PIRT is to provide an independent assessment of the relative importance of phenomena that may occur in the NuScale module during postulated transient and accident conditions in relation to specified figures of merit. This independent assessment is part of the process prescribed by Regulatory Guide 1.203 (Reference 1) and will support development of a detailed evaluation model to serve as the calculational framework for analysis of postulated transients and accidents as part of emergency core cooling performance analysis, as required by 10 CFR 50.46 (Reference 2).

2.2 Scope

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

2.3 Facility Description

2.3.1 General Information

The NuScale module is an integrated light water reactor (LWR) with passive safety features and a power rating of approximately 45 MWe (160 MWt). The pressurizer, steam generator, hot leg, cold leg, and core are all housed in a shared reactor pressure vessel. A relatively small steel containment envelopes the reactor pressure vessel. The containment vessel is partially evacuated during power operation and is capable of relatively high pressures during accident conditions. The entire module and containment are submerged in a pool of water. The reactor pool is a stainless steel-lined concrete pool shared by all of the operating modules. The module is covered by an individual concrete impact shield, and all of the modules and pool are enclosed in a single confinement building. The design is consistent with the latest U.S. Nuclear Regulatory Commission (NRC) Advanced Reactor Design Policy and NuScale's Customer Requirements Document (Reference 3).

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, denser water descends through the downcomer into the lower plenum, then re-enters the core.

2.3.2 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's 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.

2.3.3 Pressurizer

The pressurizer in the NuScale module's Nuclear Steam Supply System provides reactor coolant system pressure control. It is designed to operate in conjunction with the Stable Startup System and the Chemical Volume and Control System to bring primary fluid conditions to a stable operating temperature and pressure from which the system can be brought to power operation, as well as to maintain a constant pressure during 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 reactor coolant system 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 as it moves about this region.

2.3.4 Nuclear Core

The NuScale module's 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.

2.3.5 High-Pressure Containment

NuScale's containment vessel is an engineered safety feature that *is dry and partially evacuated under normal operating conditions*. This configuration eliminates moisture problems that could cause component corrosion and impact the reliability of instrumentation and other systems within containment. The partial vacuum reduces convection heat transfer without the use of "direct-contact" reactor vessel insulation. Due to a lack of appreciable amounts of air, the vacuum also enhances steam condensation rates during reactor vessel blowdowns and prevents the formation of combustible concentrations of hydrogen mixtures in the event of a severe accident.

2.3.6 Decay Heat Removal System

NuScale's decay heat removal system (DHRS), an engineered safety feature, removes residual heat from the reactor core due to decay heat generation. The DHRS provides cooling for the core during normal shutdowns, station blackouts, and/or transients that result in a loss of normal feedwater. It has two independent piping trains, each capable of passively removing a sufficient fraction of the post-trip core power to prevent damage due to system heat-up.

During DHRS operation, cold water from the containment cooling pool is drawn from the inlet screen and sent to the steam generator tubes, where it transfers heat from the primary fluid and is evaporated. This steam is then vented and condensed in the containment pool. 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 riser. This density difference creates natural circulation of the reactor coolant in the same manner as during normal operation, but at a reduced flow rate. The check valves at various points in the DHRS prevent reverse flow.

Each DHRS train has an inlet screen, an inlet line that connects the cooling pool to the main feedwater line, an inlet valve, an inlet isolation valve, an outlet line, an outlet isolation valve, and a vent sparger on the outlet. In addition, each feedwater line includes a pre-pressurized, water-filled accumulator to provide continual feedwater flow during natural circulation startup of the DHRS.

2.3.7 Containment Heat Removal System

Following a loss-of-coolant accident (LOCA), the containment heat removal system (CHRS) rapidly reduces the containment pressure and temperature, consistent with the functioning of other associated systems, and maintains them at acceptably low levels for extended periods of time. The CHRS, an engineered safety feature, is classified as a "system," even though it only consists of the containment cooling pool water and containment vessel. The containment cooling pool consists of a large, below-grade concrete pool that is designed to provide stable, ample cooling for the containment for at least 72 hours following any LOCA without any active heat removal from the pool. Following a postulated break in the primary system, steam released into the containment would be condensed on the inside surface of the containment wall, which, in turn, would be passively cooled by conduction and convection heat transfer with the cooling pool. Because the containment would be evacuated during normal operation, a low level of noncondensable gases would be present inside the containment, and condensation heat transfer rates would increase.

2.3.8 Emergency Core Cooling System

NuScale's emergency core cooling system (ECCS), an engineered safety feature, consists of two independent reactor vent valves (RVVs), two independent reactor recirculation valves (RRVs), and the CHRS. The ECCS provides a means of core decay heat removal in the event of loss of the main feedwater flow in conjunction with the loss of both trains of the DHRS. Long-term cooling is established via reactor recirculation cooling via the ECCS flow path in the event of a LOCA.

The ECCS is initiated by opening the RVVs and RRVs. Opening these valves creates a path by which water condensed on the inside surface of the containment flows into the reactor coolant system via the RRVs. Opening the RVVs establishes a natural circulation path whereby water that is boiled in the core leaves through the RVVs, is condensed and collected in the containment, and then is reintroduced into the downcomer through the RRVs.

3.0 POSTULATED ACCIDENT SCENARIO DESCRIPTION

3.1 Scenario Selection

Regulations require that all potential break sizes and locations be analyzed when evaluating the performance of the ECCS systems (10 CFR 50.46). Due to the large range of break sizes and locations of a standard pressurized water reactor, system processes and related phenomena are present at one portion of the break spectrum, but are not present at other portions. Since an analysis must be appropriate for the transient being analyzed, separate evaluation methodologies typically exist for the spectrums of both large-break sizes (i.e., large-break loss-of-coolant accidents [LBLOCAs]) and small-break sizes (SBLOCAs).

The NuScale design integrates all of the large primary piping within the reactor pressure vessel; therefore, the range of breaks is restricted to the spectrum of small-break sizes. The range of small-break sizes that are possible in the NuScale design eliminates the potential phenomena that are specific to an LBLOCA. Specifically, the processes of core uncovery, refilling, reflooding, and quenching are not present for any break size possible in the NuScale design. Because these processes are present in an LBLOCA event, the lack of their possibility in the NuScale design precludes the LBLOCA class of accidents; only the SBLOCA class of LOCA accidents is possible. This PIRT addresses this scenario by considering a generic sequence of events that would be characteristic of the NuScale plant design under SBLOCA conditions.

LOCA variation in the NuScale design is restricted by the limited number of penetrations of the primary system pressure boundary. Specifically, the only penetrations large enough to exceed makeup capacity are the penetrations for the ECCS system and the chemical volume and control system. Additionally, the flow area of all the ECCS valves and the chemical volume and control system are similar. This simplified condition accompanies a generic SBLOCA scenario, which was chosen in a manner consistent with the inadvertent operation of one of the ECCS valves or a break in the chemical volume and control system. The sequence of events for the generic scenario is provided in Section 3.2.

To aid understanding of the progression of the NuScale system during the generic scenario, a sample calculation was carried out using the RELAP5 systems code (Reference 5). A detailed description of the sample calculation is provided in Appendix A of the supporting PIRT engineering report (Reference 4). This sample calculation was based on preliminary information available at the time of the PIRT. Detailed design efforts to date have not introduced sufficient changes to the major characteristics of the design such that the identified phenomenon and related rankings would be expected to change.

In addition, even though the panel considered the operation of the DHRS, it was not included in the sample calculations presented to the panel. This was done because the primary system response is enhanced by lack of the DHRS; thus, removal of this system from the calculations resulted in greater fluctuations in the phenomena, making them more apparent.

3.2 Sequence of Events

This SBLOCA scenario is initiated by inadvertent actuation of one of the RVVs on top of the reactor vessel. Following the initiating event, the steam generators are isolated and the remaining RVV opens on low primary pressure. One RRV is assumed to open after a 75-second delay. The reason for making only a single RRV operable for the representative SBLOCA scenario was to model a single failure of a safety component based on this failure limiting reintroduction of coolant into the primary system. A full description of the representative calculation is presented in Appendix A of the PIRT engineering report (Reference 4). The following sequence of events describes the postulated representative SBLOCA.

Table 3-1. Sequence of Events

Time (seconds)	Sequence of Events
0	Inadvertent actuation of one RVV (flow area = 0.00456 square meters, equivalent to a 3-inch diameter pipe)
0	Loss of containment vacuum due to primary fluid entering containment space
0	Containment flooding
0	Reactor Protection System trip signal (on loss of containment vacuum)
0	Turbine trip and main steam line isolation
0	Main feed pump trip and main feed line isolation
3	Reactor trip (assumed 3 seconds to full control rod insertion)
5	DHRS spargers open (coincident with main steam isolation valve (MSIV) isolation indication, Reactor trip)
5	DHRS Sumps Opens (coincident with MSIV isolation indication, Reactor trip)
~ 25	Remaining RVV opened on low primary pressure (assumed to be 6 MPa or 870 psia)
75	One RRV opened
~100	Long-term cooling (pressure equilibrium between primary and containment)

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 the information to meet some decision-making objective. It has been applied to many nuclear technology issues to help guide research and develop regulatory requirements. The process can be broken into nine distinct steps: issues (Step1), objectives (Step 2), hardware and scenario (Step 3), evaluation criteria (Step 4), knowledge base (Step 5), identify phenomenon (Step 6), importance ranking (Step 7), knowledge-level ranking (Step 8), and documentation (Step 9). A high-level description of the components is provided in the following sections, and a complete discussion of the process used during the PIRT development is detailed in the supporting PIRT engineering report (Reference 4).

4.1 PIRT Panel Biographies

The following persons participated either as panel members, panel chair, or panel facilitator. Biographical information for each participant is provided in Appendix A.

- Dr. Lawrence E. Hochreiter
- Dr. Mujid S. Kazimi
- Dr. Brent Boyack
- Dr. Kord S. Smith
- Dr. José N. Reyes
- Dr. Graham Wallis
- Dr. Kent B. Welter
- Dr. Eric P. Young

4.2 Importance and Knowledge-Level Rankings

The expert panel ranked the phenomena in Table 4-5 relative to one or more evaluation criterion or figures of merit (Section 4.5), e.g., collapsed liquid level Each phenomenon is assigned one of the importance rankings described in Table 4-1 below:

Table 4-1. Importance Rankings

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

The expert panel assessed and ranked the current knowledge level for high-ranked phenomena. Numerical values were assigned to reflect knowledge levels and adequacy of data, and analytical tools were used to characterize the phenomena. Table 4-2 presents the scale used for knowledge levels:

Table 4-2. Knowledge Levels

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

4.3 Scenario Periods

The PIRT panel divided the SBLOCA scenario into the following three periods:

- Accident initiation and blowdown (Period 1)
- RVV actuation (Period 2)
- Long-term cooling (Period 3)

In addition to the representative SBLOCA scenario detailed in Appendix A of the PIRT engineering report (Reference 4), a number of additional full-plant RELAP5 calculations were completed at the request of the panel to clarify the effects of variations to the representative scenario. These calculations included changes to the time delay before operation of the remaining RVV and RRV, as well as changing the initiating event to inadvertent operation of a single RRV. These calculations were performed interactively during the PIRT panel discussions.

4.4 Phenomena-Specific System Breakdown

Accident phenomena are identified based on their influence on the safety of the plant during the considered event. This is accomplished by identifying phenomena with respect to their relative influence on the selected figures of merit (see Section 4.5). To aid in the identification of phenomena, the expert panel divided the NuScale module into the systems, subsystems, components, and processes presented in Table 4-3.

Table 4-3. Phenomena-Specific Systems, Subsystems, Components, and Processes

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4.5 Figures of Merit

The expert panel identified the figures of merit described in Table 4-3 for use in ranking phenomena: *critical heat flux (CHF) (for Period 1); collapsed liquid level or CHF (for Period 2); and core coolability (for Period 3).*

Table 4-4. Figures of Merit

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4.6 PIRT Tables with Rankings

Once the figures of merit were selected (Section 4.5) and the module was divided into the appropriate systems, components, and processes (Section 4.4), the panel identified the important phenomena. Table 4-5 lists all of the phenomena identified by the expert panel and shows their associated importance ranking (H=high; M=medium; L=low; I=inactive) for each period with respect to the figures of merit. The importance ranking for each phenomenon was used to determine the importance ranking for each component. A simple average of the rankings for each phenomenon identified for a component during each period of the accident was used to determine the component-level rankings.

In addition, Table 4-5 includes knowledge-level (KL) rankings for all highly important phenomena. The colors in Table 4-5 represent knowledge-level rankings of highly important phenomena: green for fully known (Reference 3), yellow for partially known, and red for very limited knowledge (Reference 2). No highly important phenomenon had a knowledge-level ranking below 2. Those phenomena not highlighted in a color are considered to be of low or medium importance for all three periods.

Table 4-5. Preliminary NuScale Module LOCA PIRT Results

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Table 4-5. Preliminary NuScale Module LOCA PIRT Results (Continued)

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Table 4-5. Preliminary NuScale Module LOCA PIRT Results (Continued)

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Table 4-5. Preliminary NuScale Module LOCA PIRT Results (Continued)

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Table 4-5. Preliminary NuScale Module LOCA PIRT Results (Continued)

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Table 4-5. Preliminary NuScale Module LOCA PIRT Results (Continued)

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5.0 DISCUSSION OF PHENOMENA AND RANKING RATIONALES

The following sections summarize the expert panel discussion for each phenomenon that is ranked highly important in at least one component.

5.1 Core Phenomena

5.1.1 Fuel Rods

The NuScale module uses half-height, standard-configuration, pressurized-water-reactor-type fuel. The panel considered the fuel rod component to include the uranium oxide fuel, gap, and zirconium alloy cladding.

5.1.1.1 Decay Heat

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5.1.1.2 Fission Power

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5.1.1.3 Stored Energy

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5.1.1.4 Axial and Radial Power Shapes (Local Power)

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5.1.1.5 Boron Plate-Out on Fuel

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5.1.2 Core Subchannel Flow

5.1.2.1 Single-Phase Pressure Drop

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5.1.2.2 Two-Phase Pressure Drop

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5.1.2.3 Flashing

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5.1.2.4 Natural Circulation Flow/Bulk Flow

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Table 5-1. Component and Phase Natural Circulation Flow Rankings for Components and Three LOCA Phases

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5.1.2.5 Interfacial Drag/Relative Motion Between Phases

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5.1.2.6 Single-Phase Convection Heat Transfer

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5.1.2.7 Two-Phase Convection Heat Transfer

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5.1.2.8 Nucleate Boiling

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5.1.2.9 Critical Heat Flux Correlations

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5.1.2.10 Pool Boiling (No Flow)

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5.1.2.11 Flow Regime Transition

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5.1.2.12 Spacer Grid Effects on Heat Transfer

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5.1.2.13 Cross-Flow/Mixing

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5.1.2.14 Void Distribution

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5.1.2.15 Initial Temperature Distribution

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5.1.2.16 Boron Blockage in Core Subchannels

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5.2 Primary System Phenomena

5.2.1 Downcomer

5.2.1.1 Primary Natural Circulation Flow/Bulk Flow

See discussion of primary natural circulation/bulk flow in Section 5.1.2, "Core Subchannel Flow."

5.2.2 Hot Leg Riser

5.2.2.1 Flashing

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5.2.2.2 Two-Phase Level Swell

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5.2.2.3 Two-Phase Pressure Drop

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5.2.2.4 Primary Natural Circulation Flow/Bulk Flow

See discussion of primary natural circulation/bulk flow in Section 5.1.2, “Core Subchannel Flow.”

5.2.2.5 Interfacial Drag/Relative Motion of Phases

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5.2.3 Lower Plenum

5.2.3.1 Primary Natural Circulation Flow/Bulk Flow

See discussion of primary natural circulation/bulk flow in Section 5.1.2, ”Core Subchannel Flow.”

5.2.4 Cold Leg (Steam Generator Annulus)

5.2.4.1 Two-Phase Pressure Drop

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5.2.4.2 Primary Natural Circulation Flow/Bulk Flow

See discussion of primary natural circulation/bulk flow in Section 5.1.2, ”Core Subchannel Flow.”

5.2.5 Break

5.2.5.1 Single-Phase Pressure Drop

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5.2.6 Two-Phase Pressure Drop

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5.2.6.1 Choked Flow

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5.3 Containment Heat Removal System Phenomena

5.3.1 Containment

5.3.1.1 Condensation Heat Transfer

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5.3.1.2 Single-Phase Convection Heat Transfer (Vessel and Containment)

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5.3.1.3 Conduction Heat Transfer (Vessel and Containment)

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5.3.2 Reactor Recirculation Valves

5.3.2.1 Single-Phase Pressure Drop

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5.3.3 Containment Cooling Pool

5.3.3.1 Single-Phase Convection Heat Transfer

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5.3.4 Intact Reactor Vent Valve

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5.3.4.1 Single-Phase Pressure Drop

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5.3.4.2 Two-Phase Pressure Drop

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5.3.4.3 Choked Flow

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5.3.5 Concrete/Stainless Steel Liner of Containment Cooling Pool

5.3.5.1 Single-phase Convection Heat Transfer

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5.3.5.2 Conduction Heat Transfer

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5.4 Decay Heat Removal System Phenomena

5.4.1 Steam Generator Tubes

5.4.1.1 Void Distribution

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5.4.1.2 Single- and Two-Phase Convection Heat Transfer

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5.4.1.3 Flashing

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5.4.1.4 Feedwater Accumulators

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5.4.2 Instrumentation

5.4.2.1 Safety Signal Sequence

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6.0 CONCLUSIONS

The expert thermal-hydraulics and neutronics panel completed a PIRT for all three phases of a postulated SBLOCA in the NuScale module. For this analysis, the NuScale module was divided into four major sections: core, primary system, CHRS, and DHRS. The panel further identified components, systems, and subsystems. After specifying the CHF, core collapsed liquid level, and core coolability as figures of merit, safety-related phenomena were identified and ranked according to their relative importance to the three figures of merit previously listed for each phase of the SBLOCA. The results of these phenomena rankings were recorded in a PIRT table.

Of the approximately 160 phenomena identified, 54 (34 percent) were identified as highly important for at least one phase of the SBLOCA. None of the phenomena identified was ranked high in importance for all three phases of the SBLOCA. The panel provided knowledge rankings for these 54 highly ranked phenomena, ranging from 2 (partially known) to 4 (fully known); the large majority were ranked as 3 (fairly well known). However, the following 16 highly important phenomena were assigned a relatively lower knowledge-level ranking of 2.

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Development of this PIRT early in the detailed design process provides the opportunity for the report results to influence the design and scaled test program. This review is an excellent way to validate the unique safety features of the NuScale module before a design is finalized. In addition to providing information to help influence the design, the PIRT results will help NuScale prioritize and focus additional research efforts as required. Results of this PIRT are commensurate with the level of design and safety analysis work that has been completed. The PIRT will be periodically revised to reflect new data affecting the original importance rankings and knowledge levels assigned to each phenomenon. In some cases, certain phenomena originally ranked high in importance may be shown to have had only moderate or little influence on the selected figure of merit. Conversely, some phenomena originally ranked low in importance may be shown to be more important. Results from future experimental programs and safety analyses will be integrated into a final version of this PIRT.

7.0 REFERENCES

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4. NuScale Power, 2010, *NP-RF-03RP-003, NuScale Preliminary Loss-Of-Coolant-Accident Phenomena Identification and Ranking Table (PIRT)* Corvallis, Oregon, February. *[Proprietary]*

5. U.S. Nuclear Regulatory Commission, 2001, *RELAP5 MOD 3.3 Code Manual, Volume IV: Models and Correlations*, NUREG/CR-5535 Revision 1, Rockville, Maryland, December.

APPENDIX A: 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.

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, Inc., 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, 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.